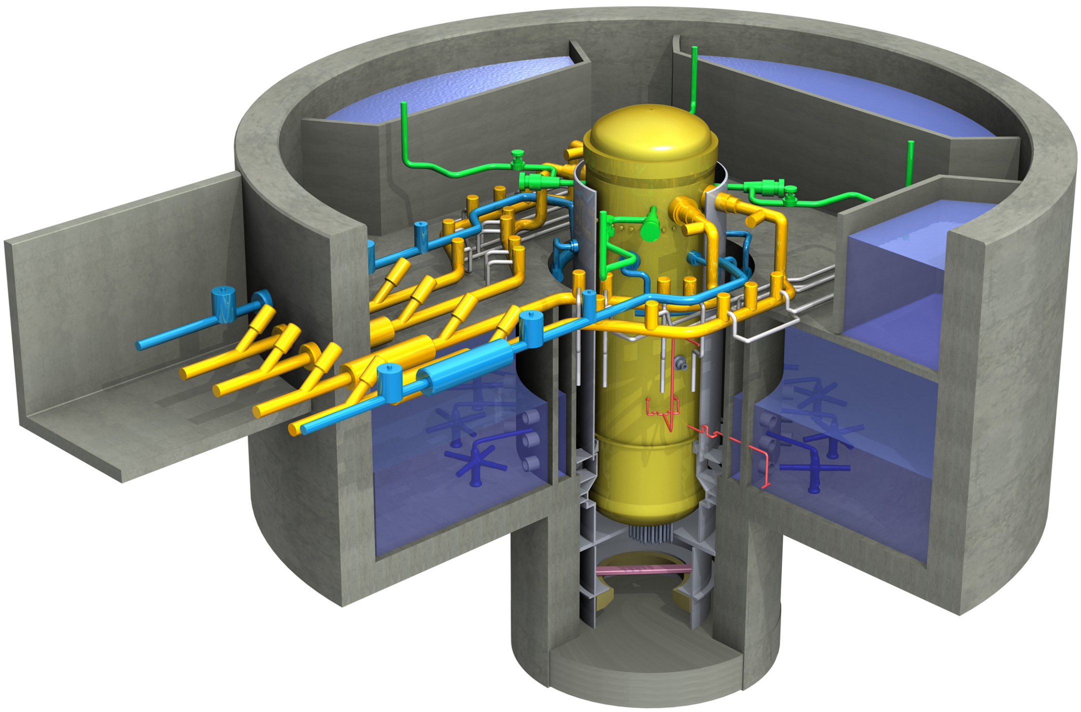
GE-Hitachi Nuclear Energy

26A6642BP
Revision 4
September 2007



ESBWR Design Control Document
Tier 2

Chapter 15
Safety Analyses

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Global Abbreviations And Acronyms List

Term	Definition
10 CFR	Title 10, Code of Federal Regulations
ABWR	Advanced Boiling Water Reactor
ADS	Automatic Depressurization System
AEC	Atomic Energy Commission
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ASD	Adjustable Speed Drive
ASME	American Society of Mechanical Engineers
AST	Alternate Source Term
ASTM	American Society of Testing Methods
AT	Unit Auxiliary Transformer
ATLM	Automated Thermal Limit Monitor
ATWS	Anticipated Transients Without Scram
BOP	Balance of Plant
BPV	Bypass Valve
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
C&I	Control and Instrumentation
CBHVAC	Control Building HVAC
CFR	Code of Federal Regulations
CMU	Control Room Multiplexing Unit
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CR	Control Rod
CRD	Control Rod Drive
DBA	Design Basis Accident
DBE	Design Basis Event
DCIS	Distributed Control and Information System
DG	Diesel-Generator
DPS	Diverse Protection System
DPV	Depressurization Valve
DW	Drywell
EBAS	Emergency Breathing Air System
ECCS	Emergency Core Cooling System
EOC	Emergency Operations Center
EOC	End of Cycle

Term	Definition
ESF	Engineered Safety Feature
FAPCS	Fuel and Auxiliary Pools Cooling System
FATT	Fracture Appearance Transition Temperature
FB	Fuel Building
FHA	Fire Hazards Analysis
FMEA	Failure Modes and Effects Analysis
FTDC	Fault-Tolerant Digital Controller
FW	Feedwater
FWCS	Feedwater Control System
GDC	General Design Criteria
GDCS	Gravity-Driven Cooling System
GE	General Electric Company
GEEN	GE Energy Nuclear
GEH	GE – Hitachi Nuclear Energy
GENE	GE Nuclear Energy
GWSR	Ganged Withdrawal Sequence Restriction
HCU	Hydraulic Control Unit
HP	High Pressure
HP_CRD	High Pressure Control Rod Drive
HSSS	Hardware/Software System Specification
HVAC	Heating, Ventilation and Air Conditioning
HVS	High Velocity Separator
HWS	Hot Water System
HX	Heat Exchanger
I&C	Instrumentation and Control
I/O	Input/Output
IC	Ion Chamber
IC	Isolation Condenser
ICD	Interface Control Diagram
ICS	Isolation Condenser System
IEEE	Institute of Electrical and Electronic Engineers
LCO	Limiting Conditions for Operation
LD	Logic Diagram
LOCA	Loss-of-Coolant-Accident
LOFW	Loss-of-feedwater
LOOP	Loss of Offsite Power
LP	Low Pressure
LPCRD	Locking Piston Control Rod Drive
LPRM	Local Power Range Monitor
MCPR	Minimum Critical Power Ratio

Term	Definition
MCR	Main Control Room
MLHGR	Maximum Linear Heat Generation Rate
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSL	Main Steamline
MSLB	Main Steamline Break
MSLBA	Main Steamline Break Accident
MT	Main Transformer
MTTR	Mean Time To Repair
NBR	Nuclear Boiler Rated
NMS	Neutron Monitoring System
NOV	Nitrogen Operated Valve
NRC	Nuclear Regulatory Commission
NRHX	Non-Regenerative Heat Exchanger
NSSS	Nuclear Steam Supply System
OLMCPR	Operating Limit Minimum Critical Power Ratio
PAS	Plant Automation System
PCC	Passive Containment Cooling
PCCS	Passive Containment Cooling System
PCT	Peak cladding temperature
PRA	Probabilistic Risk Assessment
PT	Pressure Transmitter
RB	Reactor Building
RBS	Rod Block Setpoint
RC&IS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RDA	Rod Drop Accident
RFP	Reactor Feed Pump
RG	Regulatory Guide
RHX	Regenerative Heat Exchanger
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRPS	Reference Rod Pull Sequence
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling
RWE	Rod Withdrawal Error
SAR	Safety Analysis Report
S/P	Suppression Pool
SB&PC	Steam Bypass and Pressure Control System
SBO	Station Blackout

Term	Definition
SBWR	Simplified Boiling Water Reactor
SCRRI	Selected Control Rod Run-in
SDC	Shutdown Cooling
SDS	System Design Specification
SIL	Service Information Letter
SIT	Structural Integrity Test
SLC	Standby Liquid Control
SRNM	Startup Range Neutron Monitor
SRI	Select Rod Insert
SRP	Standard Review Plan
SRV	Safety Relief Valve
SSE	Safe Shutdown Earthquake
TAF	Top of Active Fuel
TBS	Turbine Bypass System
TBV	Turbine Bypass Valve
TGCS	Turbine Generator Control System
TG	Turbine Generator
TS	Technical Specification(s)
TSV	Turbine Stop Valve
USNRC	United States Nuclear Regulatory Commission
VDU	Video Display Unit
WW	Wetwell

15 SAFETY ANALYSES

This chapter addresses all ESBWR plant safety analyses. The details of most of the safety analyses are contained within this chapter, however, per the Regulatory Guide 1.70 format, some safety analyses are addressed in detail in other DCD Tier 2 chapters (e.g. emergency core cooling system [ECCS] performance is addressed within Section 6.3). For those safety analyses not addressed in detail in Chapter 15, references are provided to their locations within Tier 2.

15.0 ANALYTICAL APPROACH

In this chapter, the effects of anticipated process disturbances and postulated component failures are examined to determine their consequences and to evaluate plant capabilities to control or accommodate such failures and events. System response analyses are based upon the equilibrium core loading shown in Reference 15.0-6, and are used to identify the limiting events for the ESBWR. Other fuel designs and core loading patterns, developed in compliance with Appendix 4B, similar to the one shown in Reference 15.0-6, do not significantly affect the sensitivities demonstrated by this study. Evaluation of these limiting events for each plant fuel cycle ensures that the criteria in Appendix 4B.5 are met.

The results of the system response analyses for the initial core loading documented in Reference 15.0-7 are provided in Reference 15.0-8. System response analyses bounding operation in the feedwater temperature operating domain are documented in Reference 15.0-9.

GEH has developed a unique systematic approach to plant safety consistent with the GEH boiling water reactor (BWR) technology base. The key to the GEH approach to plant safety is the Section 15.1 Nuclear Safety Operational Analysis (NSOA). A NSOA is a system level qualitative failure modes and effects analysis (FMEA) of plant protective functions to show compliance with the events addressed in Chapter 15. Key inputs into the NSOA are derived from the applicable regulations, through industry codes and standards, and the plant safety analyses.

In Section 15.1, all unacceptable safety results and all required safety actions are identified. In addition, an evaluation of the entire spectrum of events is consistently carried out for the ESBWR to demonstrate that a consistent level of safety has been attained.

The NSOA acceptance criteria are based on the Title 10 of the Code of Federal Regulations (10 CFR regulations) and the NUREG-0800 Standard Review Plan (SRP) acceptance criteria.

The starting point for the NSOA is the regulatory acceptance criteria and design code allowables such that the acceptability of safety analysis results can be determined. This concept enables the results of any safety analysis to be compared to applicable criteria. Unacceptable safety results represent an extension of the nuclear design criteria for plant systems and components that are used as the basis for system design. The unacceptable safety results have been selected so that they are consistent with applicable regulations and industry codes and standards.

The focal point of the NSOA is the event analysis, in which all safety-related protection sequences are evaluated until all required safety actions are successfully completed. The event analysis identifies all required front-line safety systems and their safety-related auxiliaries.

The full spectrum of initial conditions limited by the constraints placed on planned operation is evaluated. All events are analyzed until a stable condition is obtained. This ensures that the event being evaluated does not have an unevaluated long-term consideration.

In the event analysis, all safety-related systems, operator actions and limits to satisfy the required safety actions are identified. Limits are derived only for those parameters continuously available to the operator. Credit for operator action is taken only when an operator can be reasonably expected to perform the required action based on the information available to him.

In the NSOA, a complete and consistent set of safety actions (i.e., those required to prevent unacceptable results) has been developed. For all of the events that are evaluated, a single-failure-proof path to plant shutdown is identified. The application of the 10 CFR 50, Appendix A single-failure criterion (SFC) to these events is imposed as an additional measure of conservatism in the NSOA process.

15.0.1 Classification and Selection of Events

The classification of events for the ESBWR is primarily based on the classifications and terms used in the 10 CFR regulations because:

- The 10 CFR regulations have authority over all other document types;
- The SRP and Regulatory Guide (RG) 1.70 do not provide specific definitions for all versions of abnormal event categories;
- The SRP and RG 1.70 do not use the same terminology for the non-accident abnormal events, and thus, the non-accident abnormal event classifications within the SRP and RG 1.70 could be misinterpreted;
- The non-accident abnormal event classification terms in the SRP and RG 1.70 are not the same as the abnormal event classifications in the 10 CFR regulations;
- The 10 CFR regulations do specifically define an Anticipated Operational Occurrence (AOO), Loss-of-Coolant Accident (LOCA), Anticipated Transient Without Scram (ATWS), normal operation, design basis events, and a number of associated terms; and
- The use of terms is more consistent within the 10 CFR regulations than in the SRP or RG 1.70.

The most recently certified BWR (i.e., the ABWR) licensing documents are used for additional guidance.

The design basis events (DBEs) in the 10 CFR regulations assume an initiating event (and any resultant failures) with or without a single active component failure or operator error. The postulating of design basis events that assume a failure beyond the SFC or a common-mode failure is not specifically required by the 10 CFR regulations. However, the 10 CFR regulations do require evaluations of three specific event scenarios, i.e., Safe Shutdown Fire, Station Blackout (SBO) and ATWS, and some of these event scenarios do assume failures beyond the single failure criterion (SFC) and/or common-mode failures. Therefore, these events should not be classified as DBEs. However, their safety analyses are included in the ESBWR DCD.

Historically, DBEs should have annual probabilities $\geq 10^{-6}$. Therefore, any event with an annual probability of $< 10^{-6}$ is not considered credible, and thus, is not classified as a DBE.

The 10 CFR regulations, SRP and RG 1.70 postulate events that (for the ESBWR with its advanced design features and additional redundancy) require failures beyond the SFC and/or require common-mode failures. Those events are included in the ESBWR DCD, but not as DBEs.

Per the 10 CFR regulations, AOOs are expected to occur once in a plant's lifetime, while accidents are low probability events that are not expected to occur during a plant's lifetime. Because the ESBWR has a design life of 60 years, any abnormal event that has an annual probability of occurrence ≥ 1/60 could be classified as an AOO. However, historically, a value of > 1/100 has been used.

Based on the 10 CFR regulations, the SRP or an NRC reviewed Licensing Topical Report (LTR), the safety analysis acceptance criteria for each of the special events is developed on an event-specific basis.

The 10 CFR regulations consistently refer to any failure of a fission product barrier that results in an offsite radiological consequence as an accident.

15.0.1.1 Approach For Determining Event Classifications

(1) Per the 10 CFR regulations, the 10 CFR 50 App. A definitions, GDC, the 10 CFR 50.49 design basis event definition, SRP 6.1.1, SRP 15.0.1, RG 1.183 and guidance from events addressed in the SRP;

   a. Divide the types of events as DBEs, and by exclusion, all other events as special events,

   b. Provide the basis for which events should be classified as AOOs,

   c. Provide the basis for a (non-AOO and non-accident) event classification for events with lower probabilities than AOOs, but for conservatism have historically not been treated or classified as accidents, and

   d. Generate the criterion for determining which type of accidents shall be classified as design basis accidents (DBAs), and by exclusion, all other accidents are not DBAs.

(2) Per the regulatory definition of an AOO (event probability), historical information and guidance from the SRP determine specific criteria for classifying events as AOOs.

(3) Based on (a) the 10 CFR regulations associating accidents with radiological consequences, (b) application of SFC, (c) SRP and RG 1.70 guidance for the types of events that should be addressed in Chapter 15, (d) SRP acceptance criteria for transient/AOO events that result in fuel failure, and (e) historical consistently used terms, generate a classification term and criteria for determining non-AOO and non-accidents events, which (a) should be treated as design basis events and (b) result from an initiating event with or without assuming a single active component failure or single operator error. Include this new DBE term in the DBE classifications.

(4) Based on the 10 CFR regulations, SRPs, RG 1.183 generate a definition for an accident.

(5) Based on (a) reviewing the 10 CFR regulations that have added other abnormal events (e.g., ATWS, SBO, Safe Shutdown Fire), (b) that DBEs do not include common-mode failures and/or additional failure(s) beyond the SFC, (c) reviewing the SRP events that include common-mode failures and/or failure(s) beyond the SFC, and (d) historically evaluated

non-DBE events and used associated classification terms, generate classification term for non-DBEs that are addressed in the DCD Chapter 15.

15.0.1.2 Results of Event Classification Determinations

Table 15.0-1 provides the results of the event classifications in the form of a determination criterion vs. event classification matrix. Table 15.0-1 is based on the results from the following evaluation.

(1) a. Per 10 CFR 50.49, and the fact that the SRP treats all postulated abnormal initiating events with or without assuming a single active component failure or single operator error as if they are all design basis events, the following are classified as design basis events:

- Normal operation, including AOOs;
- Infrequent events [see Item (3) for additional details];
- Accidents;
- External events; and
- Natural phenomena.

b. AOOs, by definition, are classified as part of normal operations, do not have radiological consequences (except if in combination with an additional single active component failure or single operator error), have more restrictive acceptance criteria (e.g., GDC 10 or 10 CFR 20 vs. 10 CFR 50.34) than accidents, and thus, are not accidents and shall not be treated as accidents.

c. A classification term for events with lower probabilities than AOOs, but for conservatism should not be treated as accidents, should be provided.

d. Except for AOOs, the 10 CFR regulations, SRP and RG 1.70 do not explicitly or implicitly apply any quantitative event frequency criterion for defining any other abnormal event classification. Therefore, event frequencies should not be used to determine accident type event classifications.

SRP 6.1.1, SRP 15.0.1 and RG 1.183 are consistent in categorization of DBAs. A DBA is an accident postulated and analyzed to confirm the adequacy of a plant engineered safety feature.

By exclusion, all other accidents are not classified as DBAs.

(2) An AOO *is any abnormal event that has an event probability of ≥ 1/100 per year*.

(3) The other (non-AOO and non-DBA) postulated abnormal events are classified as "*infrequent events*." An infrequent event *is defined as a DBE (with or without assuming a single active component failure or single operator error) with probability of occurrence of < 1/100 per year, and a radiological consequence less than a design basis accident.*

(4) The other (non-AOO and non-infrequent incident) DBEs should be classified as accidents with DBAs as a subset. An accident *is defined as a postulated DBE that is not expected to occur during the lifetime of a plant, which equates to either an ASME Code Service Level C*

or D incident, and results in radioactive material releases with calculated doses comparable to (but not to exceed) the 10 CFR 50.34(a) exposures.

(5) Historically, non-DBEs that are evaluated in BWR safety analysis reports or DCD have been termed as "special events." As no better term has been specified in a regulatory document, it is judged reasonable to maintain that term in the ESBWR DCD.

Special events

a. Are not included as design basis events in 10 CFR 50.49, and

i. are postulated in the 10 CFR regulations to demonstrate some specified prevention, coping or mitigation capabilities, without specifically requiring a radiological evaluation, and/or

ii. include a common mode equipment failure or additional failure(s) beyond the SFC.

Note: Special events do not include severe accidents or other events that are only evaluated as part of the plant PRA.

Because of the ESBWR's advanced engineering and additional redundant features, some of the abnormal events for earlier plants are classified differently for the ESBWR.

15.0.2 Abnormal Events To Be Evaluated

In selecting the AOOs to be analyzed as part of the plant safety analysis, the nuclear system parameter variations listed below are considered possible initiating causes of challenges to the fuel or the reactor coolant pressure boundary (RCPB).

- Decrease in Core Coolant Temperature
- Increase in Reactor Pressure
- Increase in Reactor Coolant Inventory
- Decrease in Reactor Coolant Inventory

The AOOs considered in the ESBWR safety analyses are listed in Table 15.0-2.

The parameter variations listed above include all the effects within the nuclear system (caused by AOOs) that can challenge the integrity of the reactor fuel or RCPB. The variation of any one parameter may cause a change in another parameter. However, for analysis purposes, challenges to barrier integrity are evaluated by groups according to the parameter variation initiating the plant challenge and which typically dominates the event response.

The potentially limiting AOOs are identified in Subsection 15.2.6.

The infrequent events considered in the ESBWR safety analyses are listed in Table 15.0-2, and are discussed in detail in Section 15.3. These consist of reactivity, power and pressure anomalies such as the Control Rod Withdrawal Error, the Loss of Feedwater Heating With Failure of Selected Control Rod Run-In and Generator Load rejection with Total Turbine Bypass Failure. The potentially limiting infrequent events are identified in Subsection 15.3.17.

The accidents considered in the ESBWR safety analyses are listed in Table 15.0-2.

The following accidents pose the most limiting challenge to plant design and radiological exposure limits:

- Loss of Coolant Accident (LOCA) Inside Containment
- Main Steamline Break Outside Containment
- Fuel Handling Accident

The LOCA is re-evaluated as part of the process for establishing the core operating limits for new fuel types.

Each of the accidents listed in Table 15.0-2 is discussed in detail in Section 15.4.

The special events evaluated as part of the ESBWR safety analysis are listed in Table 15.0-2, and discussed in detail in Section 15.5. The Main Steamline Isolation Valve (MSIV) Closure With Flux Scram (Overpressure Protection) event is re-evaluated for each reload, to ensure that the Reactor Coolant System Pressure Safety Limit in the Technical Specifications cannot be exceeded by any Design Basis Event. Additional special events that require re-evaluation are identified in the analysis subsections for those events.

The computer codes used in each event analysis are listed in Table 15.0-8.

15.0.3 Determination of Safety Analysis Acceptance Criteria

Where acceptance criteria are specified in the 10 CFR regulations, those criteria or their equivalent SRP criteria shall be used. However, if an acceptance criterion in the SRP conflicts with the associated acceptance criterion in a regulation, then the criterion specified in the regulation is used. Where an acceptance criterion is not specified in the 10 CFR regulations, then the criterion in the SRP is used. Where an acceptance criterion is not specified in regulations nor the SRP, then the criterion is developed primarily based on a review of the regulations, and secondarily based on reviews of regulatory guide(s) and the SRP.

A listing of the ESBWR abnormal events and their event classifications and relevant SRP section is provided in Table 15.0-2. Table 15.0-2 is subject to change due to future probabilistic analyses or regulatory considerations, and thus, may be revised in the future.

15.0.3.1 Anticipated Operational Occurrences

For AOOs, the GDC 10 acceptance criterion is that "The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences." To meet the intent of GDC 10, SRP 15.1 and SRP 15.2, detailed acceptance criteria for AOOs both not in combination and in combination with an additional single active component failure (SACF) or single operator error (SOE) are provided. For an AOO, which is not in combination with an additional SACF or SOE, the SRP 15.1 and SRP 15.2 criterion is "Fuel cladding integrity shall be maintained by ensuring that the minimum CPR remains above the MCPR safety limit for BWRs based on acceptable correlations." This is equivalent to the fuel cladding integrity (greater than 99.9% of the fuel rods in the core would be expected to avoid boiling transition) safety limit (SL) included in the ESBWR Technical Specifications (TS).

A SACF or SOE is a non-coincidental failure/error that is independent of the initiating fault that caused the AOO. For an AOO in combination with an additional SACF or SOE, the SRP 15.1 and SRP 15.2 criterion is "fuel failure must be assumed for all rods for which the CPR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding." However, the SRP does not provide a specific radiological acceptance criterion, in the event that fuel cladding failures do occur. As AOOs are part of normal operation, GDC 60 and 10 CFR 20 apply.

The 10 CFR 20.1301(a)(1) 0.1 rem annual dose limit combined with (i.e., subtracting) the 10 CFR 20.1302(b)(2)(ii) 0.05 rem annual limit (for normal airborne releases) is the appropriate radiological acceptance limit for an AOO In Combination With An Additional SACF or SOE (i.e., an AOO with an additional single failure). This position is conservatively based on an assumption that an individual at the exclusion boundary annually receives 100% of the 10 CFR 20.1302(b)(2)(ii) 0.05 rem annual limit from normal operations (which is conservative, when compared to the 10 CFR 50, Appendix I 10 millirad ALARA annual airborne gamma dose guideline), and applying the 10 CFR 20.1301(a)(1) 0.1 rem annual dose limit. Therefore, the radiological acceptance criterion for an AOO with a single failure should generically be (0.1 - 0.05) 0.05 rem total effective dose equivalent (TEDE).

For AOOs, the GDC 15 acceptance criterion is that "The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences." The equivalent criterion in SRP 15.1 and SRP 15.2 is "Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values," which corresponds to the ASME Code Service Level B limit. However, for completeness the Reactor Coolant System Pressure Safety Limit in the TS should be addressed.

The SRP provides an AOO related acceptance criterion that is not addressed in GDC 10 or 15, which is "An incident of moderate frequency (i.e., an AOO) should not generate a more serious plant condition without other faults occurring independently."

Consistent with GDC 38, if an AOO involves Safety Relief Valve (SRV) or Depressurization Valve (DPV) discharge, containment and suppression pool pressures and temperatures shall be maintained below their design values.

Based on the above, Table 15.0-3 lists the DCD Chapter 15 safety analysis acceptance criteria for AOO. Except for event-specific differences, Table 15.0-4 lists the Tier 2 Chapter 15 safety analysis acceptance criteria for AOOs in combination with an additional SACF or SOE. These sets of acceptance criteria assume that all related safety analyses are performed with accepted models.

15.0.3.2 Infrequent Events

The ESBWR is designed such that any infrequent event would not result in the reactor water level dropping to below the top of the core (i.e., active fuel).

For a new plant, the 10 CFR regulations associate the consequences of postulated accidents with the exposures in 10 CFR 50.34(a)(1). Infrequent events do not result in a larger consequence than the least severe of the DBAs, and thus, their maximum radiological acceptance criteria should be ≤ 2.5 rem TEDE. However, if the SRP specifies a different or additional radiological acceptance criterion (e.g., a 10 CFR 20 limit or a different TEDE value), then the SRP acceptance criteria apply.

Based on the 10 CFR regulations and the SRP, GDC 19 is the only basis for the acceptance criterion on control room doses for all non-AOO abnormal event evaluations, such as infrequent events and accidents.

Based on ASME code classification of events with their associated stress limits and historical accepted criterion, infrequent events most closely correlate with ASME Code Service Level C limits. Therefore, reactor coolant system pressure should be based on the ASME Code Service Level C limit, which is conservatively interpreted to correspond to 120% of design pressure.

If an infrequent event results in an SRV/DPV discharge or fission product release to the containment, then containment stresses (i.e., pressures and temperatures) should be limited such that there is no loss of a containment barrier safety function, and thus, the containment must remain within its design limits/values.

Except for event-specific differences, Table 15.0-5 provides a generic set of acceptance criteria for infrequent event safety analyses.

15.0.3.3 Accidents

For a new plant, the 10 CFR regulations associate the consequences of postulated accidents with the exposures in 10 CFR 50.34(a)(1). Non-DBA accidents should not result in a larger consequence than the least severe of the DBAs, and thus, their radiological acceptance criteria should usually be limited to 2.5 rem TEDE. However, (like infrequent events) if the applicable SRP specifies a different or additional radiological acceptance criterion (e.g., a 10 CFR 20 limit or a different TEDE value), then the SRP acceptance criterion applies.

Based on the 10 CFR regulations and the SRP, GDC 19 is the only basis for the acceptance criterion on control room doses for all postulated accidents.

For the DBAs, the SRP 15.0.1 and RG 1.183 provide the consequence acceptance criteria of 2.5 rem TEDE, 6.3 rem TEDE and 25 rem TEDE [equivalent to 10%, 25% and 100% of the 10 CFR 50.34(a)(1) exposures], depending on the specific DBA. For DBAs, which do not have a consequence acceptance criterion specified in SRP 15.0.1 and/or RG 1.183, the smallest (i.e., 2.5 rem TEDE) criterion is applied.

For any accident that involves emergency core cooling system (ECCS) activation, the 10 CFR 50.46(a)(3)(b) acceptance criteria apply, and thus, the calculated changes in core geometry shall be such that the core remains amenable to cooling.

RG 1.70 classifies accidents as "limiting faults," which can be correlated to different service levels or design conditions in the applicable industry code, e.g., ASME Code Service Level C or D. To ensure conservatism and minimize the number of acceptance condition options, for DBAs, reactor coolant pressure boundary components shall be limited to ASME Code Service Level C limits.

If an accident results in an SRV/DPV discharge or fission product release to the containment, then containment stresses (i.e., pressures and temperatures) should be limited such that there is no loss of a containment barrier safety function, and thus, the containment must remain within its design limits/values.

The set of acceptance criteria for accident safety analyses is provided in Table 15.0-6.

Because radiological acceptance criteria vary for the different event scenarios, for each non-AOO design basis event scenario applicable to an ESBWR, Table 15.0-7 provides radiological acceptance criteria.

15.0.3.4 Special Events

The acceptance criteria for each of the special event safety analyses is developed on an event-specific basis, based on the coping, mitigation or acceptance criteria specified in the 10 CFR regulations, the SRP or an NRC reviewed LTR.

15.0.3.4.1 MSIV Closure With Flux Scram

For every fuel cycle, a MSIV Closure With Flux Scram analysis (commonly referred to as the Overpressure Protection Analysis) is performed. With respect to the reactor coolant pressure boundary (RCPB) pressure response, the event scenario is specifically chosen to bound all of the design basis events.

The event requires/assumes:

- An operator error, multiple equipment failures or a common mode failure cause(s) the MSIVs in all four main steamlines (MSLs) to simultaneously close;
- The two MSIV position switch circuits on three to six MSIVs fail, which causes the MSIV position scram function to fail; and
- The reactor is shutdown by a high neutron flux scram trip.

The MSIV Closure With Flux Scram analysis demonstrates that the SRVs have adequate pressure relief capacity to prevent the RCPB ASME Code Service Level B pressure limit(s) and the Reactor Coolant System Pressure Safety Limit in the Technical Specifications from being exceeded.

Therefore, this event only needs/has the following acceptance criteria:

- Pressures in the reactor coolant and main steam systems shall be maintained below 110% of their design values (i.e., not exceed ASME Code Service Level B).
- The reactor steam dome pressure shall be maintained less than or equal to the Reactor Coolant System Pressure Safety Limit in the Technical Specifications.

15.0.3.4.2 Shutdown Without Control Rods

Assuming all control rod insertion mechanisms fail, for every fuel cycle, cold shutdown core k-effective (k_{eff}) calculations are performed at various cycle exposure points, to ensure that the Standby Liquid Control (SLC) system can inject adequate (boron solution) negative reactivity into the core to allow for cold shutdown. This analysis plus the normal control rod shutdown margin calculations demonstrate compliance to GDC 26.

The Shutdown Without Control Rods event only needs/has the following acceptance criterion:

- Under the most reactive core conditions, k_{eff} shall be < 1.0.

15.0.3.4.3 Shutdown from Outside Main Control Room

A Shutdown from Outside Main Control Room safety analysis shall demonstrate that the plant can achieve and maintain safe shutdown, assuming the reactor is scrammed by the operators before they vacate the main control room.

The ability to cope with a Shutdown from Outside Main Control Room event is based on meeting the following acceptance criteria:

- Adequate reactor coolant inventory is maintained such that reactor water level is maintained above the core (i.e. top of active fuel);
- Achieve and maintain the plant to those shutdown conditions specified in plant TS as Hot Shutdown; and
- If containment isolation is involved, containment and suppression pool pressures and temperatures shall be maintained below their design values.

15.0.3.4.4 Anticipated Transient Without Scram (ATWS)

As documented in Reference 15.0-4, the generic BWR ATWS performance analysis acceptance criteria are summarized below.

- Pressures in the reactor coolant and main steam systems is maintained below ASME Service Level C limit, which is conservatively interpreted to correspond to 120% of design pressure.
- Peak cladding temperature is within the 10 CFR 50.46 limit of 2200°F.
- Peak cladding oxidation is within the requirements of 10 CFR 50.46.
- Peak suppression pool temperature does not exceed its design temperature.
- Peak containment pressure does not exceed containment design pressure.

15.0.3.4.5 Station Blackout (SBO)

An SBO safety analysis shall demonstrate that the plant can cope with the effects (i.e., with minimum equipment available) of an SBO for the duration of the SBO. The ability to cope with an SBO is based on meeting the following acceptance criteria.

- Adequate reactor coolant inventory is maintained such that reactor water level is maintained above the core (i.e. top of active fuel).
- Achieve and maintain the plant to those shutdown conditions specified in plant Technical Specifications as Hot Shutdown or Stable Condition.
- If containment isolation is involved, containment and suppression pool pressures and temperatures shall be maintained below their design values.

15.0.3.4.6 Safe Shutdown Fire

The following acceptance criteria are derived from 10 CFR Part 50.48 and Appendix R.

- Core subcriticality is achieved and maintained with adequate core shutdown margin, as specified in the plant Technical Specifications.
- Adequate reactor coolant inventory is maintained such that reactor water level is maintained above the core (i.e., top of active fuel).
- Hot Shutdown or Stable Condition is achieved and maintained.
- Cold shutdown conditions are achieved within 72 hours.
- Cold shutdown conditions are maintained thereafter.
- If containment isolation is involved, containment and suppression pool pressures and temperatures shall be maintained below their design values.
- Safety-Related and Nonsafety-Related equipment may be used to meet the above criteria.

15.0.3.4.7 Waste Gas System Leak or Failure

Because the ESBWR Offgas System pressure boundary is designed to withstand dynamic overpressure from potential hydrogen detonation of at least 17 times the normal system operating pressure, a structural failure in the Offgas System is not a credible event. For the ESBWR, the only plausible event scenario that could result in a waste gas release requires two independent operator errors and an instrumentation isolation trip or (mechanical) isolation function failure to occur, and would result in only the release of noble gases. The postulation of a Waste Gas System Failure for the ESBWR goes beyond the 10 CFR 50 Appendix A single failure criterion, and thus, it does not qualify as a design basis event. This conclusion is consistent with SRP 15.7.1, which no longer requires this event to be analyzed within Chapter 15. Therefore, the Waste Gas System Failure for the ESBWR is classified as a special event.

The radiological analysis acceptance criterion for the Waste Gas System Failure comes from Branch Technical Position (BTP) 11-5 from NUREG-0800. As is directly applicable to the ESBWR, BTP 11-5 states "the dose criterion in every case should not exceed 25 mSv (2.5 rem) at the exclusion area boundary."

15.0.3.4.8 Potential Special Events

The 10 CFR regulations and the SRP do not contain a generic set of safety analysis acceptance criteria for special events. The safety analysis acceptance criteria for these events are on an event-specific basis. It is expected that any (potential) future special event will also have event-specific safety analysis acceptance criteria.

15.0.4 Event Analysis Format

For each event, the following information is provided in Sections 15.2, 15.3, 15.4 and 15.5.

15.0.4.1 Identification of Causes

Situations that lead to the analyzed events are described in their associated event descriptions. The frequency of occurrence of each event is summarized based upon the NSOA, currently available operating plant history for the abnormal event, and the evaluations in Appendix 15A. Events for which inconclusive data exist are discussed separately within each event section.

15.0.4.2 Sequence of Events and Systems Operations

Each event evaluated is discussed and evaluated in terms of:

- A step-by-step sequence of events from initiation to final stabilized condition.
- The extent to which normally operating plant instrumentation controls are assumed to function.
- The extent to which the plant and reactor protection systems are required to function.
- The credit taken for the functioning of normally operating plant systems.
- The operation of engineered safety systems that is required.

Each event's sequence of events is supported by the NSOA. The effect of a single equipment failure or malfunction or an operator error on the event is shown in the NSOA.

15.0.4.3 Evaluation of Results

The results of the design basis events analyses are presented in Sections 15.2, 15.3 and 15.4. The limiting events can be identified, based on those results. Reasons why the other events are not limiting are given in the event documentation.

For the equilibrium core loading in Reference 15.0-6, a representative MCPR operating limit is determined**.** Results of the AOO analyses for individual plant-specific core loading patterns will differ slightly from the results shown in this chapter. However, the relative results between core associated events do not change. For the initial core loading in Reference 15.0-7, a representative MCPR operating limit is determined.

15.0.4.4 Barrier Performance

The significant areas of interest for internal pressure damage are the high-pressure portions of the RCPB (i.e., the reactor vessel and the high pressure pipelines attached to the reactor vessel).

15.0.4.5 Radiological Consequences

This subsection describes the consequences of radioactivity releases for the core loading, during DBEs. For all events whose consequences are limiting, a detailed quantitative evaluation is presented. For non-limiting events, a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

15.0.5 Single Failure Criterion

From 10 CFR 50, Appendix A:

"A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither

(1) A single failure of any active component (assuming passive components function properly) nor

(2) A single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety function.

Single failures of passive components in electric systems should be assumed in designing against a single failure."

The single failure criterion (SFC) requires the plant design to be capable of providing specific functions during any design basis event (DBE) assuming a single failure in addition to the event initiating occurrence and any other coincident failures specified in the required DBE analysis assumptions. The application of the SFC to:

- Fluid systems are described in ANSI/ANS 58.9; and
- Electrical items are described in IEEE 379.

The IEEE criteria specify that electrical systems be designed to accommodate either a passive or an active single failure. For fluid systems in DBE analyses, the SFC only applies to active failures. The SFC is applicable to:

- Emergency core reactivity control (scram);
- Emergency core cooling;
- Reactor coolant pressure boundary isolation;
- Reactor coolant system pressure relief;
- Containment cooling;
- Containment isolation;
- Containment atmosphere clean up; and
- Their required supporting functions such as cooling water and electrical power.

Only one failure needs to be assumed per plant DBE.

This subsection describes the application of single failure relative to DBEs. Single failure is defined in 10 CFR 50, Appendix A, and is specifically applied to multiple GDCs.

The treatment of plant capability evaluation events (i.e., special events) is consistent with their specific event definitions that are typically beyond the safety design bases of the plant. As a result, an additional single failure is not applied unless there is a specific licensing commitment.

15.0.5.1 Single Failures as Event Initiators

The AOOs identified in the safety analysis are frequently associated with transients that result from a single component failure or operator error, and are postulated during specific, applicable mode(s) of normal plant operation. Operator error is usually only considered as an event initiator.

Operator error is defined as a deviation from written operating procedures or operating practices. An operator error includes action(s) that are a direct consequence of one operator's single erroneous decision. An operator error does not include subsequent actions performed in response to the initiating event that resulted from the initial operator error.

Operator errors include:

- Erroneous selection and withdrawal of a control rod or control rod group.
- The manual isolation of the main steam lines caused by operator misinterpretation of an alarm or indicator.

15.0.5.2 Application of Single Failure Criterion to Event Analysis

The single-failure requirements for DBEs in the safety analysis and the NSOA are applied as follows:

- For DBEs, the protection sequences within mitigation systems are to be single-component-failure-proof. This position is in addition to any single-component failure or single operator error that is assumed as the event initiator. The requirement for assuming an additional single failure in the mitigation system adds a significant level of conservatism to the safety analysis. However, the event acceptance limits for DBEs are not changed by the application of an additional single-failure requirement.
- For AOOs, it is not always necessary to assume a single failure in normal operating systems in addition to the failure assumed as the event initiator. The basic logic for this assumption is based upon the probability of occurrence of a double failure in normal operating systems, which may be less than once per plant lifetime and exceeds the probability of occurrence definition for AOOs in 10 CFR 50, Appendix A.
- For infrequent events and accidents, single failures are considered consistent with plant-specific licensing commitments (e.g., valve malfunctions for LOCA).
- For mitigation systems included in the NSOA, single failures of active electrical and fluid components, and passive electrical components are treated in the same manner in the development of the event diagrams.
- During Technical Specifications surveillance testing or when complying with an Action, while not meeting the associated Limiting Condition for Operation (LCO), applying the SFC for affected components/systems is not required.

The single failures identified above are considered in the design of the plant, as required by specific GDC, and are utilized in the safety analysis of the specific events.

15.0.6 COL Information

None

15.0.7 References

15.0-1 (Deleted)

15.0-2 (Deleted)

15.0-3 USNRC, “Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design,” NUREG-1503, Volume 1, July 1994.

15.0-4 General Electric Company, "Assessment of BWR Mitigation of ATWS, Volume II (NUREG 0460 Alternate No. 3)." NEDE-24222, Class III (proprietary), December 1979, and NEDO-24222, Class I (non-proprietary), February 1981.

15.0-5 (Deleted)

15.0-6 Global Nuclear Fuel, "GE14 for ESBWR Nuclear Design Report", NEDC-33239-P, Class III (Proprietary), Revision 2, April 2007, NEDO-33239, Class I (Non proprietary), Revision 2, April 2007.

15.0-7 Global Nuclear Fuel, "ESBWR Initial Core Nuclear Design Report", NEDC-33326-P, Class III (Proprietary), Revision 0, July 2007, NEDO-33326, Class I (Non-proprietary), Revision 0, July 2007.

15.0-8 GE-Hitachi Nuclear Energy, "ESBWR Initial Core Transient Analyses", NEDO-33337 Class I, Revision 0, Scheduled September 2007.

15.0-9 GE-Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis", NEDO-33338 Class I, Revision 0, Scheduled September 2007.

Table 15.0-1

Chapter 15 Abnormal Event Classification Determination Matrix

Determination Criteria vs. Event Classification	**Annual Probability ≥ 10^{-2}**	**Thermal Hydraulic Basis**	**Radiological Analysis Basis**		**Assumes An Additional SACF or SOE**		**Event Not Included As A Design Basis Event in 10 CFR 50.49(b)(1)(ii) and**		
			10 CFR 20	10 CFR 50.34(a)(1) & GDC 19	Yes	No	Is Postulated In A Regulation	Assumes Common-Mode Failure(s)	Assumes Failures, Beyond SFC
AOO	X	Greater than 99.9% of the fuel rods in the core would be expected to avoid boiling transition safety limit	(Not needed)			X			
		Maintain 100% Core Coverage	X		X				
Infrequent Event		Maintain 100% Core Coverage	X*	X*	X*	X*			
Accident		10 CFR 50.46		X	X				
Special Event **		*	***	***			X * +	X * +	X * +

* Specific event dependent.

** Does not include severe accidents and other events that are only evaluated as part of the plant PRA.

*** If applicable to a specific special event.

\+ Or any combination of these conditions.

Table 15.0-2

ESBWR Abnormal Event Classifications

Abnormal Event	Event Classification	Relevant SRP(s)
Loss of Feedwater Heating	AOO	15.1.1 - 4
Closure of One Turbine Control Valve	AOO	15.2.1 – 5
Generator Load Rejection with Turbine Bypass	AOO	15.2.1 – 5
Generator Load Rejection with a Single Failure in the Turbine Bypass System	AOO*	15.2.1 – 5
Turbine Trip with Turbine Bypass	AOO	15.2.1 – 5
Turbine Trip with a Single Failure in the Turbine Bypass System	AOO*	15.2.1 – 5
Closure of One Main Steamline Isolation Valve	AOO	15.2.1 – 5
Closure of All Main Steamline Isolation Valves	AOO	15.2.1 – 5
Loss of Condenser Vacuum	AOO	15.2.1 – 5
Loss of Shutdown Cooling Function of RWCU/SDC	AOO	15.2.1 – 5
Inadvertent Isolation Condenser Initiation	AOO	15.1.1 – 4
Runout of One Feedwater Pump	AOO	15.1.1 – 4
Opening of One Turbine Control or Bypass Valve	AOO	15.1.1 – 4
Loss of Unit Auxiliary Transformer **	AOO	15.2.6
Loss of Grid Connection **	AOO	15.2.6
Loss of All Feedwater Flow	AOO	15.2.7
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	Infrequent Event	15.1.1 – 4
Feedwater Controller Failure – Maximum Demand	Infrequent Event	15.1.1 – 4
Pressure Regulator Failure - Opening of All Turbine Control and Bypass Valves	Infrequent Event	15.1.1 – 4
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	Infrequent Event	15.1.1 – 4
Generator Load Rejection with Total Turbine Bypass Failure	Infrequent Event	15.2.1-5
Turbine Trip with Total Turbine Bypass Failure	Infrequent Event	15.2.1-5
Control Rod Withdrawal Error During Refueling	Infrequent Event	15.4.1

Table 15.0-2

ESBWR Abnormal Event Classifications

Abnormal Event	Event Classification	Relevant SRP(s)
Control Rod Withdrawal Error During Startup	Infrequent Event	15.4.1
Control Rod Withdrawal Error During Power Operation	Infrequent Event	15.4.2
Fuel Assembly Loading Error, Mislocated Bundle	Infrequent Event	15.4.7
Fuel Assembly Loading Error, Misoriented Bundle	Infrequent Event	15.4.7
Inadvertent SDC Function Operation	Infrequent Event	15.4.9
Inadvertent Opening of a Safety Relief Valve	Infrequent Event	15.6.1
Inadvertent Opening of a Depressurization Valve	Infrequent Event	15.6.1, 15.6.5
Stuck Open Safety Relief Valve	Infrequent Event	15.6.1
Liquid-Containing Tank Failure	Infrequent Event	15.7.3
Fuel Handling Accident	Accident	15.7.4
LOCA Inside Containment	Accident	15.6.5 & 5a
Main Steamline Break Outside Containment	Accident	15.6.4
Control Rod Drop Accident	See Subsection 15.4.6	
Feedwater Line Break Outside Containment	Accident	15.3.5
Failure of Small Line Carrying Primary Coolant Outside Containment	Accident	15.6.2
RWCU/SDC System Line Failure Outside Containment	Accident	15.6.4, 15.6.5
Spent Fuel Cask Drop Accident	Accident	15.7.5
MSIV Closure With Flux Scram (Overpressure Protection)	Special Event	5.2.2
Shutdown Without Control Rods (i.e., SLC system shutdown capability)	Special Event	9.3.5
Shutdown from Outside Main Control Room	Special Event	7.5
Anticipated Transients Without Scram	Special Event	15.8
Station Blackout	Special Event	8.2 (and RG 1.155)
Safe Shutdown Fire	Special Event	9.5.1
Waste Gas System Leak or Failure	Special Event	11.3

* An AOO in combination with an additional SACF or SOE, as discussed in SRP 15.1 and SRP 15.2.

** Both covered by the Loss of Non-Emergency AC Power to Station Auxiliaries event.

Table 15.0-3

Safety Analysis Acceptance Criteria for AOOs

- Pressures in the reactor coolant and main steam systems shall be maintained below 110% of their design values (i.e., not exceed ASME Code Service Level B), and the reactor steam dome pressure shall be maintained less than or equal to the Reactor Coolant System Pressure Safety Limit in the Technical Specifications.
- Fuel-cladding integrity should be maintained by ensuring that the reactor core is designed with appropriate margin during any conditions of normal operation, including the effects of AOOs. The minimum value of the critical power ratio (CPR) reached during the AOO should be such that 99.9 percent of the fuel rods in the core would not be expected to experience boiling transition during core-wide transients. (This criterion corresponds to the <u>greater than 99.9% of the fuel rods in the core would be expected to avoid boiling transition related safety limit</u> in the Technical Specifications.)
- Uniform cladding strain ≤ 1%.
- No fuel centerline melt.
- Reactor water level shall be maintained above the top of the core (i.e., active fuel).
- Containment and suppression pool pressures and temperatures shall be maintained below their design values.
- An AOO should not generate a more serious plant condition unless other faults occur independently.
- There is no loss of function of any fission product barrier (Safety Relief Valve or Depressurization Valve discharge does not apply).

Table 15.0-4

Safety Analysis Acceptance Criteria for AOOs In Combination With An Additional Single Active Component Failure or Single Operator Error

- Reactor water level shall be maintained above the top of the core (i.e., active fuel).
- Pressures in the reactor coolant and main steam systems shall be maintained below 110% of their design values (i.e., not exceed ASME Code Service Level B), and the reactor steam dome pressure shall be maintained less than or equal to the Reactor Coolant System Pressure Safety Limit in the Technical Specifications.
- If containment isolation is involved, containment and suppression pool pressures and temperatures shall be maintained below their design values.
- Except for fuel cladding, there shall be no loss of function of any fission product barrier.
- Fuel cladding failures shall be limited such that the radiological consequence shall be ≤ 0.05 rem TEDE.

Table 15.0-5

Safety Analysis Acceptance Criteria for Infrequent Events

- Reactor water level shall be maintained above the top of the core (i.e., active fuel).
- Pressures in the reactor coolant and main steam systems shall be maintained below the ASME Service Level C limit, which corresponds to 120% of design pressure.
- Radiological consequence shall be ≤ 2.5 rem TEDE. However, if the applicable SRP section specifies an accident-specific (i.e., different or additional) radiological acceptance criterion, then the accident-specific SRP acceptance criterion/criteria is/are applied. *
- Containment and suppression pool pressures and temperatures shall be maintained below their design values.
- Control room personnel shall not receive a radiation exposure in excess of 5 rem TEDE for the duration of the event.

* For example, the liquid radwaste tank failure must meet 10 CFR 20, Table 2, Column 2 for the liquid release.

Table 15.0-6

Safety Analysis Acceptance Criteria for Accidents

- Pressures in the reactor coolant and main steam systems shall be maintained below the ASME Service Level C limit, which corresponds to 120% of design pressure.
- Radiological consequence shall be ≤ 2.5 rem TEDE, 6.3 rem TEDE, or 25 rem TEDE, depending on the accident-specific acceptance criterion in NUREG-0800, SRP 15.0.1.
- The calculated maximum fuel element cladding temperature shall not exceed 1204ºC (2200ºF).
- The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.
- Containment and suppression pool pressures and temperatures shall be maintained below their design values.
- Control room personnel shall not receive a radiation exposure in excess of 5 rem TEDE for the duration of the accident.

Table 15.0-7

ESBWR Event Classifications and Radiological Acceptance Criteria

Event*	Accident Class**		Radiological Acceptance Criteria***					
	Infrequent Event	Accident	10 CFR 20, App. B, Table 2, Column 2	10 CFR 20.1301	GDC 19, 5 rem TEDE	2.5 rem TEDE	6.3 rem TEDE	25 rem TEDE
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	X				+	X		
Inadvertent SDC Function Operation	X				+	X		
Control Rod Withdrawal Error During Refueling	X				+	X		
Control Rod Withdrawal Error During Startup	X				+	X		
Control Rod Withdrawal Error During Power Operation	X				+	X		
Inadvertent Opening of a Depressurization Valve	X				+	X		
Inadvertent Opening of a Safety Relief Valve	X				+	X		
Stuck Open Safety Relief Valve	X				+	X		
Feedwater Controller Failure – Maximum Demand	X				+	X		
Pressure Regulator Failure - Opening of All Turbine Control and Bypass Valves	X				+	X		
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	X				+	X		
Generator Load Rejection with Total Turbine Bypass Failure	X				+	X		
Turbine Trip with Total Turbine Bypass Failure	X				+	X		
Liquid-Containing Tank Failure	X		X		X	X		

Table 15.0-7

ESBWR Event Classifications and Radiological Acceptance Criteria

Event*	Accident Class**		Radiological Acceptance Criteria***					
	Infrequent Event	Accident	10 CFR 20, App. B, Table 2, Column 2	10 CFR 20.1301	GDC 19, 5 rem TEDE	2.5 rem TEDE	6.3 rem TEDE	25 rem TEDE
Fuel Assembly Loading Errors (mislocated and misoriented)	X				+	X		
Spent Fuel Cask Drop Accident		X			+		X	
Failure of Small Line Carrying Primary Coolant Outside Containment		X			+	X		
Feedwater Line Break Outside Containment		X			+	X		
Reactor Water Cleanup / Shutdown Cooling System Failure Outside Containment		X			+	X		
Control Rod Drop Accident (radiological analysis)	See Subsection 15.4.6							
Main Steamline Break Outside Containment		X			X	X+		X+
LOCA Inside Containment Radiological Analysis, (including all leakage paths)		X			X			X
Fuel Handling Accident		X			++		X	
Waste Gas System Leak or Failure +++					++	X		

* Based on SRP 15 and ABWR FSER (Reference 15.0-3) events involving a radiological consequence.

** From Table 15.0-2.

*** Based on the 10 CFR regulations, SRP 15 and BTP 11-5.

\+ 2.5 rem assuming equilibrium iodine activity in reactor coolant, and 25 rem assuming a pre-incident iodine spike in the reactor coolant.

++ Bounded by the LOCA Inside Containment and Main Steamline Break Outside Containment radiological analyses.

+++ Classified as a special event.

Table 15.0-8

ESBWR Safety Analysis Codes

Safety Analysis	Analysis Code
Stability Evaluation (Chapter 4)	TRACG04[1]
Reactor Building Compartment Pressurization Analysis (Chapter 6)	CONTAIN 2.0
Loss of Feedwater Heating	TRACG04[1]
Closure of One Turbine Control Valve	TRACG04[1]
Generator Load Rejection with Turbine Bypass	TRACG04[1]
Generator Load Rejection with a Single Failure in the Turbine Bypass System	TRACG04[1]
Turbine Trip with Turbine Bypass	TRACG04[1]
Turbine Trip with a Single Failure in the Turbine Bypass System	TRACG04[1]
Closure of One Main Steamline Isolation Valve	TRACG04[1]
Closure of All Main Steamline Isolation Valves	TRACG04[1]
Loss of Condenser Vacuum	TRACG04[1]
Loss of Shutdown Cooling Function of RWCU/SDC	N/A
Inadvertent Isolation Condenser Initiation	TRACG04[1]
Runout of One Feedwater Pump	TRACG04[1]
Opening of One Turbine Control or Bypass Valve	TRACG04[1]
Loss of Non-Emergency AC Power to Station Auxiliaries	TRACG04[1]
Loss of All Feedwater Flow	TRACG04[1]
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	TRACG04[1] / RADTRAD 3.03
Feedwater Controller Failure – Maximum Demand	TRACG04[1]
Pressure Regulator Failure - Opening of All Turbine Control and Bypass Valves	TRACG04[1]
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	TRACG04[1]
Generator Load Rejection with Total Turbine Bypass Failure	TRACG04[1]
Turbine Trip with Total Turbine Bypass Failure	TRACG04[1]
Control Rod Withdrawal Error During Refueling	N/A
Control Rod Withdrawal Error During Startup	PANAC11
Control Rod Withdrawal Error During Power Operation	N/A
Fuel Assembly Loading Error, Mislocated Bundle	N/A
Fuel Assembly Loading Error, Misoriented Bundle	N/A

Table 15.0-8

ESBWR Safety Analysis Codes

Safety Analysis	Analysis Code
Inadvertent SDC Function Operation	N/A
Inadvertent Opening of a Safety/Relief Valve	TRACG04[1]
Inadvertent Opening of a Depressurization Valve	N/A
Stuck Open Safety/Relief Valve	TRACG04[1]
Liquid-Containing Tank Failure	RADTRAD 3.03
Fuel Handling Accident	RADTRAD 3.03
LOCA Inside Containment – Containment Pressurization	TRACG04[2]
LOCA Inside Containment – ECCS Performance	TRACG04[2]
LOCA Inside Containment – Radiological	RADTRAD 3.03[3]
Main Steamline Break Outside Containment	RADTRAD 3.03
Control Rod Drop Accident	N/A
Feedwater Line Break Outside Containment	CONAC04A
Failure of Small Line Carrying Primary Coolant Outside Containment	RADTRAD 3.03
RWCU/SDC System Line Failure Outside Containment	RADTRAD 3.03
Spent Fuel Cask Drop Accident	N/A
MSIV Closure With Flux Scram (Overpressure Protection)	TRACG04[1]
Shutdown Without Control Rods (i.e., SLC system shutdown capability)	PANAC11
Shutdown from Outside Main Control Room	N/A
Anticipated Transients Without Scram (except ATWS Stability) Stability During ATWS	TRACG04[2] TRACG04[1]
Station Blackout	TRACG04[1]
Safe Shutdown Fire	N/A
Waste Gas System Leak or Failure	N/A

1. TRACG04A (Alpha VMS version) is used with core inputs from PANAC11 and fuel gap thermal conductivity input from GSTRM07A.
2. TRACG04P (PC version) is used with fuel gap thermal conductivity input from GSTRM07A.
3. RADTRAD 3.03 is used with inputs from MELCOR 1.8.6 YK.

15.1 NUCLEAR SAFETY OPERATIONAL ANALYSIS

The Nuclear Safety Operational Analysis (NSOA) is a system level qualitative failure modes and effects analysis (FMEA) of plant protective functions that shows which systems and functions are required for the events addressed in Chapter 15 to meet their associated acceptance criteria.

15.1.1 Analytical Approach

15.1.1.1 NSOA Objective

The objective of the NSOA is to identify, for each event in the Chapter 15 safety analyses, the system level requirements that ensure the plant can be brought to a stable safe condition. Specifically, the NSOA considers the entire duration of each event from the spectrum of possible initial conditions and aftermath until either some mode of planned operation is resumed or the plant is in a stable shutdown condition.

The NSOA process uses operational criteria and required actions to identify the required systems, automatic instrument trips, monitored parameters (associated with required operator actions), and auxiliary systems to bring the plant to a stable shutdown condition for each event. The system-level requirements identified as required in the NSOA reflect the licensing basis of the plant and constitute the minimum required actions to bring the plant to a stable shutdown condition. In actual plant operation, additional procedural guidance and plant equipment are available to prevent or further mitigate these events. Finally, the NSOA focuses primarily on active plant features used to bring the plant to a stable shutdown condition; passive plant features are implicitly considered but not explicitly documented in the event evaluations and diagrams.

15.1.1.2 NSOA Relationship to Safety Analysis

The safety analysis is performed to demonstrate compliance with appropriate event acceptance criteria (Subsection 15.0.3) for limiting event paths. Review of the event acceptance criteria illustrates the safety analysis focus on event consequences. The event acceptance criteria are either fission product barrier design basis limits or radiological dose limits derived from applicable regulatory requirements.

As such, the event paths analyzed as "limiting" in the safety analysis generally correspond to one of the event paths for each event in the NSOA, or a conservative representation of one.

This safety analysis limiting-event path is selected to pose the most significant challenge to the applicable event acceptance criteria, and thus, typically concentrates on the short-term response to the event. Therefore, the safety analysis is consequences oriented, focusing on the limiting short-term response to the event, and the NSOA is event/system oriented, focusing on the system-level required actions necessary over the entire duration of the event (long-term response) to bring the plant to a stable configuration.

15.1.2 Method of Analysis

15.1.2.1 Operational Criteria

The operational criteria are identified in Table 15.1-1.

The operational criteria establish the requirements for:

- Satisfying the applicable required actions to bring the plant to a stable condition consistent with the plant licensing basis;
- Applying the single failure criterion; and
- Satisfying requirements unique to certain events.

Operational criteria are based upon the applicable regulatory requirements and guidance, industry codes and standards, plant-specific licensing requirements, nuclear steam supply system (NSSS) requirements, and fuel supplier design requirements.

15.1.2.2 Analysis Assumptions and Initial Conditions

15.1.2.2.1 Operating Modes

The ESBWR operating modes, encompassing the entire operating envelope in which the plant can exist, are defined in Table 15.1-2. These operating mode definitions are consistent with the MODES defined in the Technical Specifications.

The main objective in defining operating modes is to divide the plant operating spectrum into sets of initial conditions. The ESBWR operating modes associated with planned operations define the operating envelope from which anticipated operational occurrences (AOOs), Infrequent Events, Accidents, and Special Events are initiated. ESBWR operating modes define the physical condition (e.g., pressure, temperature) of the reactor. The events associated with each operating mode are provided in Table 15.1-3.

Each operating mode includes an allowable range of values for important plant parameters. Within each mode, these parameters are considered over their entire range.

For each event, the operating mode(s) in which the event can occur are determined. An event is considered applicable within an operating mode if the event can be initiated from the operating envelope that characterizes the operating mode.

15.1.2.2.2 Planned Operation

Planned operation refers to normal plant operation under planned conditions within the allowable operating envelope in the absence of significant abnormalities. Following an event, planned operation is not considered to have resumed until the plant operating mode is identical to a planned operating mode that could have been attained had the event not occurred. As defined, planned operation can be considered as a chronological sequence:

Plant outage > achieving criticality > heatup > power operation > achieving

shutdown > cooldown > plant outage

15.1.2.3 Event Analysis Rules

The event analysis rules are consistent with applicable regulatory requirements and guidance, and applicable industry codes and standards. Table 15.1-4 provides the event analysis rules used in performing the NSOA, along with explanations of the individual rules.

15.1.3 NSOA Results

15.1.3.1 Event Evaluations and Diagrams

The individual event evaluations in conjunction with their respective event diagrams document the detailed results of the NSOA. The event diagram format is shown in Figure 15.1-1. The locations of the event evaluations are identified in Subsection 15.1.4 and the associated event diagrams are shown in Figures 15.1-2 through 15.1-47.

An event diagram for each event evaluated identifies the applicable operating mode(s) (for the overall event evaluation and, where applicable, for event paths that only apply to specific operating modes), the required actions, the relationship of system operation and operator actions to the required actions, and the required functional redundancy. In addition, event diagrams identify each signal that initiates automatic system operation or alerts the operator to the need for action.

15.1.3.2 Summary Matrices

A system, instrument trip, or operator action is considered "required" if identified on an event diagram as necessary to satisfy a required action or the operational criteria.

Based upon the event evaluations and diagrams, matrices are provided in Table 15.1-5 and Table 15.1-6 to identify the required systems and automatic instrument trips, respectively for the events evaluated in the NSOA and the safety analyses.

15.1.4 Event Evaluations

For each of the events considered in the NSOA, Table 15.1-7 shows the locations of the event description and the relevant protection sequence diagram.

15.1.5 COL Information

None

15.1.6 References

None

Table 15.1-1

Operational Criteria

Applicability	Criteria
1. Planned operation	The plant is operated observing operating mode monitoring requirements identified to preserve safety analysis assumptions and establish initial conditions for event analyses. Normal plant operating procedures are followed as applicable.
2. All events	All required actions to bring the plant to a stable condition consistent with the plant licensing basis are satisfied.
3. All events	Emergency Operating Procedures (EOPs) are followed when applicable.
4. AOOs	The plant is designed and operated such that no single failure in mitigation systems can prevent required actions from being satisfied.
5. Infrequent Events and Accidents	The plant is designed and operated to satisfy required actions, considering limiting single failure as defined by applicable regulatory requirements and licensing commitments.
6. AOOs, Infrequent Events and Accidents	Single-failure criterion is not applicable during periods of system or component testing required by Technical Specifications (TS) or when operating under limiting conditions for operation required by Technical Specifications.
7. Special events	The plant is designed and operated consistent with applicable regulatory requirements and licensing commitments.

Table 15.1-2

ESBWR Operating Modes

MODE 6 – REFUELING

- Allowable Mode Switch Positions: SHUTDOWN or REFUEL
- Power Considerations: Decay Heat Only
- One or more reactor vessel head closure bolts less than fully tensioned

MODE 6 S – RPV VENTED AND REACTOR NOT SHUTDOWN (Applies when performing Shutdown Margin test in accordance with Technical Specifications)

- Allowable Mode Switch Positions: SHUTDOWN REFUEL STARTUP
- Power Considerations: Decay Heat Only
- One or more reactor vessel head closure bolts less than fully tensioned

MODE 5 – COLD SHUTDOWN

- Allowable Mode Switch Positions: SHUTDOWN
- 93.3ºC (200ºF)≥ Average Reactor Coolant Temperature
- Power Considerations: Decay Heat Only

MODE 4 – STABLE SHUTDOWN

- Allowable Mode Switch Positions: SHUTDOWN
- 215.6ºC (420ºF)≥ Average Reactor Coolant Temperature > 93.3ºC (200ºF)
- Power Considerations: Decay Heat Only

MODE 3 – HOT SHUTDOWN

- Allowable Mode Switch Positions: SHUTDOWN
- Average Reactor Coolant Temperature > 215.6ºC (420ºF)
- Power Considerations: Decay Heat Only

MODE 2 – STARTUP

- Allowable Mode Switch Positions: REFUEL or STARTUP
- Power Considerations: APRM Fixed Neutron Flux – High, Setdown Power ≥ Reactor Power ≥ Decay Heat

MODE 1 – POWER OPERATION

- Allowable Mode Switch Positions: RUN
- Power Considerations: Licensed Power Level ≥ Reactor Power ≥ Decay Heat

Table 15.1-3

ESBWR Events Associated With Operating Modes

Abnormal Event	Applicable Operating Mode(s)
Loss of Feedwater Heating	1 & 2
Closure of One Turbine Control Valve	1 & 2
Generator Load Rejection with Turbine Bypass	1 & 2
Generator Load Rejection with a Single Failure in the Turbine Bypass System	1 & 2
Turbine Trip with Turbine Bypass	1 & 2
Turbine Trip with a Single Failure in the Turbine Bypass System	1 & 2
Closure of One Main Steamline Isolation Valve	1 - 4
Closure of All Main Steamline Isolation Valves	1 - 4
Loss of Condenser Vacuum	1 - 4
Loss of Shutdown Cooling Function of RWCU/SDC System	2 - 6 & 6S
Inadvertent Isolation Condenser Initiation	1 - 6 & 6S
Runout of One Feedwater Pump	1 & 2
Opening of One Turbine Control or Bypass Valve	1 - 4
Loss of Non-Emergency AC Power to Station Auxiliaries	1 - 6 & 6S
Loss of All Feedwater Flow	1 & 2
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	1 & 2
Feedwater Controller Failure – Maximum Demand	1 & 2
Pressure Regulator Failure Opening of All Turbine Control and Bypass Valves	1 - 4
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	1 & 2
Generator Load Rejection with Total Turbine Bypass Failure	1 & 2
Turbine Trip with Total Turbine Bypass Failure	1 & 2
Control Rod Withdrawal Error During Refueling	6
Control Rod Withdrawal Error During Startup	2 - 5 & 6S

Table 15.1-3

ESBWR Events Associated With Operating Modes

Abnormal Event	Applicable Operating Mode(s)
Control Rod Withdrawal Error During Power Operation	1 & 2
Fuel Assembly Loading Error, Mislocated Bundle	1 - 6 & 6S
Fuel Assembly Loading Error, Misoriented Bundle	1 - 6 & 6S
Inadvertent SDC Function Operation	1 - 5
Inadvertent Opening of a Safety Relief Valve	1 - 6 & 6S
Inadvertent Opening of a Depressurization Valve	1 - 6 & 6S
Stuck Open Safety Relief Valve	1 - 6 & 6S
Liquid-Containing Tank Failure	1 - 6 & 6S
Fuel Handling Accident	1 - 6 & 6S
LOCA Inside Containment	1 - 6 & 6S
Main Steamline Break Outside Containment	1 - 5
Control Rod Drop Accident	1 - 6 & 6S
Feedwater Line Break Outside Containment	1 - 6 & 6S
Failure of Small Line Carrying Primary Coolant Outside Containment	1 - 6 & 6S
RWCU/SDC System Line Failure Outside Containment	1 - 6 & 6S
Spent Fuel Cask Drop Accident	1 - 6 & 6S
MSIV Closure With Flux Scram (Overpressure Protection)	1 & 2
Shutdown Without Control Rods (i.e., SLCS shutdown capability)	1 & 2 & 6S
Shutdown from Outside Main Control Room	1 & 2 & 6S
Anticipated Transients Without Scram	1 & 2
Station Blackout	1 & 2
Safe Shutdown Fire	1 & 2
Waste Gas System Leak or Failure	1 & 2

Table 15.1-4

Event Analysis Rules

A. General Rules	**Explanation**
A.1 Include all events that are part of the plant safety analysis.	All events considered in the plant safety analysis are included in the NSOA, consistent with NSOA goals and objectives.
A.2 Identify on event diagrams all required systems, automatic trips, and operator actions necessary to either satisfy operational criteria or perform required actions.	Systems, automatic trips, and operator actions are identified only if they are uniquely necessary to either accomplish required actions or satisfy operational criteria.
A.3 Consider all plant systems, including passive plant features required in the mitigation of events.	The functions of passive plant features (e.g., MSL flow restrictors and CRD housing supports) used to mitigate the consequences of events are identified.
A.4 Consider hardware restrictions included in the plant design to prevent operation outside the operating envelope.	Hardware restrictions (e.g., control rod withdrawal restrictions and refueling interlocks) are included in the plant design to constrain plant operation to within the allowable operating envelope.
B. Planned Operation Rules	**Explanation**
B.1 Consider only systems, limits, and restrictions necessary to attain planned operation and satisfy operational criteria.	Consideration of planned operation is limited and not followed through to completion, because planned operation is constrained by normal plant operating procedures.
B.2 Limit the initial conditions for AOOs, infrequent events, accidents, and special events to operating modes and envelopes allowed during planned operation in the applicable operating mode.	All events in the safety analysis are initiated from an operating mode within the allowable operating envelope.
B.3 Consider the full range of initial conditions for each event analyzed.	This rule assures that all event paths are identified. Different initial conditions can lead to different paths that may establish unique requirements.
B.4 Apply hardware restrictions only to planned operation.	Restrictions are hardware-implemented constraints on normal plant operation to limit the consequences of postulated events.

Table 15.1-4

Event Analysis Rules

B. Planned Operation Rules	Explanation
B.5 Identify normal operating systems considered for a planned operation function during an event as "Planned Operation - Specific System Available."	Normal operating systems are considered if the system is employed in the same manner during the event as it was prior to the event or if continued operation can significantly change the event path.
C. Event Diagram Rules	**Explanation**
C.1 Consider the entire duration of the event from the spectrum of possible initial conditions and aftermath until either some mode of planned operation is resumed or the plant is in a stable condition with continuity of core cooling.	Planned operation is considered "resumed" when normal operating procedures are being followed and plant operation is identical to that used in any operating mode consistent with allowable operating modes and envelopes. A stable operating condition is defined as the completion of all required actions and the stabilization of plant parameters.
C.2 Identify systems, automatic trips, and operator actions if there is a unique requirement as a result of the event. If a normal operating system that was operating prior to the event will be employed in the same manner during the event and if the event did not affect system operation, the system does not appear as a unique requirement on the event diagram.	Systems, limits, and operator actions are identified as "required" only if a unique requirement to satisfy either required actions or operational criteria is established. When normal operating systems are considered, specific systems assumed to be available are identified.
C.3 Credit operator action only if the operator can reasonably be expected to accomplish the required action under existing conditions and has availability of necessary information to implement required plant procedures.	Operator action may be necessary to either attain planned operation or a stable condition.
C.4 Identify two types of parameters: Parameters that initiate an automatic trip or system actuation and monitored parameters (available to the operator) that require action.	Parameters are instrument setpoints at which either an automatic trip or system initiation or operator action is assumed to occur. Where either an automatic action or operator action accomplishes the same function, the automatic action is identified.

Table 15.1-4

Event Analysis Rules

C. Event Diagram Rules	Explanation
C.5 Consider a system that plays a unique role in response to an AOO, infrequent event, accident, or special event to be "required" unless the system's effects are not included in the event analysis.	Systems that have a unique role in an event are considered "required" unless the safety analysis for the event provides a basis that operation of the system is not required.
C.6 Identify operating mode(s) in which the event is applicable.	Because of plant operational considerations and the definition of operating modes, not all events can occur in all operating modes.
C.7 Identify the safety-related paths that include: • Required actions. • Front-line systems. • Automatic trips. • Monitored parameters. • Normal operating systems evaluated in analysis.	Event diagrams are the primary source of documentation of NSOA results. Notes identify required actions that are not applicable and required actions satisfied by the normal operating systems.
C.8 Identify passive plant features necessary at the system level.	Passive plant features are associated with system level requirements. (To avoid adding unnecessary complexity to the event diagrams, the event diagrams do not show applicable passive plant features.)

Table 15.1-5

NSOA System Event Matrix

	SRV – Safety Relief Mode	SRV – Power Actuated Mode (ADS)	DPV Actuation	ICS – MSIV Position	ICS – RPV High Dome Pressure (10-second delay)	ICS – RPV Low Water Level (L2 30-sec delay)	ICS – RPV Low Water Level (L1)	TBV Closure – Low-low condenser vacuum	ICS – Los of Power Generation Bus (Loss of Feedwater Flow)	TBV Initiation – TSV Closure	TBV Initiation – TCV Fast Closure	TSV Closure – RPV High Water Level (L8)	TSV Closure – Low Condenser Vacuum	TCV Fast Closure – Load Rejection	MSIV Closure – RPV Low Water Level (L2 w/30 sec)	MSIV Closure – RPV Low Water Level (L1)	MSIV Closure – Low Turbine Inlet/Main Streamline Pressure	MSIV Closure – Low Main Condenser Vacuum	MSIV Closure - High Steamline Flow	FW Pump Runback – L8	CRD Makeup Water – RPV Low Water Level (L2)	RWCU/SDC Operation	ATWS – Feedwater Flow Runback	ATWS – ADS Inhibition	SLCS – RPV Dome High Pressure - APRM not downscale	SLCS – DPV Open	SLCS – RPV Low Water Level (L2) - APRM not downscale	FAPCS – High Suppression Pool Temperature	SCRRI/SRI	GDCS	GDCS Equilizing Lines	High Radiation MCR Recirculation	Passive Containment Cooling PCCS
Loss of Feedwater Heating																													X				
Closure of One Turbine Control Valve																																	
Generator Load Rejection with Bypass											X																		X				
Generator Load Rejection with a Single Failure in the Turbine Bypass System				X	X	X					X				X		X																
Turbine Trip with Bypass										X																			X				
Turbine Trip with a Single Failure in the Turbine Bypass System				X	X	X				X					X		X																
Closure of One Main Steamline Isolation Valve				X															X														
Closure of All Main Steamline Isolation Valves				X																													

Table 15.1-5

NSOA System Event Matrix

	SRV – Safety Relief Mode	SRV – Power Actuated Mode (ADS)	DPV Actuation	ICS – MSIV Position	ICS – RPV High Dome Pressure (10-second delay)	ICS – RPV Low Water Level (L2 30-sec delay)	ICS – RPV Low Water Level (L1)	TBV Closure – Low-low condenser vacuum	ICS – Los of Power Generation Bus (Loss of Feedwater Flow)	TBV Initiation – TSV Closure	TBV Initiation – TCV Fast Closure	TSV Closure – RPV High Water Level (L8)	TSV Closure – Low Condenser Vacuum	TCV Fast Closure – Load Rejection	MSIV Closure – RPV Low Water Level (L2 w/30 sec)	MSIV Closure – RPV Low Water Level (L1)	MSIV Closure – Low Turbine Inlet/Main Streamline Pressure	MSIV Closure – Low Main Condenser Vacuum	MSIV Closure - High Steamline Flow	FW Pump Runback – L8	CRD Makeup Water – RPV Low Water Level (L2)	RWCU/SDC Operation	ATWS – Feedwater Flow Runback	ATWS – ADS Inhibition	SLCS – RPV Dome High Pressure - APRM not downscale	SLCS – DPV Open	SLCS – RPV Low Water Level (L2) - APRM not downscale	FAPCS – High Suppression Pool Temperature	SCRRI/SRI	GDCS	GDCS Equilizing Lines	High Radiation MCR Recirculation	Passive Containment Cooling PCCS
Loss of Condenser Vacuum				X				X					X					X															
Loss of Shutdown Cooling Function of RWCU/SDC System					X	X																								X			
Inadvertent Isolation Condenser Initiation																																	
Runout of One Feedwater Pump																																	
Opening of One Turbine Control or Bypass Valve																																	
Loss of Non-Emergency AC Power to Station Auxiliaries								X	X		X			X	X			X															
Loss of All Feedwater Flow									X						X																		
Loss of Feedwater Heating With Failure of SCRRI																																	

Table 15.1-5

NSOA System Event Matrix

	SRV – Safety Relief Mode	SRV – Power Actuated Mode (ADS)	DPV Actuation	ICS – MSIV Position	ICS – RPV High Dome Pressure (10-second delay)	ICS – RPV Low Water Level (L2 30-sec delay)	ICS – RPV Low Water Level (L1)	TBV Closure – Low-low condenser vacuum	ICS – Los of Power Generation Bus (Loss of Feedwater Flow)	TBV Initiation – TSV Closure	TBV Initiation – TCV Fast Closure	TSV Closure – RPV High Water Level (L8)	TSV Closure – Low Condenser Vacuum	TCV Fast Closure – Load Rejection	MSIV Closure – RPV Low Water Level (L2 w/30 sec)	MSIV Closure – RPV Low Water Level (L1)	MSIV Closure – Low Turbine Inlet/Main Streamline Pressure	MSIV Closure – Low Main Condenser Vacuum	MSIV Closure - High Steamline Flow	FW Pump Runback – L8	CRD Makeup Water – RPV Low Water Level (L2)	RWCU/SDC Operation	ATWS – Feedwater Flow Runback	ATWS – ADS Inhibition	SLCS – RPV Dome High Pressure - APRM not downscale	SLCS – DPV Open	SLCS – RPV Low Water Level (L2) - APRM not downscale	FAPCS – High Suppression Pool Temperature	SCRRI/SRI	GDCS	GDCS Equilizing Lines	High Radiation MCR Recirculation	Passive Containment Cooling PCCS
Feedwater Controller Failure – Maximum Demand				X	X	X				X		X			X		X																
Pressure Regulator Failure Opening of All Turbine Control and Bypass Valves				X													X																
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves				X	X	X									X		X																
Generator Load Rejection with Total Turbine Bypass Failure				X	X	X								X	X		X																
Turbine Trip with Total Turbine Bypass Failure				X	X	X									X		X																
Control Rod Withdrawal Error During Refueling																																	
Control Rod Withdrawal Error During Startup																																	

Table 15.1-5

NSOA System Event Matrix

	SRV – Safety Relief Mode	SRV – Power Actuated Mode (ADS)	DPV Actuation	ICS – MSIV Position	ICS – RPV High Dome Pressure (10-second delay)	ICS – RPV Low Water Level (L2 30-sec delay)	ICS – RPV Low Water Level (L1)	TBV Closure – Low-low condenser vacuum	ICS – Los of Power Generation Bus (Loss of Feedwater Flow)	TBV Initiation – TSV Closure	TBV Initiation – TCV Fast Closure	TSV Closure – RPV High Water Level (L8)	TSV Closure – Low Condenser Vacuum	TCV Fast Closure – Load Rejection	MSIV Closure – RPV Low Water Level (L2 w/30 sec)	MSIV Closure – RPV Low Water Level (L1)	MSIV Closure – Low Turbine Inlet/Main Streamline Pressure	MSIV Closure – Low Main Condenser Vacuum	MSIV Closure - High Steamline Flow	FW Pump Runback – L8	CRD Makeup Water – RPV Low Water Level (L2)	RWCU/SDC Operation	ATWS – Feedwater Flow Runback	ATWS – ADS Inhibition	SLCS – RPV Dome High Pressure - APRM not downscale	SLCS – DPV Open	SLCS – RPV Low Water Level (L2) - APRM not downscale	FAPCS – High Suppression Pool Temperature	SCRRI/SRI	GDCS	GDCS Equilizing Lines	High Radiation MCR Recirculation	Passive Containment Cooling PCCS
Control Rod Withdrawal Error During Power Operation																																	
Fuel Assembly Loading Error, Mislocated Bundle																																	
Fuel Assembly Loading Error, Misoriented Bundle																																	
Inadvertent SDC Function Operation																																	
Inadvertent Opening of a Safety Relief Valve																																	
Inadvertent Opening of a DPV		X	X																											X			X
Stuck Open Safety Relief Valve		X	X	X													X													X			X
Liquid-Containing Tank Failure																																	
Fuel Handling Accident																																	
LOCA Inside Containment		X	X	X	X	X	X		X						X	X										X				X	X	X	X

Table 15.1-5

NSOA System Event Matrix

	SRV – Safety Relief Mode	SRV – Power Actuated Mode (ADS)	DPV Actuation	ICS – MSIV Position	ICS – RPV High Dome Pressure (10-second delay)	ICS – RPV Low Water Level (L2 30-sec delay)	ICS – RPV Low Water Level (L1)	TBV Closure – Low-low condenser vacuum	ICS – Los of Power Generation Bus (Loss of Feedwater Flow)	TBV Initiation – TSV Closure	TBV Initiation – TCV Fast Closure	TSV Closure – RPV High Water Level (L8)	TSV Closure – Low Condenser Vacuum	TCV Fast Closure – Load Rejection	MSIV Closure – RPV Low Water Level (L2 w/30 sec)	MSIV Closure – RPV Low Water Level (L1)	MSIV Closure – Low Turbine Inlet/Main Streamline Pressure	MSIV Closure – Low Main Condenser Vacuum	MSIV Closure - High Steamline Flow	FW Pump Runback – L8	CRD Makeup Water – RPV Low Water Level (L2)	RWCU/SDC Operation	ATWS – Feedwater Flow Runback	ATWS – ADS Inhibition	SLCS – RPV Dome High Pressure - APRM not downscale	SLCS – DPV Open	SLCS – RPV Low Water Level (L2) - APRM not downscale	FAPCS – High Suppression Pool Temperature	SCRRI/SRI	GDCS	GDCS Equilizing Lines	High Radiation MCR Recirculation	Passive Containment Cooling PCCS
Main Steamline Break Outside Containment		X	X	X	X	X	X		X						X	X	X		X							X				X	X		
Control Rod Drop Accident																																	
Feedwater Line Break Outside Containment		X	X	X	X	X	X		X						X	X										X				X	X		X
Failure of Small Line Carrying Primary Coolant Outside Containment		X	X	X	X	X	X		X						X	X										X				X	X		X
RWCU/SDC System Line Failure Outside Containment		X	X	X	X	X	X		X						X	X										X				X	X		X
Spent Fuel Cask Drop Accident																																	
MSIV Closure With Flux Scram (Overpressure Protection)	X																																
Shutdown Without Control Rods (i.e., SLCS shutdown capability)																																	

Table 15.1-5
NSOA System Event Matrix

	SRV – Safety Relief Mode	SRV – Power Actuated Mode (ADS)	DPV Actuation	ICS – MSIV Position	ICS – RPV High Dome Pressure (10-second delay)	ICS – RPV Low Water Level (L2 30-sec delay)	ICS – RPV Low Water Level (L1)	TBV Closure – Low-low condenser vacuum	ICS – Los of Power Generation Bus (Loss of Feedwater Flow)	TBV Initiation – TSV Closure	TBV Initiation – TCV Fast Closure	TSV Closure – RPV High Water Level (L8)	TSV Closure – Low Condenser Vacuum	TCV Fast Closure – Load Rejection	MSIV Closure – RPV Low Water Level (L2 w/30 sec)	MSIV Closure – RPV Low Water Level (L1)	MSIV Closure – Low Turbine Inlet/Main Streamline Pressure	MSIV Closure – Low Main Condenser Vacuum	MSIV Closure - High Steamline Flow	FW Pump Runback – L8	CRD Makeup Water – RPV Low Water Level (L2)	RWCU/SDC Operation	ATWS – Feedwater Flow Runback	ATWS – ADS Inhibition	SLCS – RPV Dome High Pressure - APRM not downscale	SLCS – DPV Open	SLCS – RPV Low Water Level (L2) - APRM not downscale	FAPCS – High Suppression Pool Temperature	SCRRI/SRI	GDCS	GDCS Equilizing Lines	High Radiation MCR Recirculation	Passive Containment Cooling PCCS
Shutdown from Outside Main Control Room				X		X	X								X	X																	
Anticipated Transients Without Scram	X			X	X	X	X														X		X	X	X		X	X					
Station Blackout		X	X						X						X	X														X	X		X
Safe Shutdown Fire				X		X									X						X												
Waste Gas System Leak or Failure																																	

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – Loss of Power on Four Power Generation Buses	Scram – MSIV Position	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with insufficient bypass available)	Scram – TCV Fast Closure (with insufficient bypass available)	Scram – Low Condenser Vacuum	Scram – (Loss of Power Generation Bus) – Loss of Feedwater Flow	Scram – SRNM Period	Scram – High Drywell Pressure	Rod Block – SRNM Period or ALTM Parameter Exceeded
Loss of Feedwater Heating														
Closure of One Turbine Control Valve														
Generator Load Rejection with Bypass														
Generator Load Rejection with a Single Failure in the Turbine Bypass System									X					
Turbine Trip with Bypass														
Turbine Trip with a Single Failure in the Turbine Bypass System								X						

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – Loss of Power on Four Power Generation Buses	Scram – MSIV Position	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with insufficient bypass available)	Scram – TCV Fast Closure (with insufficient bypass available)	Scram – Low Condenser Vacuum	Scram – (Loss of Power Generation Bus) – Loss of Feedwater Flow	Scram – SRNM Period	Scram – High Drywell Pressure	Rod Block – SRNM Period or ALTM Parameter Exceeded
Closure of One Main Steamline Isolation Valve						X								
Closure of All Main Steamline Isolation Valves						X								
Loss of Condenser Vacuum										X				
Loss of Shutdown Cooling Function of RWCU/SDC System														
Inadvertent Isolation Condenser Initiation														
Runout of One Feedwater Pump														
Opening of One Turbine Control or Bypass Valve														

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – Loss of Power on Four Power Generation Buses	Scram – MSIV Position	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with insufficient bypass available)	Scram – TCV Fast Closure (with insufficient bypass available)	Scram – Low Condenser Vacuum	Scram – (Loss of Power Generation Bus) – Loss of Feedwater Flow	Scram – SRNM Period	Scram – High Drywell Pressure	Rod Block – SRNM Period or ALTM Parameter Exceeded
Loss of Non-Emergency AC Power to Station Auxiliaries											X			
Loss of All Feedwater Flow											X			
Loss of Feedwater Heating With Failure of SCRRI		No credit												
Feedwater Controller Failure – Maximum Demand				X										
Pressure Regulator Failure Opening of All Turbine Control and Bypass Valves						X								

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – Loss of Power on Four Power Generation Buses	Scram – MSIV Position	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with insufficient bypass available)	Scram – TCV Fast Closure (with insufficient bypass available)	Scram – Low Condenser Vacuum	Scram – (Loss of Power Generation Bus) – Loss of Feedwater Flow	Scram – SRNM Period	Scram – High Drywell Pressure	Rod Block – SRNM Period or ALTM Parameter Exceeded
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	X													
Generator Load Rejection with Total Turbine Bypass Failure									X					
Turbine Trip with Total Turbine Bypass Failure								X						
Control Rod Withdrawal Error During Refueling														
Control Rod Withdrawal Error During Startup												X		X
Control Rod Withdrawal Error During Power Operation														X

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – Loss of Power on Four Power Generation Buses	Scram – MSIV Position	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with insufficient bypass available)	Scram – TCV Fast Closure (with insufficient bypass available)	Scram – Low Condenser Vacuum	Scram – (Loss of Power Generation Bus) – Loss of Feedwater Flow	Scram – SRNM Period	Scram – High Drywell Pressure	Rod Block – SRNM Period or ALTM Parameter Exceeded
Fuel Assembly Loading Error, Mislocated Bundle														
Fuel Assembly Loading Error, Misoriented Bundle														
Inadvertent SDC Function Operation	X													
Inadvertent Opening of a Safety Relief Valve							X							
Inadvertent Opening of a DPV													X	
Stuck Open Safety Relief Valve														
Liquid-Containing Tank Failure														

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – Loss of Power on Four Power Generation Buses	Scram – MSIV Position	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with insufficient bypass available)	Scram – TCV Fast Closure (with insufficient bypass available)	Scram – Low Condenser Vacuum	Scram – (Loss of Power Generation Bus) – Loss of Feedwater Flow	Scram – SRNM Period	Scram – High Drywell Pressure	Rod Block – SRNM Period or ALTM Parameter Exceeded
Fuel Handling Accident														
LOCA Inside Containment			X								X		X	
Main Steamline Break Outside Containment			X			X					X			
Control Rod Drop Accident														
Feedwater Line Break Outside Containment			X			X					X			
Failure of Small Line Outside Containment			X			X					X			
RWCU/SDC System Line Failure Outside Containment			X			X					X			
Spent Fuel Cask Drop Accident														

Table 15.1-6

NSOA Automatic Instrument Trip/Event Matrix

	Scram – APRM High Neutron Flux	Scram – APRM High Simulated Thermal Power	Scram – RPV Low Water Level (L3)	Scram – RPV High Water Level (L8)	Scram – Loss of Power on Four Power Generation Buses	Scram – MSIV Position	Scram – High Suppression Pool Temperature	Scram – TSV Closure (with insufficient bypass available)	Scram – TCV Fast Closure (with insufficient bypass available)	Scram – Low Condenser Vacuum	Scram – (Loss of Power Generation Bus) – Loss of Feedwater Flow	Scram – SRNM Period	Scram – High Drywell Pressure	Rod Block – SRNM Period or ALTM Parameter Exceeded
MSIV Closure Flux Scram (Overpressure Protection)	X													
Shutdown W/O Control Rods (SLCS capability)														
Shutdown from Outside Main Control Room														
Anticipated Transients Without Scram (ATWS)														
Station Blackout											X			
Safe Shutdown Fire														
Waste Gas System Leak or Failure														

Table 15.1-7

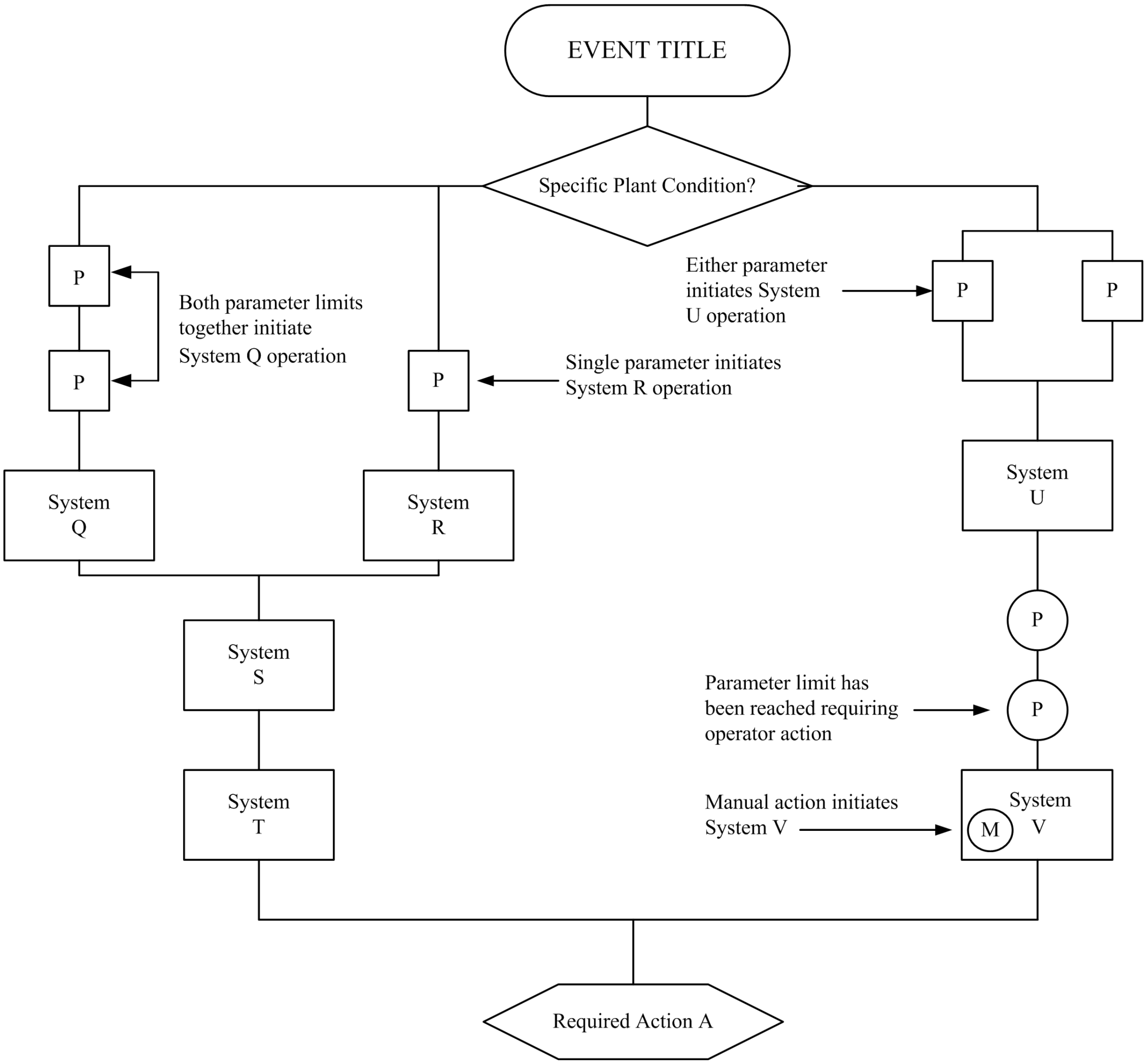
ESBWR NSOA Events

NSOA Event	Subsection Describing Event	Relevant Event Diagram
Loss of Feedwater Heating	15.2.1.1	15.1-2
Closure of One Turbine Control Valve	15.2.2.1	15.1-3
Generator Load Rejection with Turbine Bypass	15.2.2.2	15.1-4
Generator Load Rejection with a Single Failure in the Turbine Bypass System	15.2.2.3	15.1-5
Turbine Trip with Turbine Bypass	15.2.2.4	15.1-6
Turbine Trip with a Single Failure in the Turbine Bypass System	15.2.2.5	15.1-7
Closure of One Main Steamline Isolation Valve	15.2.2.6	15.1-8
Closure of All Main Steamline Isolation Valves	15.2.2.7	15.1-9
Loss of Condenser Vacuum	15.2.2.8	15.1-10
Loss of Shutdown Cooling Function of RWCU/SDC System	15.2.2.9	15.1-11
Inadvertent Isolation Condenser Initiation	15.2.4.1	15.1-12
Runout of One Feedwater Pump	15.2.4.2	15.1-13
Opening of One Turbine Control or Bypass Valve	15.2.5.1	15.1-14
Loss of Non-Emergency AC Power to Station Auxiliaries	15.2.5.2	15.1-15
Loss of All Feedwater Flow	15.2.5.3	15.1-16
Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	15.3.1	15.1-17
Feedwater Controller Failure – Maximum Demand	15.3.2	15.1-18
Pressure Regulator Failure Opening of All Turbine Control and Bypass Valves	15.3.3	15.1-19
Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	15.3.4	15.1-20
Generator Load Rejection with Total Turbine Bypass Failure	15.3.5	15.1-21
Turbine Trip with Total Turbine Bypass Failure	15.3.6	15.1-22
Control Rod Withdrawal Error During Refueling	15.3.7	15.1-23
Control Rod Withdrawal Error During Startup	15.3.8	15.1-24

Table 15.1-7

ESBWR NSOA Events

NSOA Event	Subsection Describing Event	Relevant Event Diagram
Control Rod Withdrawal Error During Power Operation	15.3.9	15.1-25
Fuel Assembly Loading Error, Mislocated Bundle	15.3.10	15.1-26
Fuel Assembly Loading Error, Misoriented Bundle	15.3.11	15.1-27
Inadvertent SDC Function Operation	15.3.12	15.1-28
Inadvertent Opening of a Safety Relief Valve	15.3.13	15.1-29
Inadvertent Opening of a Depressurization Valve	15.3.14	15.1-30
Stuck Open Safety Relief Valve	15.3.15	15.1-31
Liquid-Containing Tank Failure	15.3.16	15.1-32
Fuel Handling Accident	15.4.1	15.1-33
LOCA Inside Containment	15.4.2, 15.4.3, 15.4.4	15.1-34
Main Steamline Break Outside Containment	15.4.5	15.1-35
Control Rod Drop Accident	15.4.6	15.1-36
Feedwater Line Break Outside Containment	15.4.7	15.1-37
Failure of Small Line Carrying Primary Coolant Outside Containment	15.4.8	15.1-38
RWCU/SDC System Line Failure Outside Containment	15.4.9	15.1-39
Spent Fuel Cask Drop Accident	15.4.10	15.1-40
MSIV Closure with Flux Scram (Overpressure Protection)	15.5.1	15.1-41
Shutdown Without Control Rods (i.e., SLCS shutdown capability)	15.5.2	15.1-42
Shutdown from Outside Main Control Room	15.5.3	15.1-43
Anticipated Transients Without Scram	15.5.4	15.1-44
Station Blackout	15.5.5	15.1-45
Safe Shutdown Fire	15.5.6	15.1-46
Waste Gas System Leak or Failure	15.5.7	15.1-47



NOTE:
Either System Q or System R operation is required
Operation of both System S and System T is required

Figure 15.1-1. Event Diagram Format

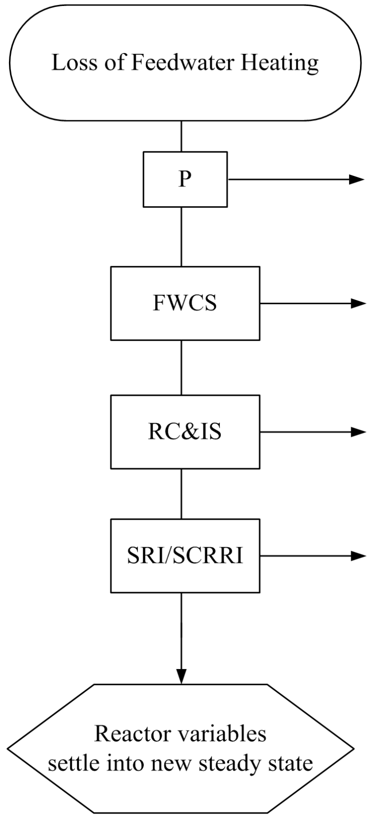


Figure 15.1-2. Event Diagram – Loss of Feedwater Heating

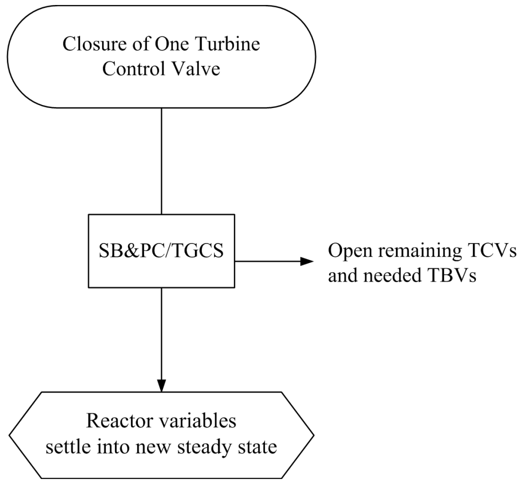


Figure 15.1-3. Event Diagram – Closure of One Turbine Control Valve

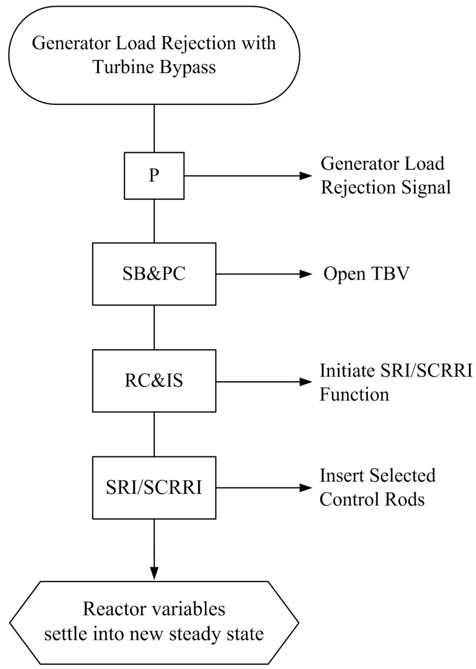


Figure 15.1-4. Event Diagram – Generator Load Rejection with Turbine Bypass

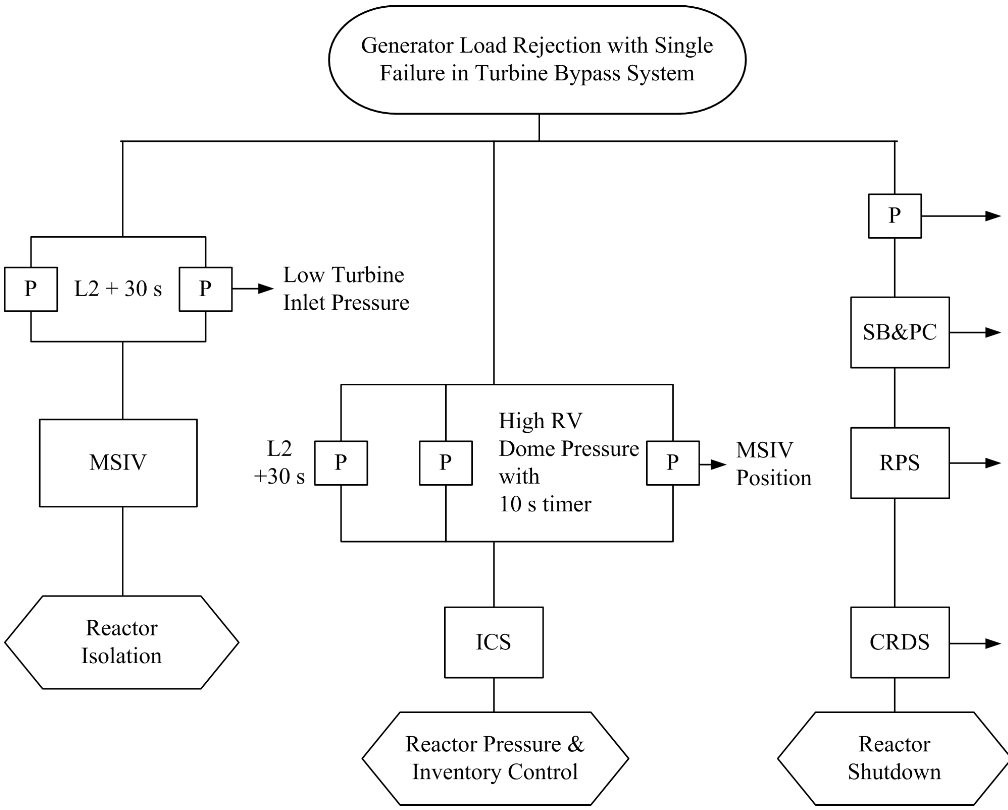


Figure 15.1-5. Event Diagram – Generator Load Rejection with a Single Failure in the Turbine Bypass System

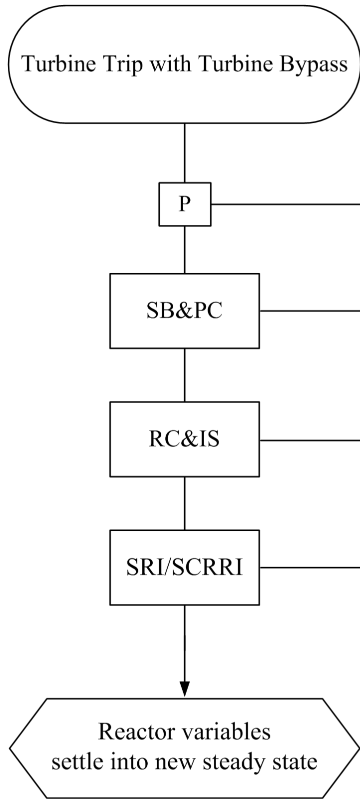


Figure 15.1-6. Event Diagram – Turbine Trip with Turbine Bypass

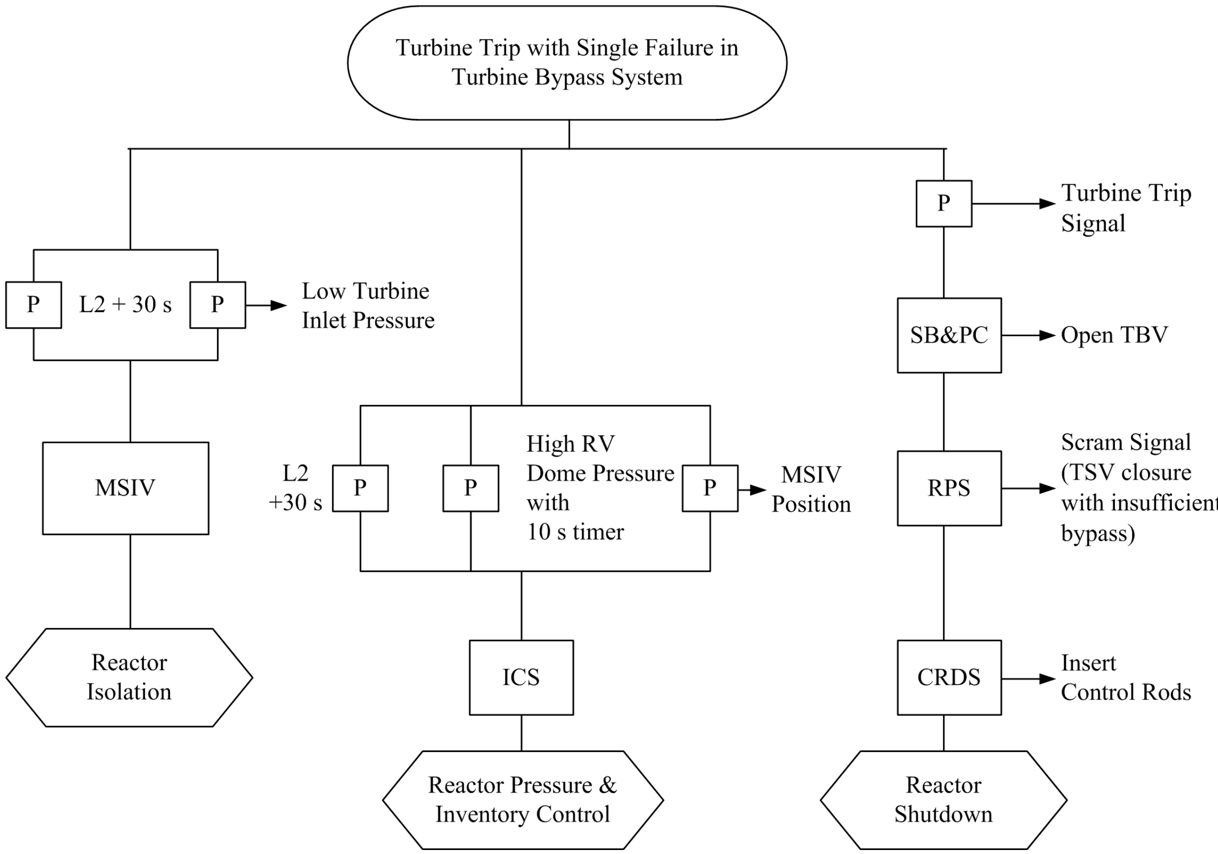


Figure 15.1-7. Event Diagram – Turbine Trip with a Single Failure in the Turbine Bypass System

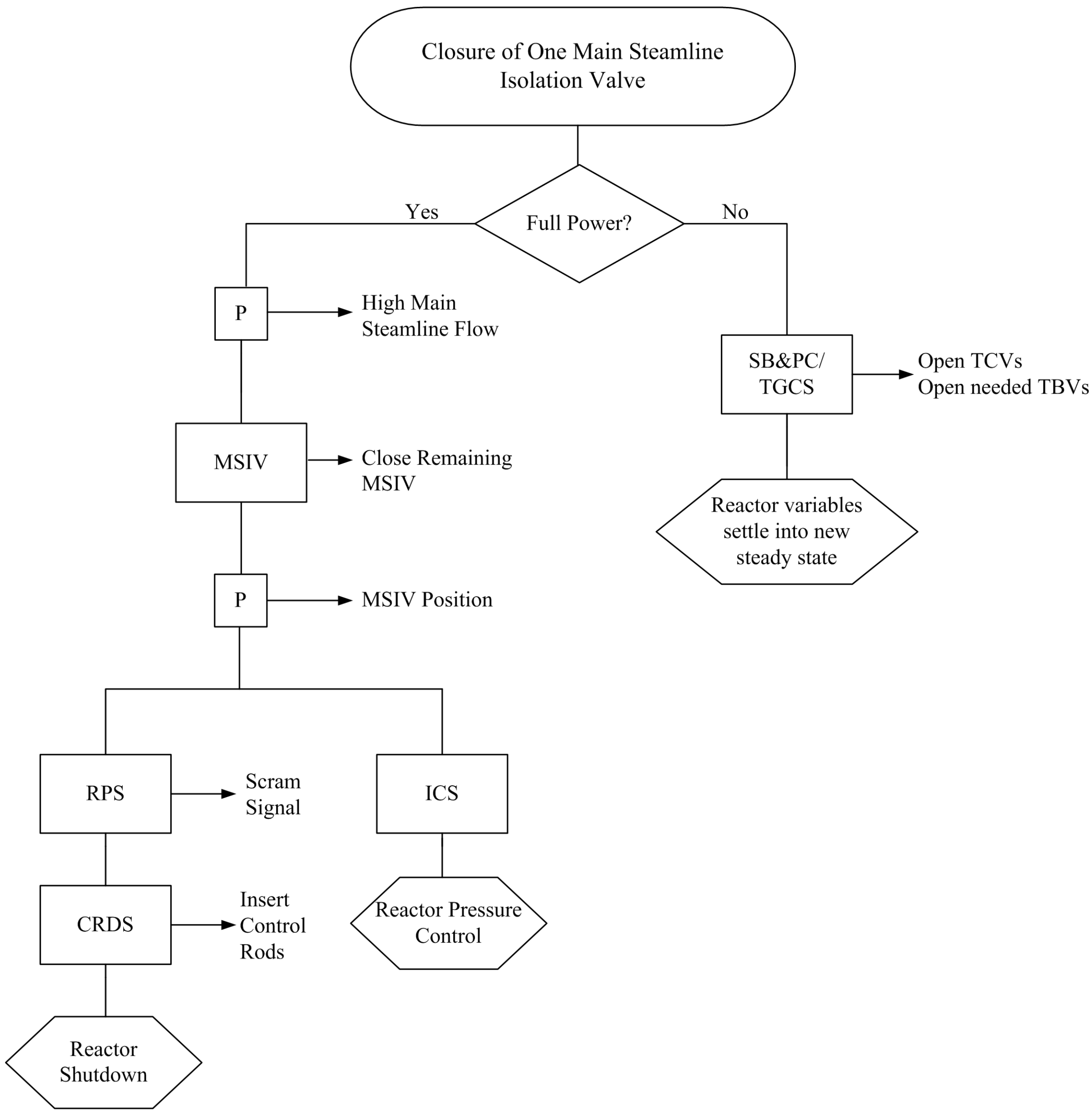


Figure 15.1-8. Event Diagram – Closure of One Main Steamline Isolation Valve

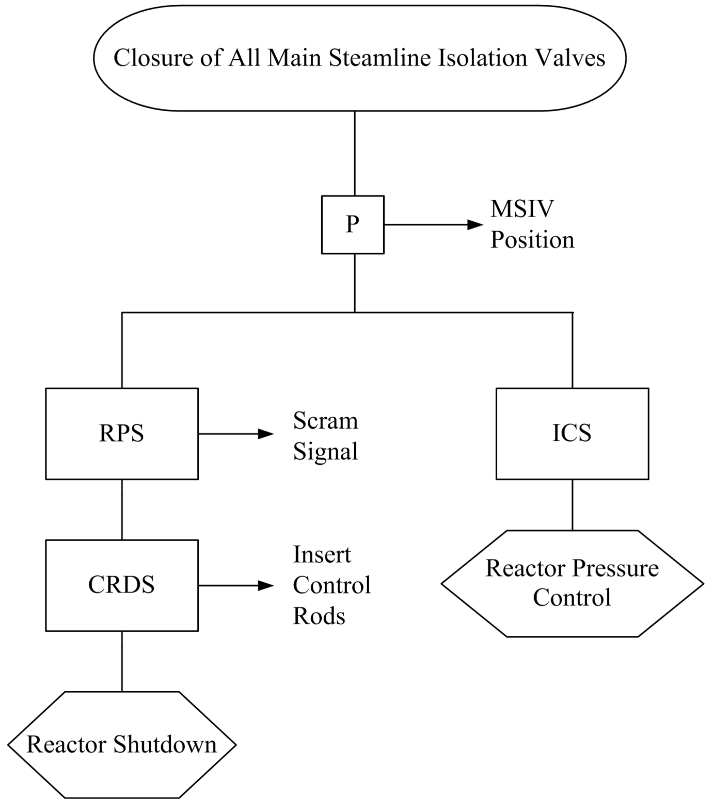


Figure 15.1-9. Event Diagram – Closure of All Main Steamline Isolation Valves

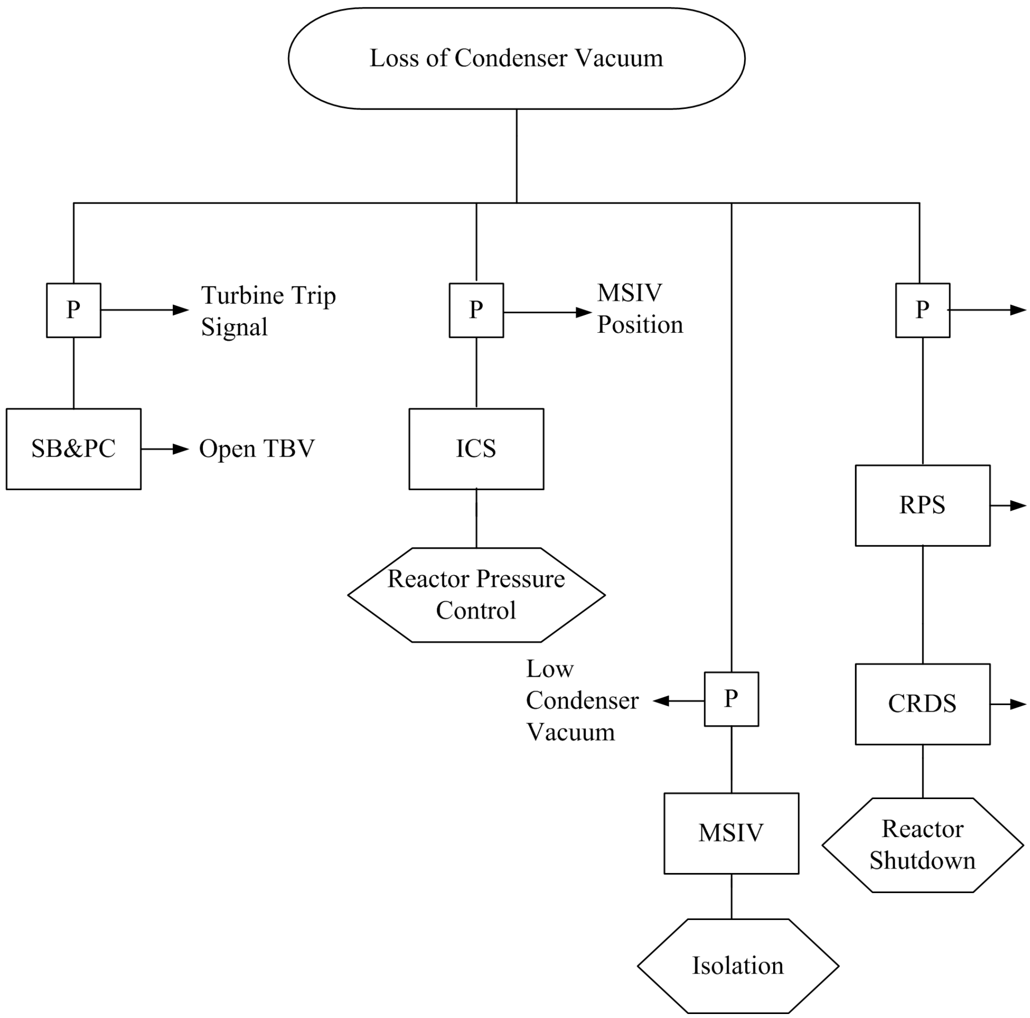


Figure 15.1-10. Event Diagram – Loss of Condenser Vacuum

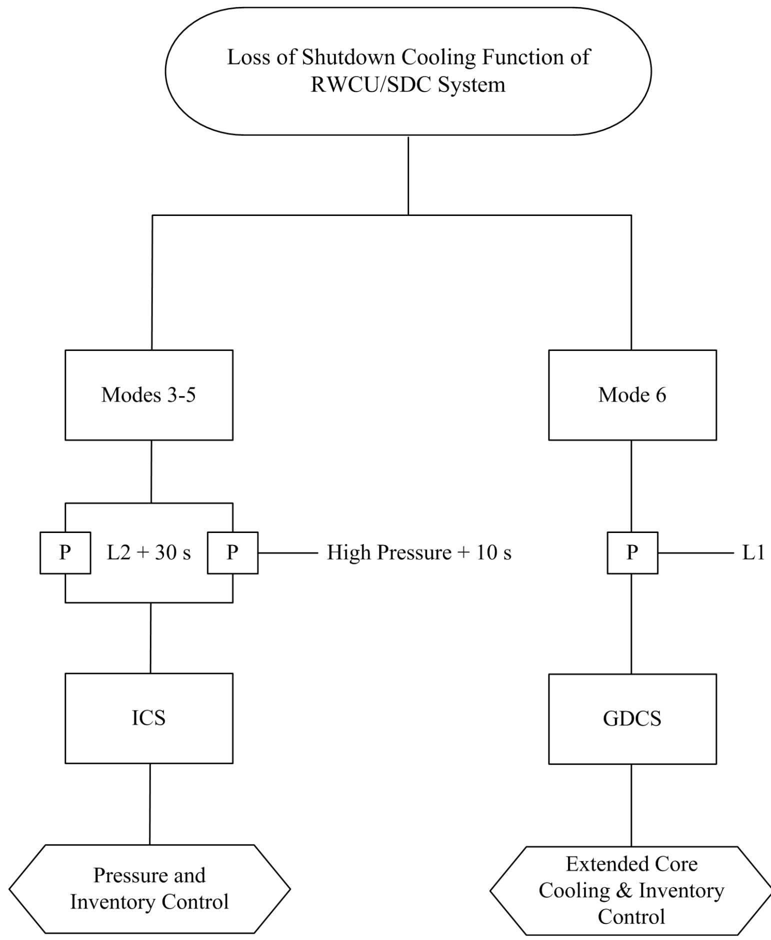


Figure 15.1-11. Event Diagram – Loss of Shutdown Cooling Function of RWCU/SDC System

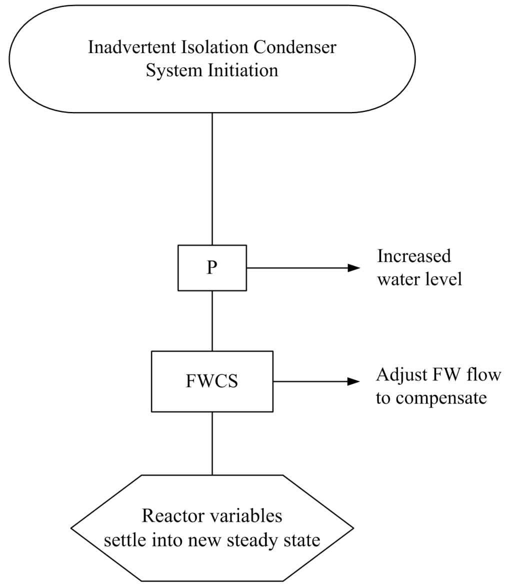


Figure 15.1-12. Event Diagram – Inadvertent Isolation Condenser Initiation

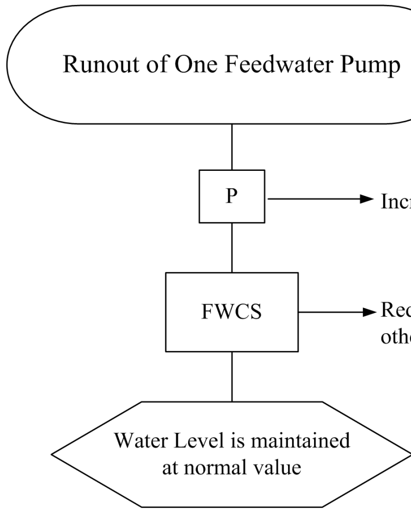


Figure 15.1-13. Event Diagram – Runout of One Feedwater Pump

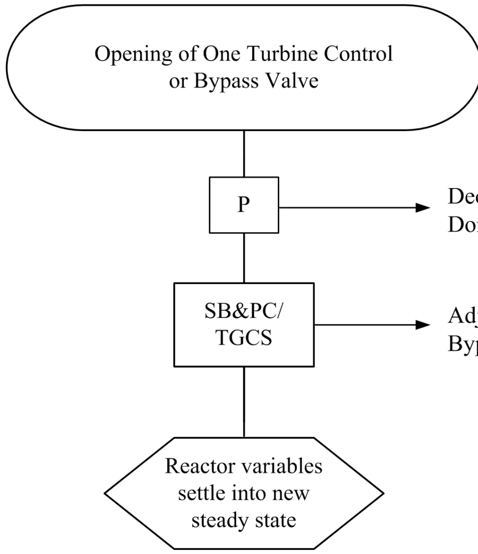


Figure 15.1-14. Event Diagram – Opening of One Turbine Control or Bypass Valve

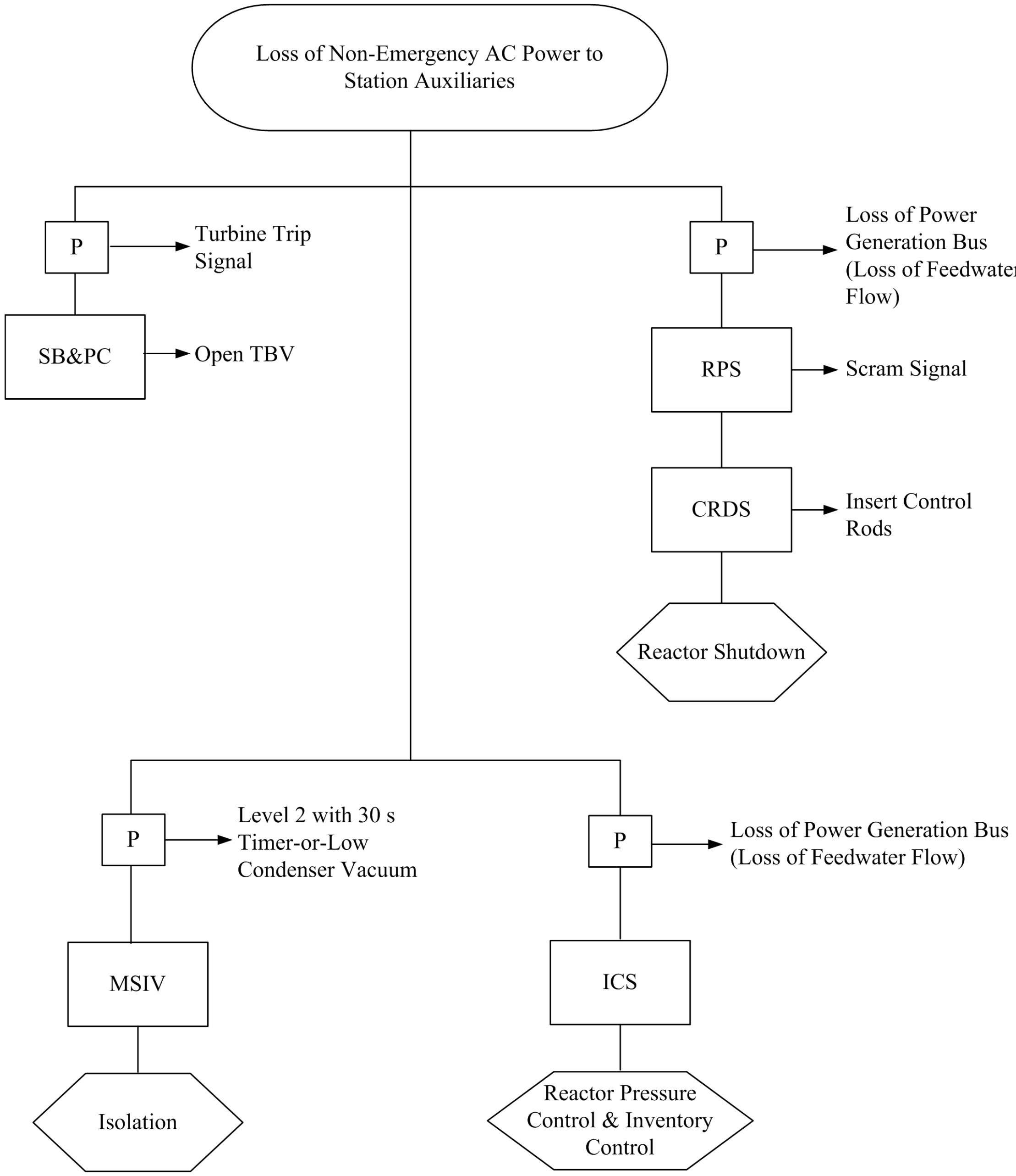


Figure 15.1-15. Event Diagram – Loss of Non-Emergency AC Power to Station Auxiliaries

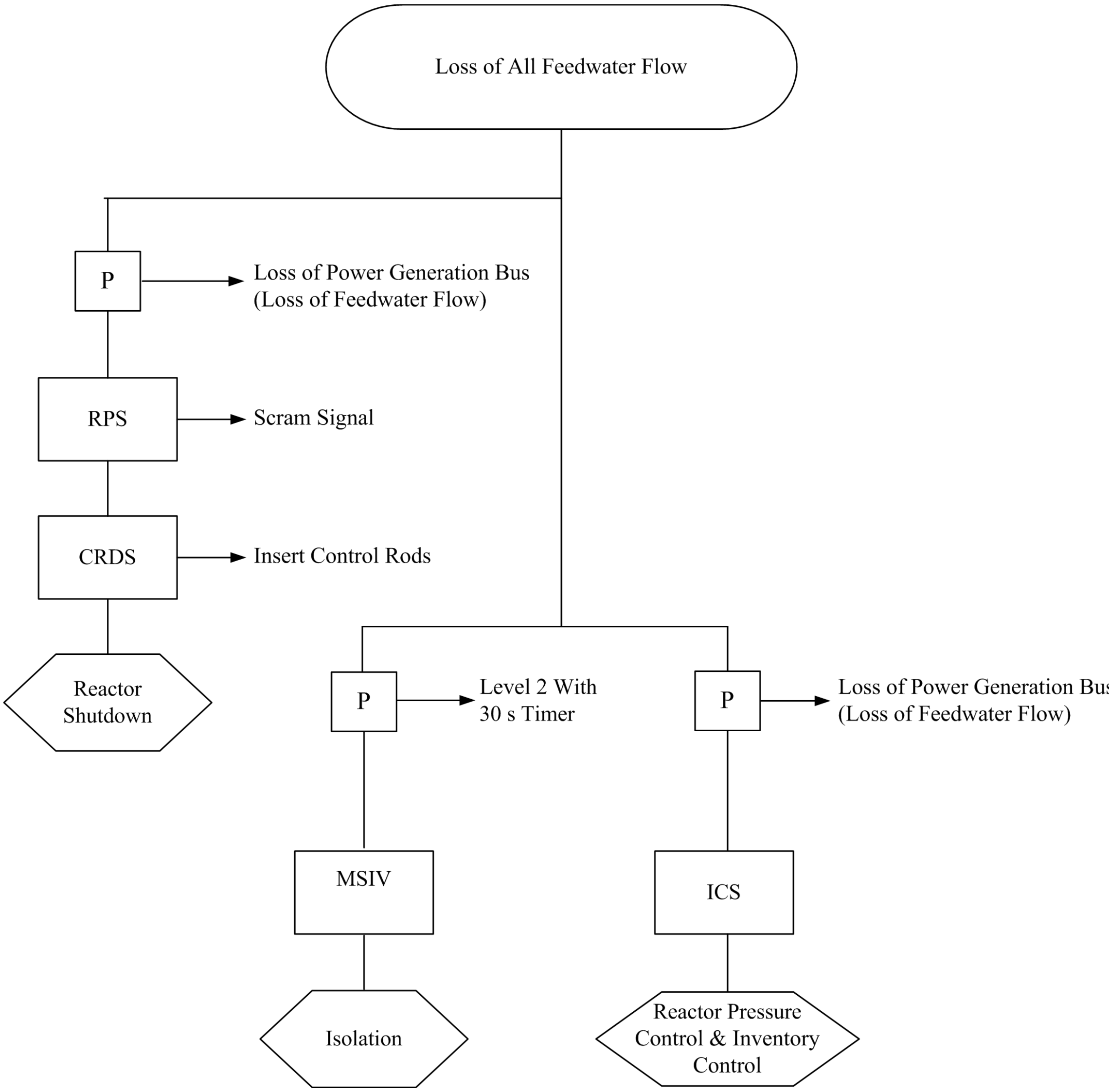


Figure 15.1-16. Event Diagram – Loss of All Feedwater Flow

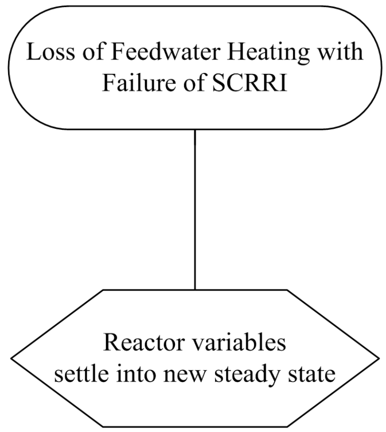


Figure 15.1-17. Event Diagram – Loss of Feedwater Heating With Failure of Selected Control Rod Run-In

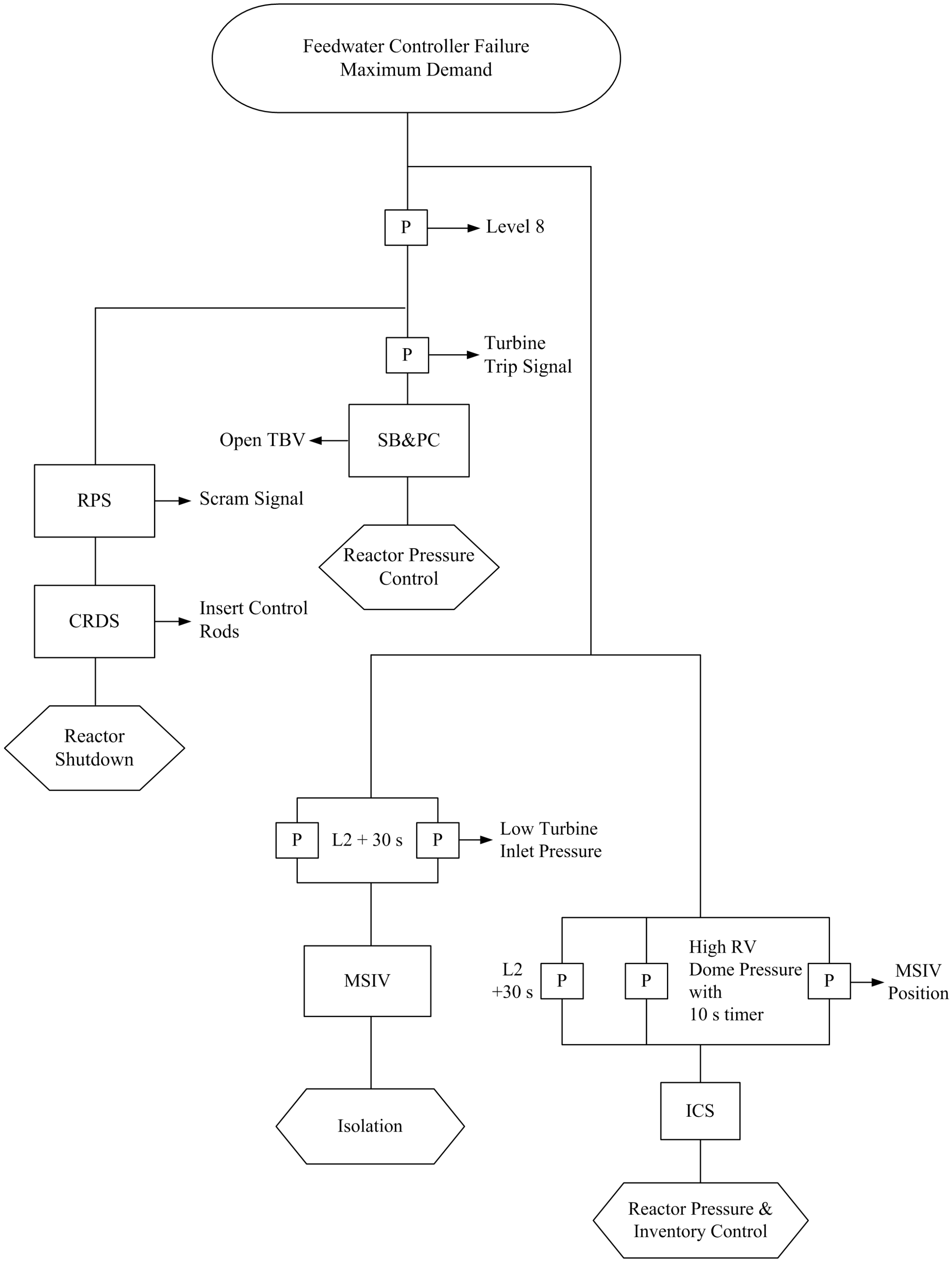


Figure 15.1-18. Event Diagram – Feedwater Controller Failure – Maximum Demand

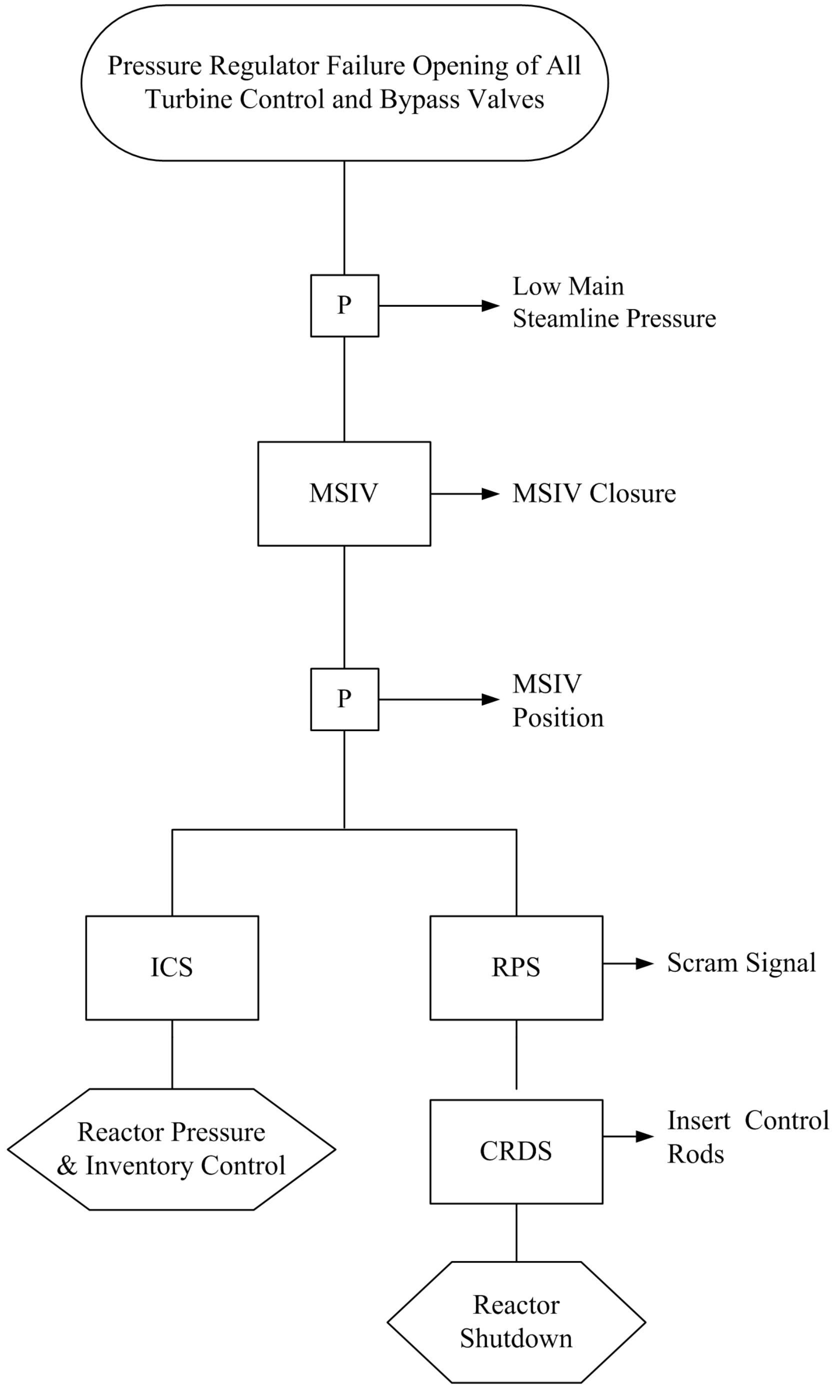


Figure 15.1-19. Event Diagram – Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

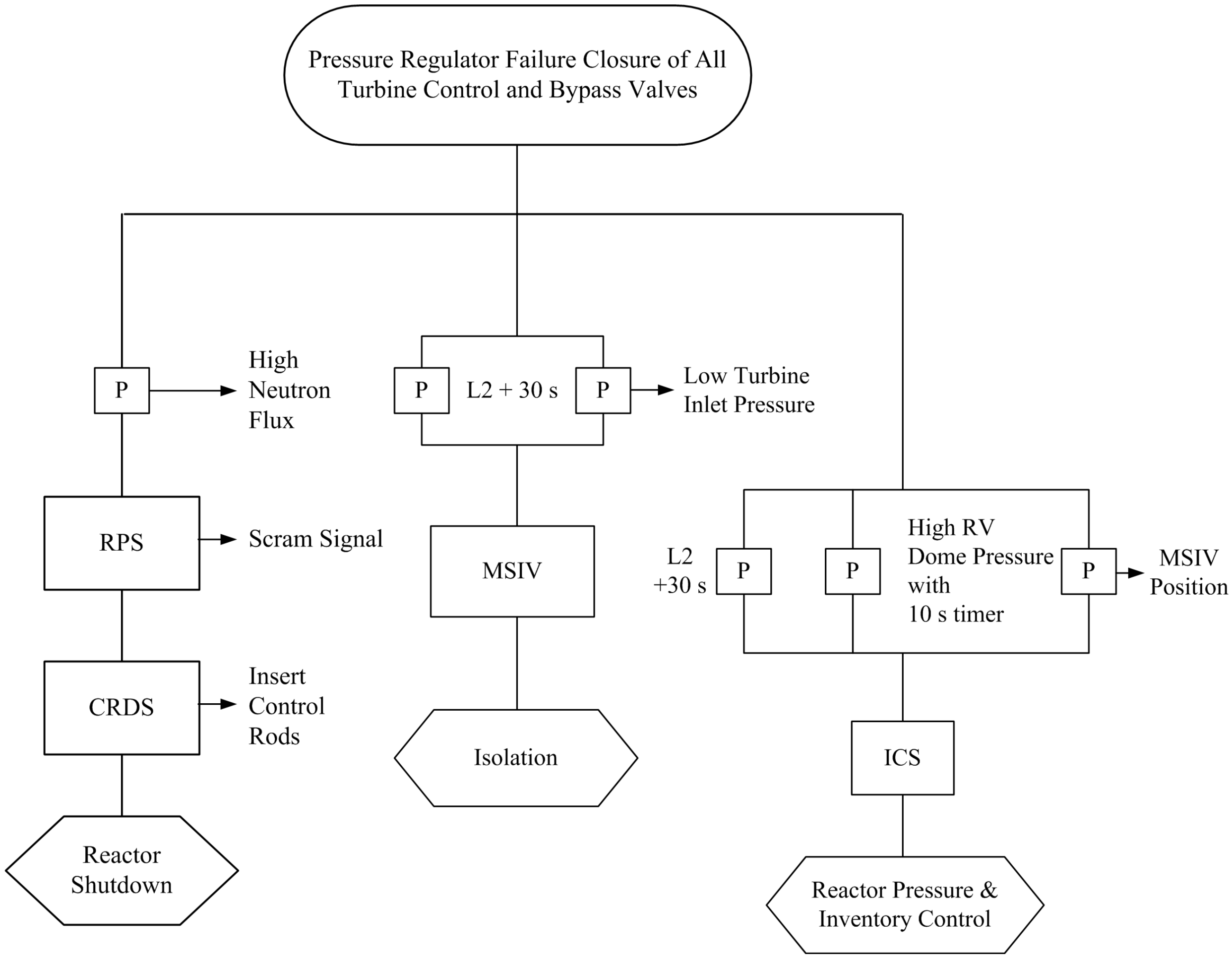


Figure 15.1-20. Event Diagram – Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

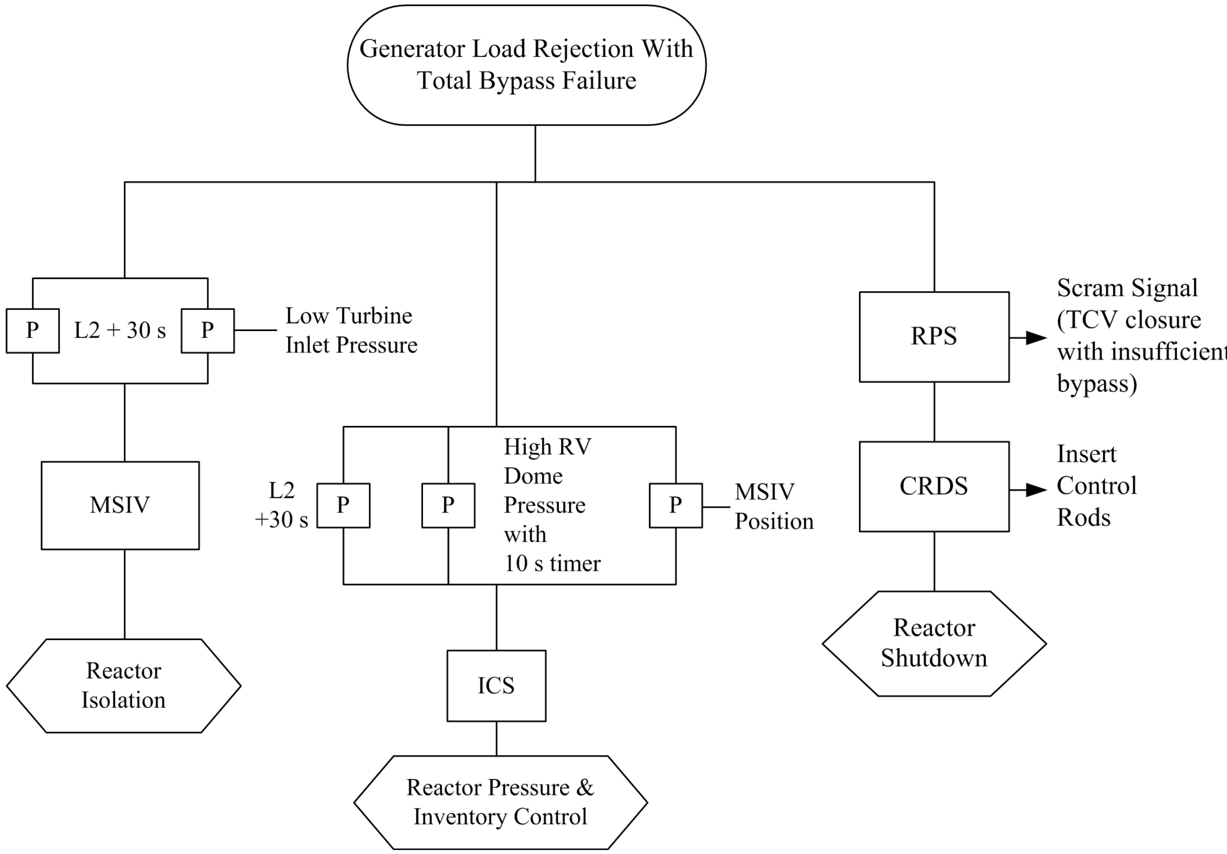


Figure 15.1-21. Event Diagram – Generator Load Rejection with Total Bypass Failure

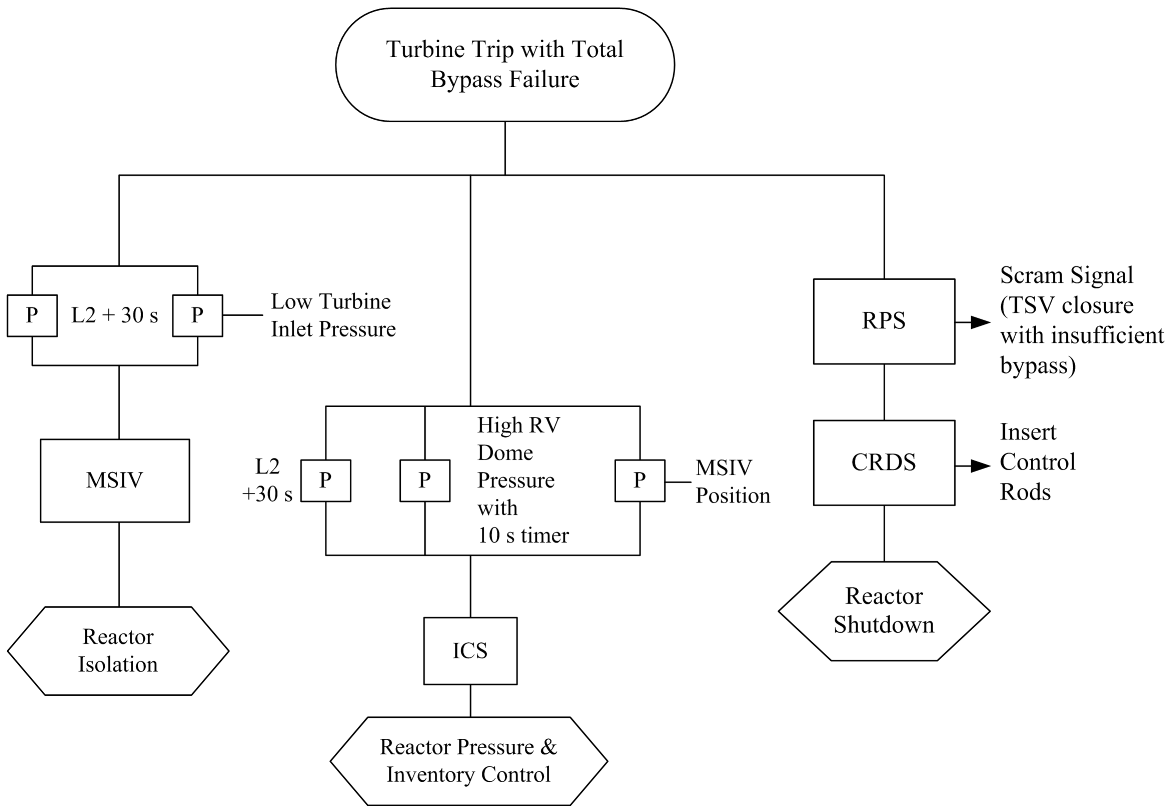


Figure 15.1-22. Event Diagram – Turbine Trip with Total Bypass Failure

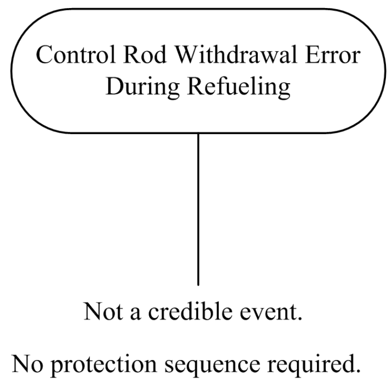


Figure 15.1-23. Event Diagram – Control Rod Withdrawal Error During Refueling

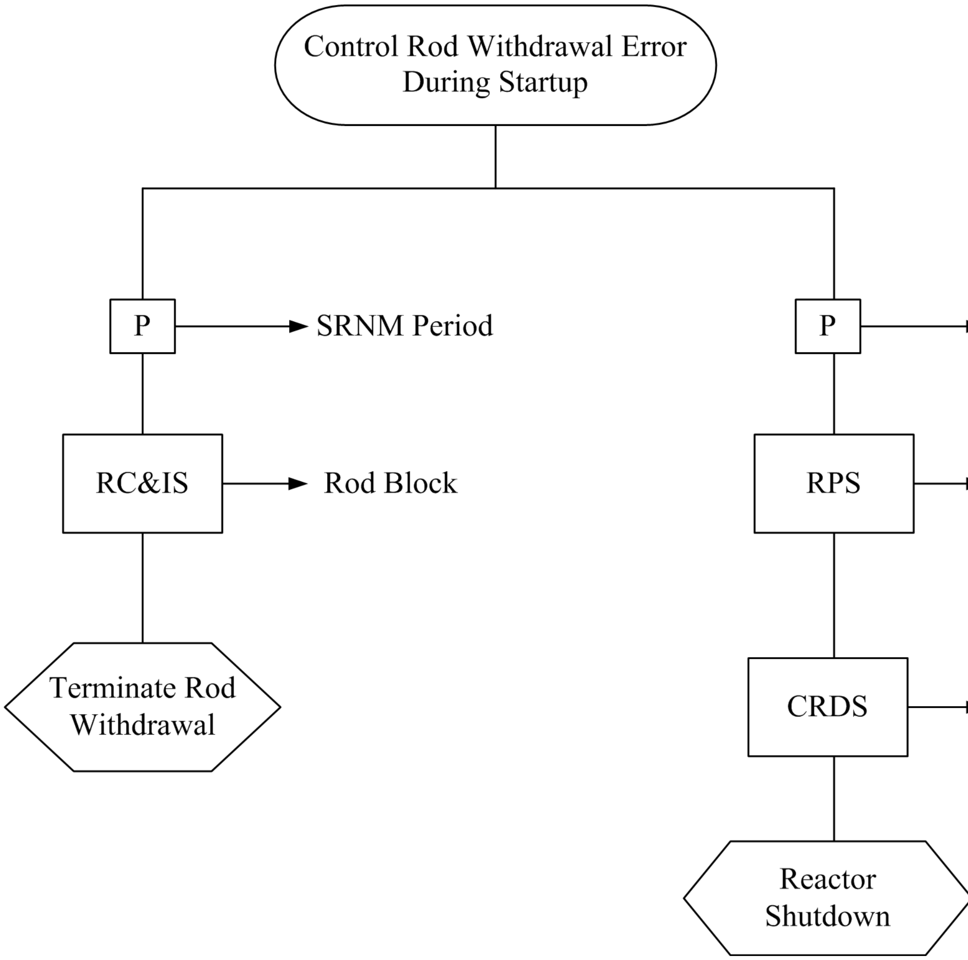


Figure 15.1-24. Event Diagram – Control Rod Withdrawal Error During Startup

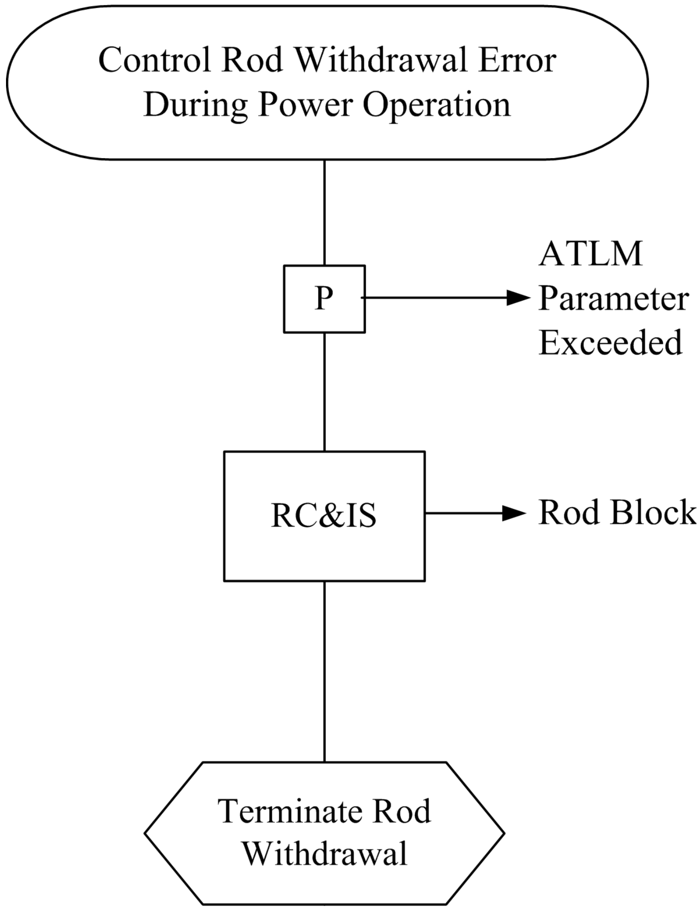


Figure 15.1-25. Event Diagram – Control Rod Withdrawal Error During Power Operation

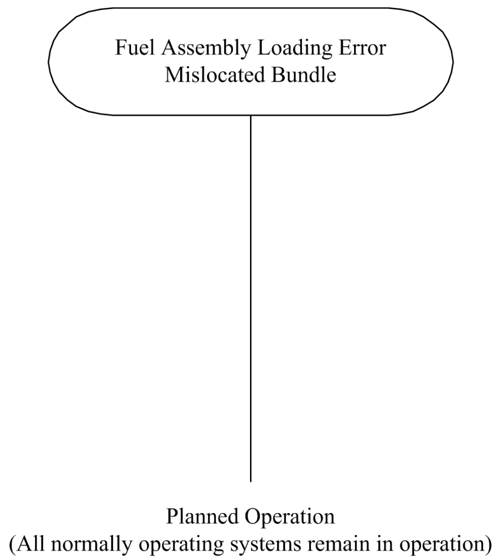


Figure 15.1-26. Event Diagram – Fuel Assembly Loading Error – Mislocated Bundle

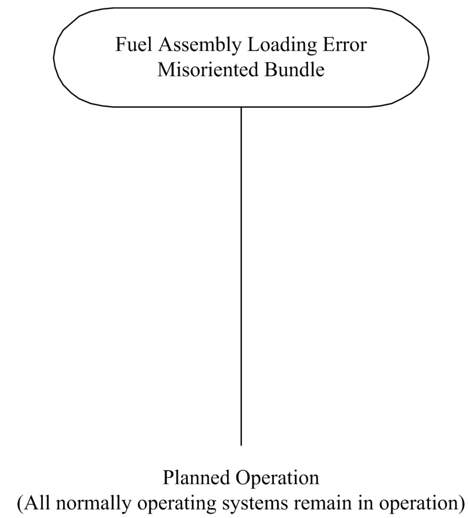


Figure 15.1-27. Event Diagram – Fuel Assembly Loading Error – Misoriented Bundle

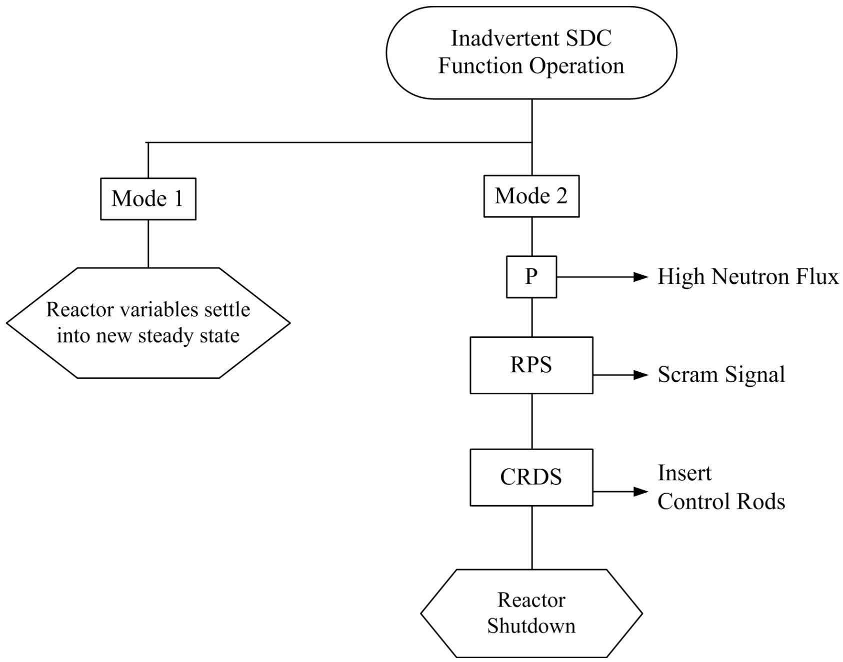


Figure 15.1-28. Event Diagram – Inadvertent SDC Function Operation

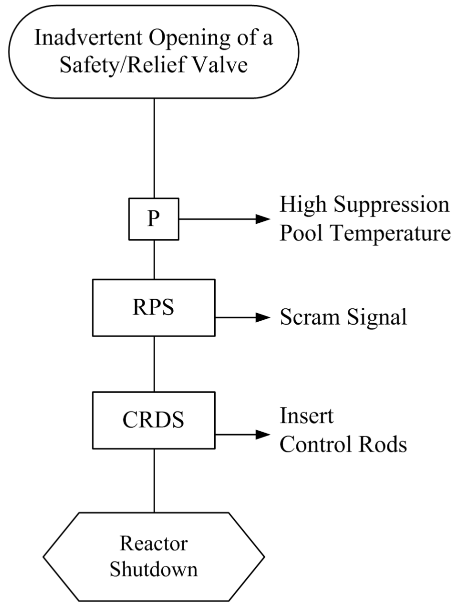


Figure 15.1-29. Event Diagram – Inadvertent Opening of a Safety Relief Valve

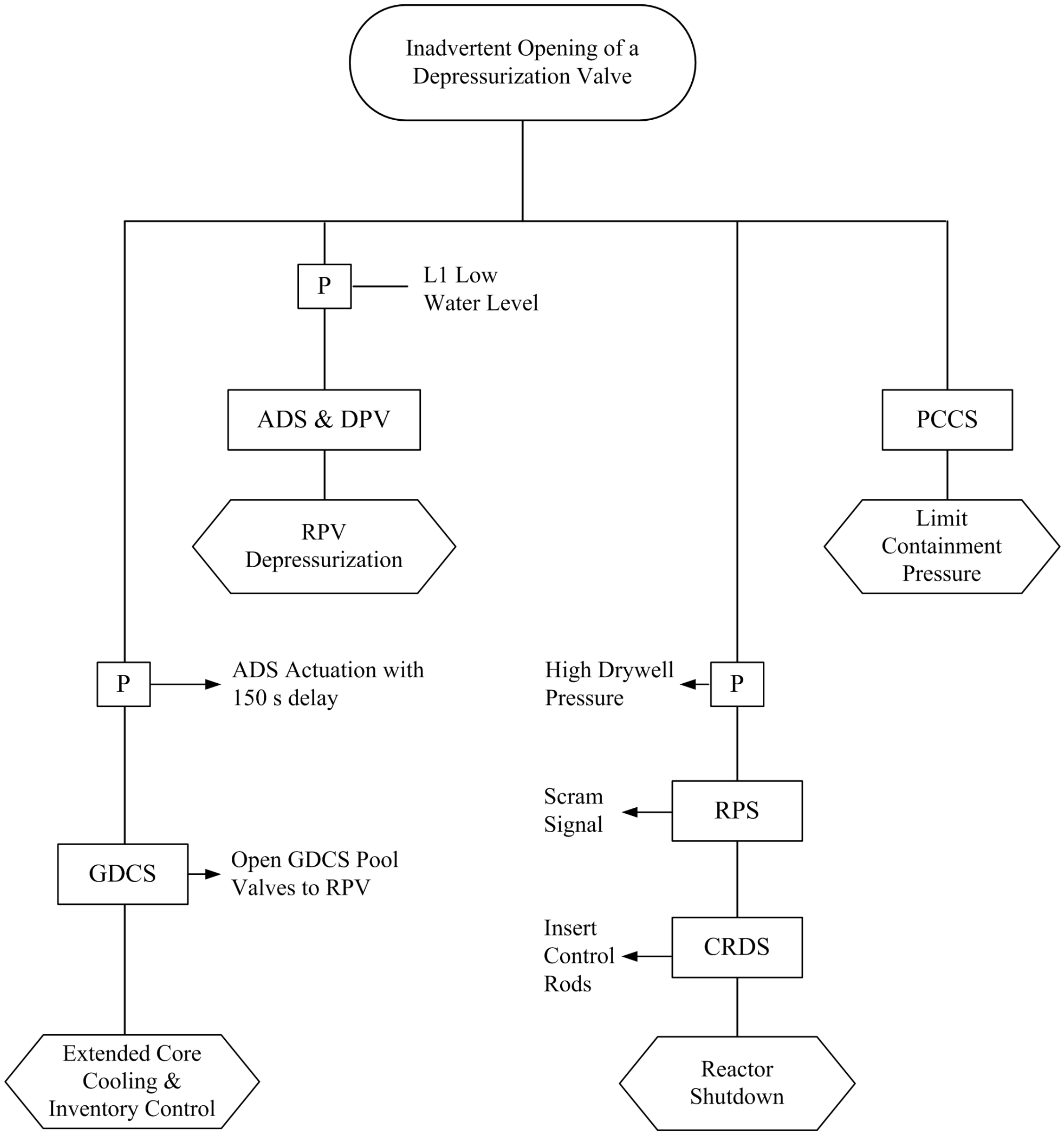


Figure 15.1-30. Event Diagram – Inadvertent Opening of a Depressurization Valve

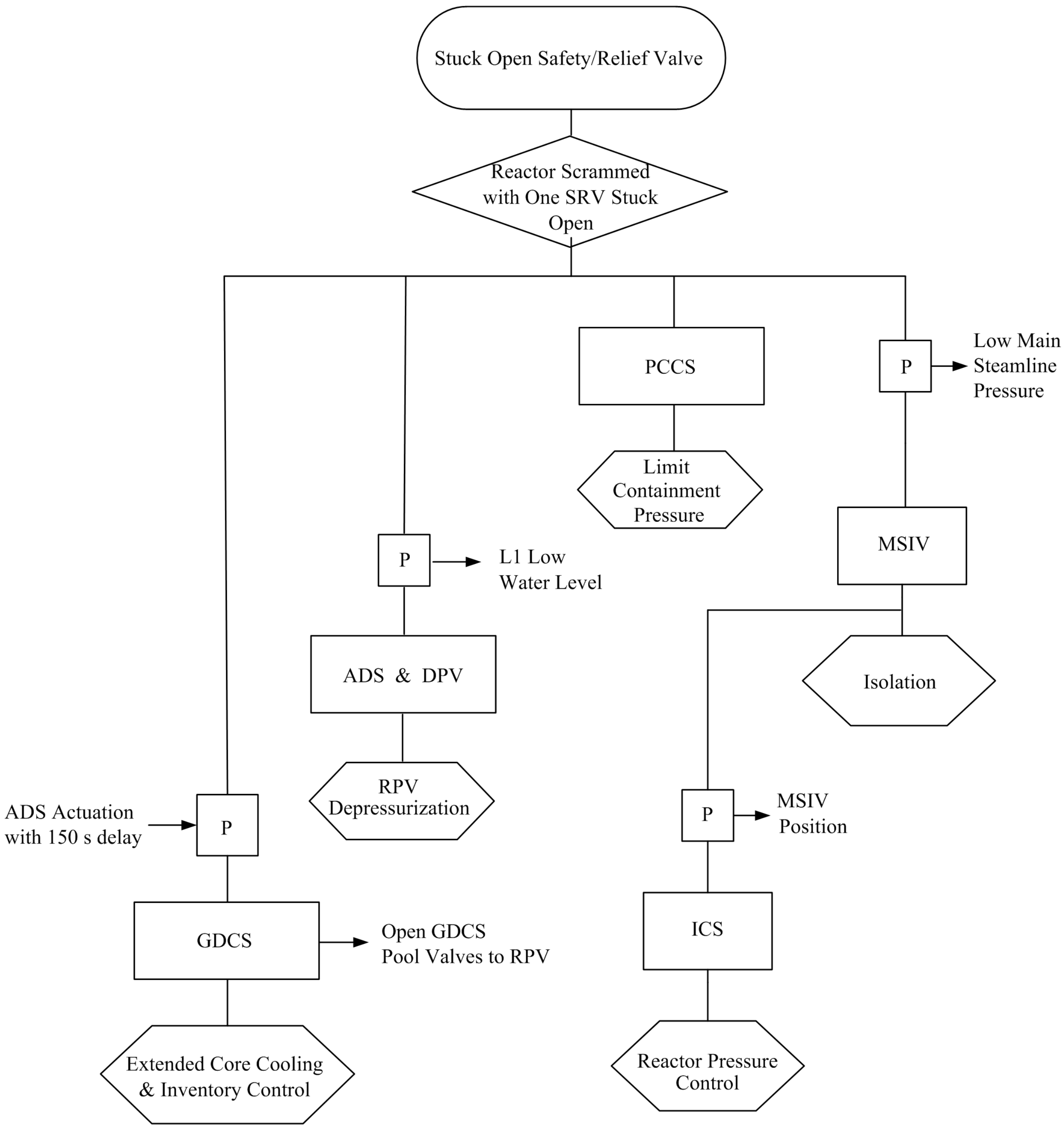


Figure 15.1-31. Event Diagram – Stuck Open Safety Relief Valve

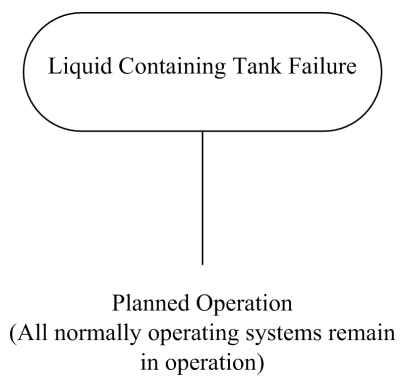


Figure 15.1-32. Event Diagram – Liquid-Containing Tank Failure

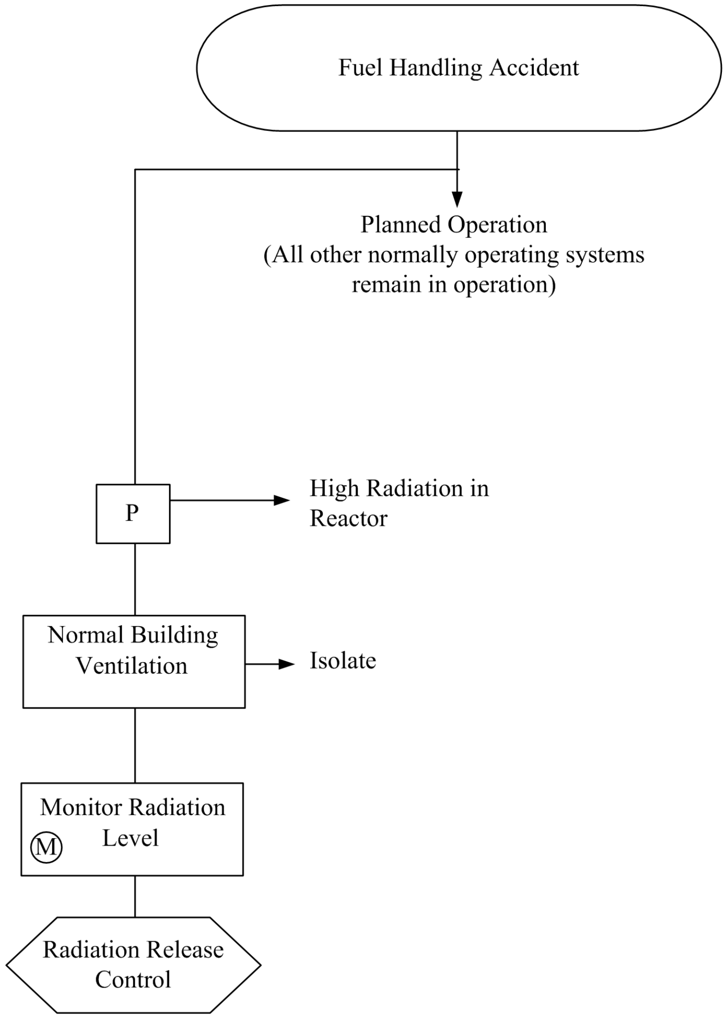


Figure 15.1-33. Event Diagram – Fuel Handling Accident

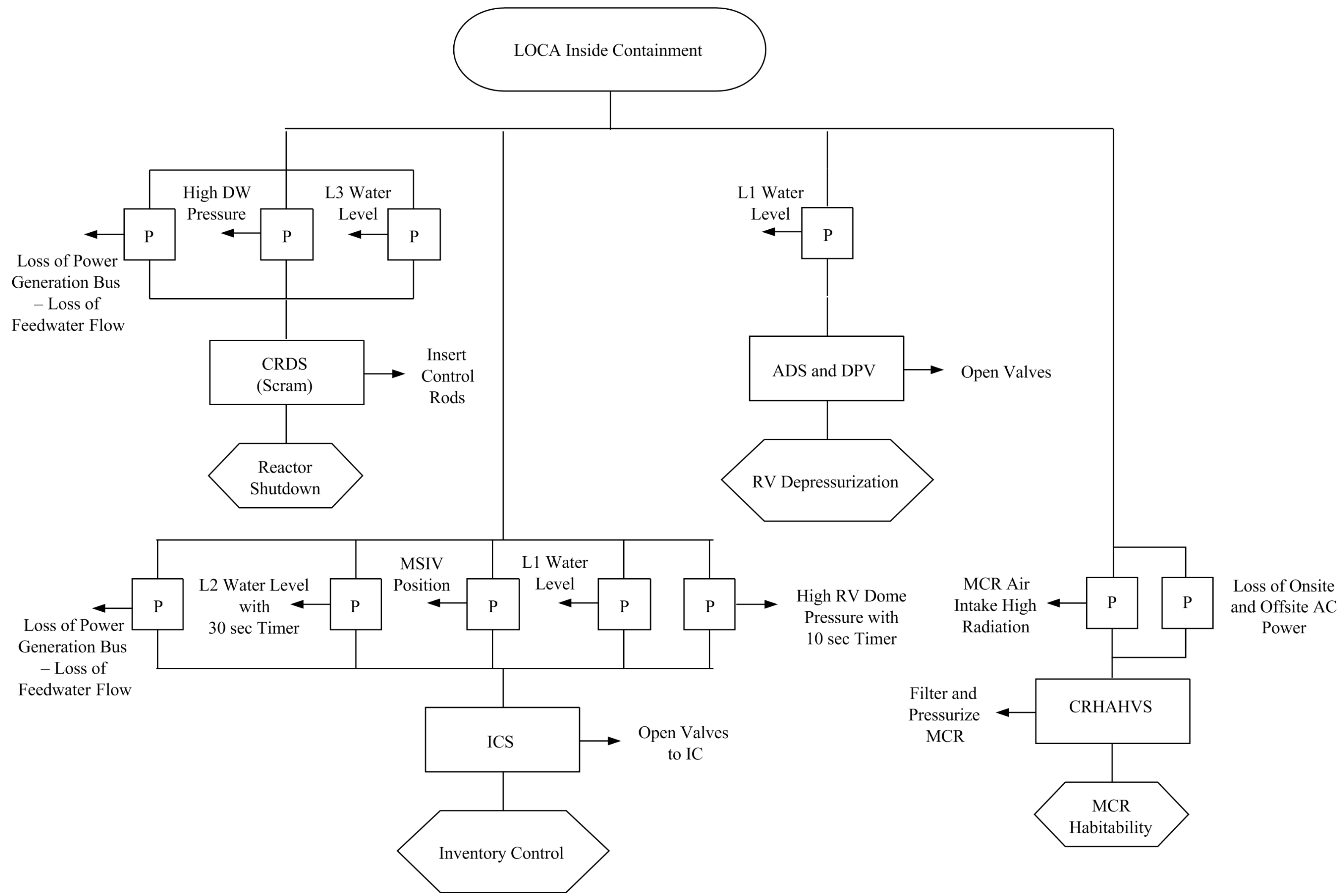


Figure 15.1-34a. Event Diagram – Loss-of-Coolant Accident Inside Containment

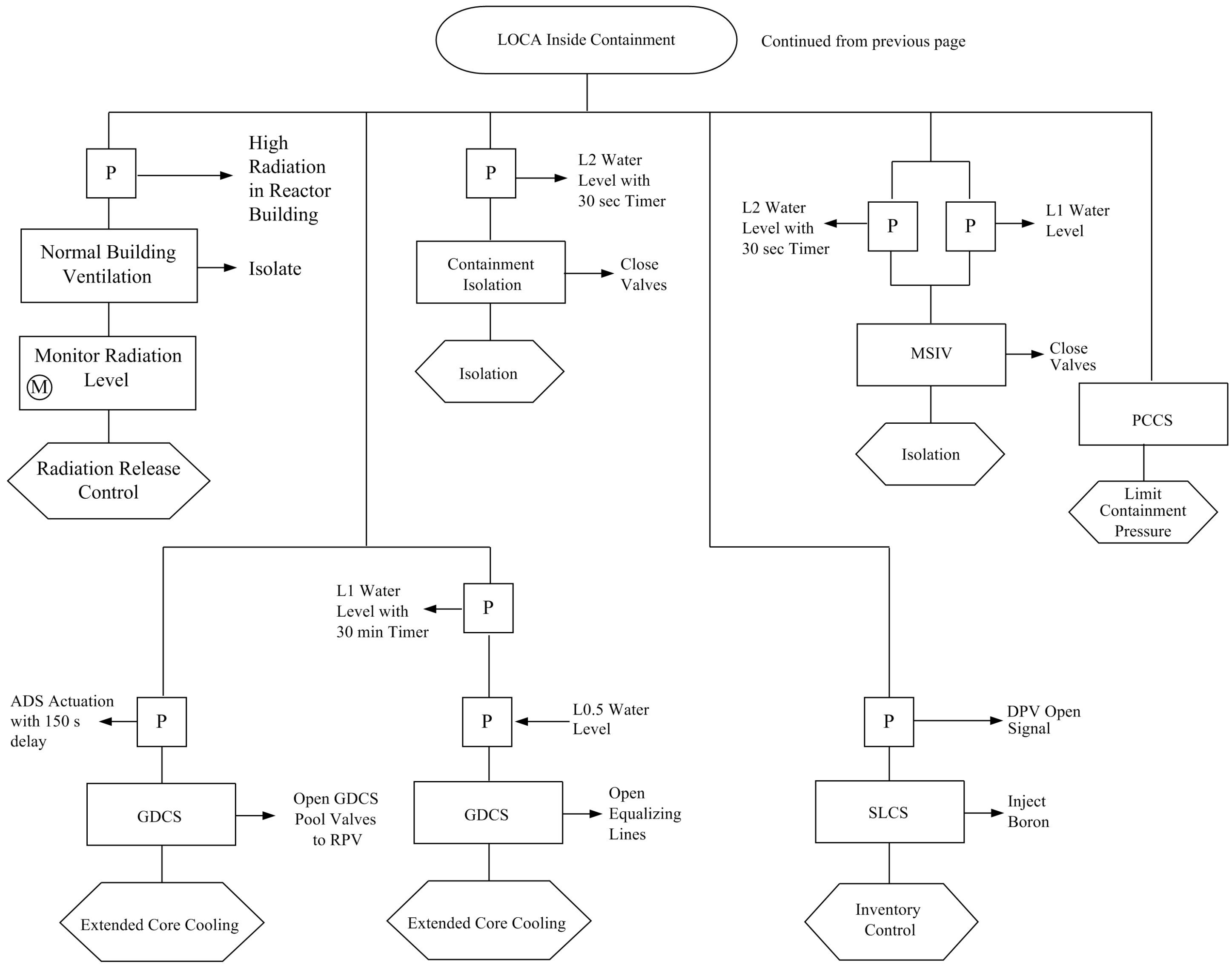


Figure 15.1-34b. Event Diagram – Loss-of-Coolant Accident Inside Containment

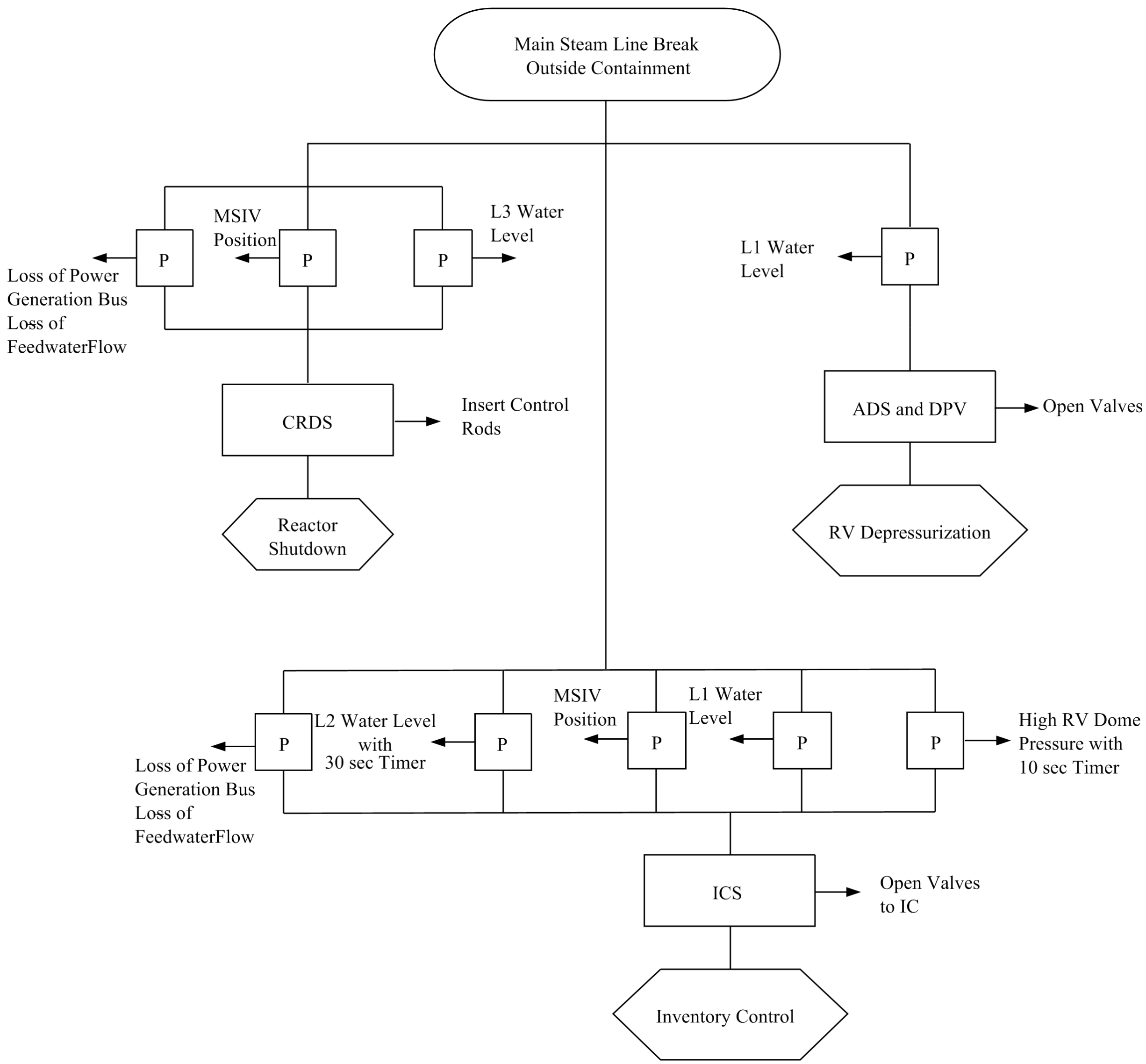


Figure 15.1-35a. Event Diagram – Main Steamline Break Outside Containment

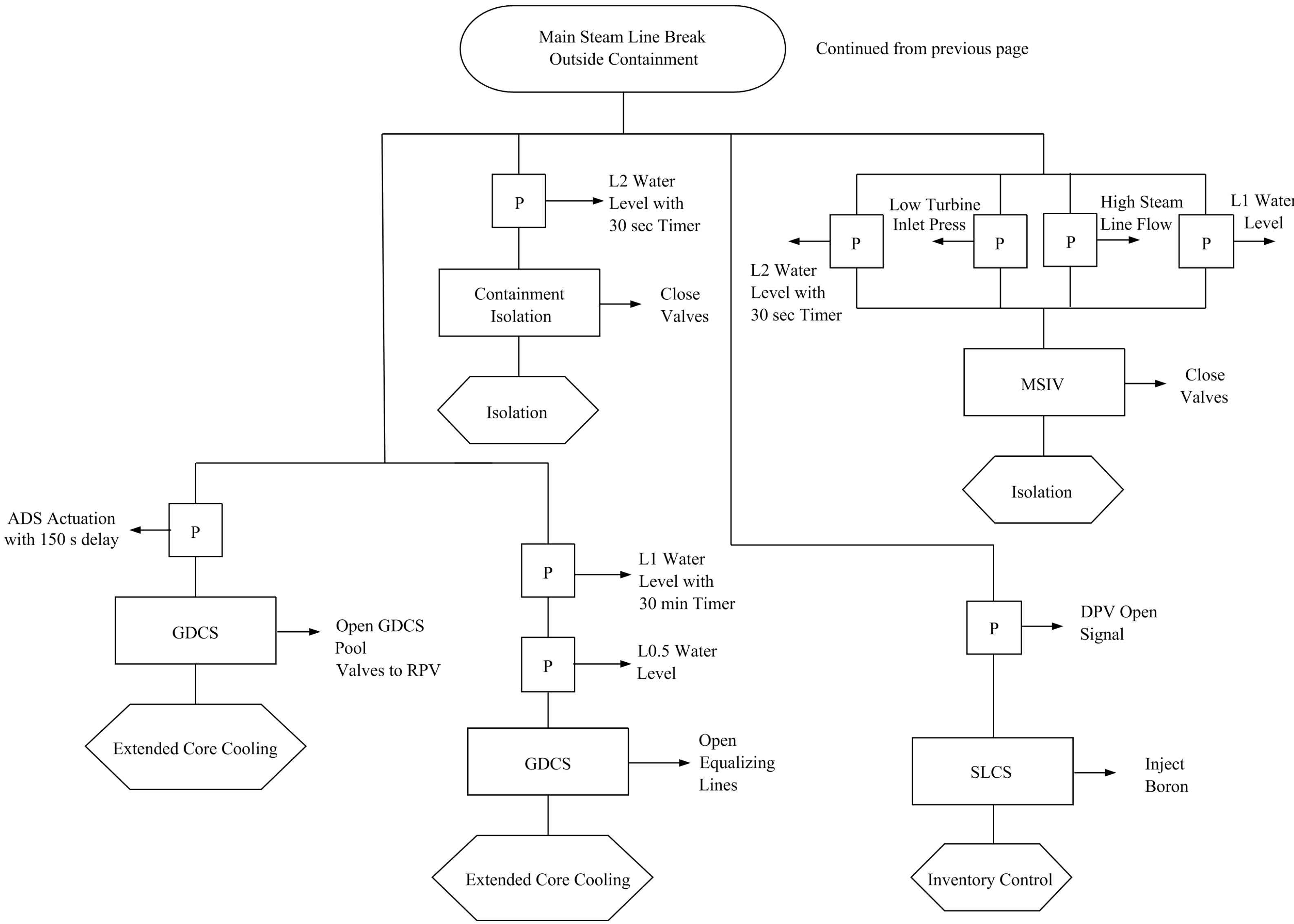


Figure 15.1-35b. Event Diagram – Main Steamline Break Outside Containment

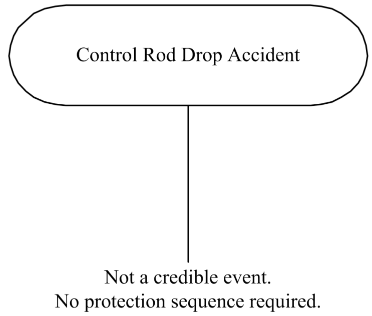


Figure 15.1-36. Event Diagram – Control Rod Drop Accident

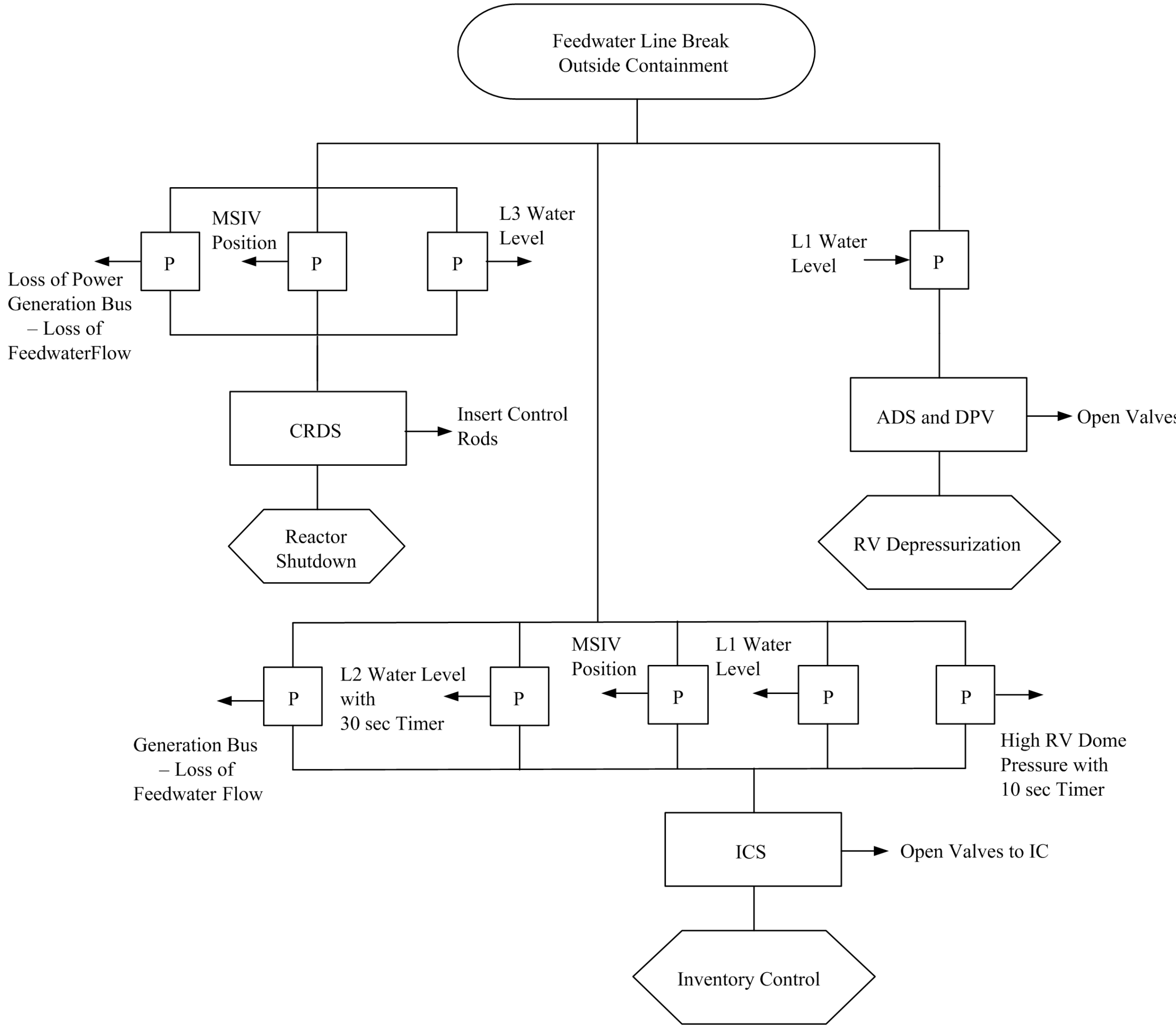


Figure 15.1-37a. Event Diagram – Feedwater Line Break Outside Containment

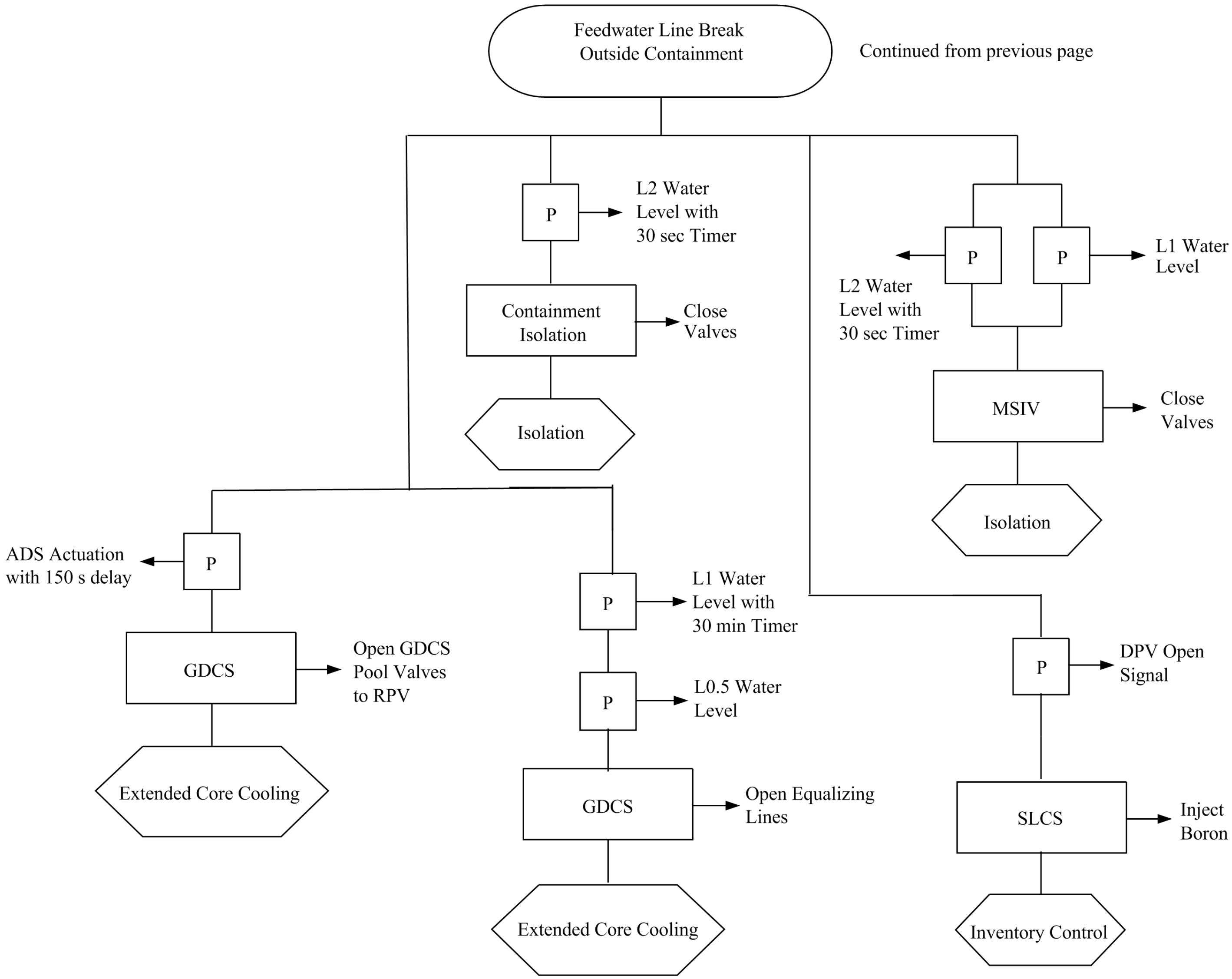


Figure 15.1-37b. Event Diagram – Feedwater Line Break Outside Containment

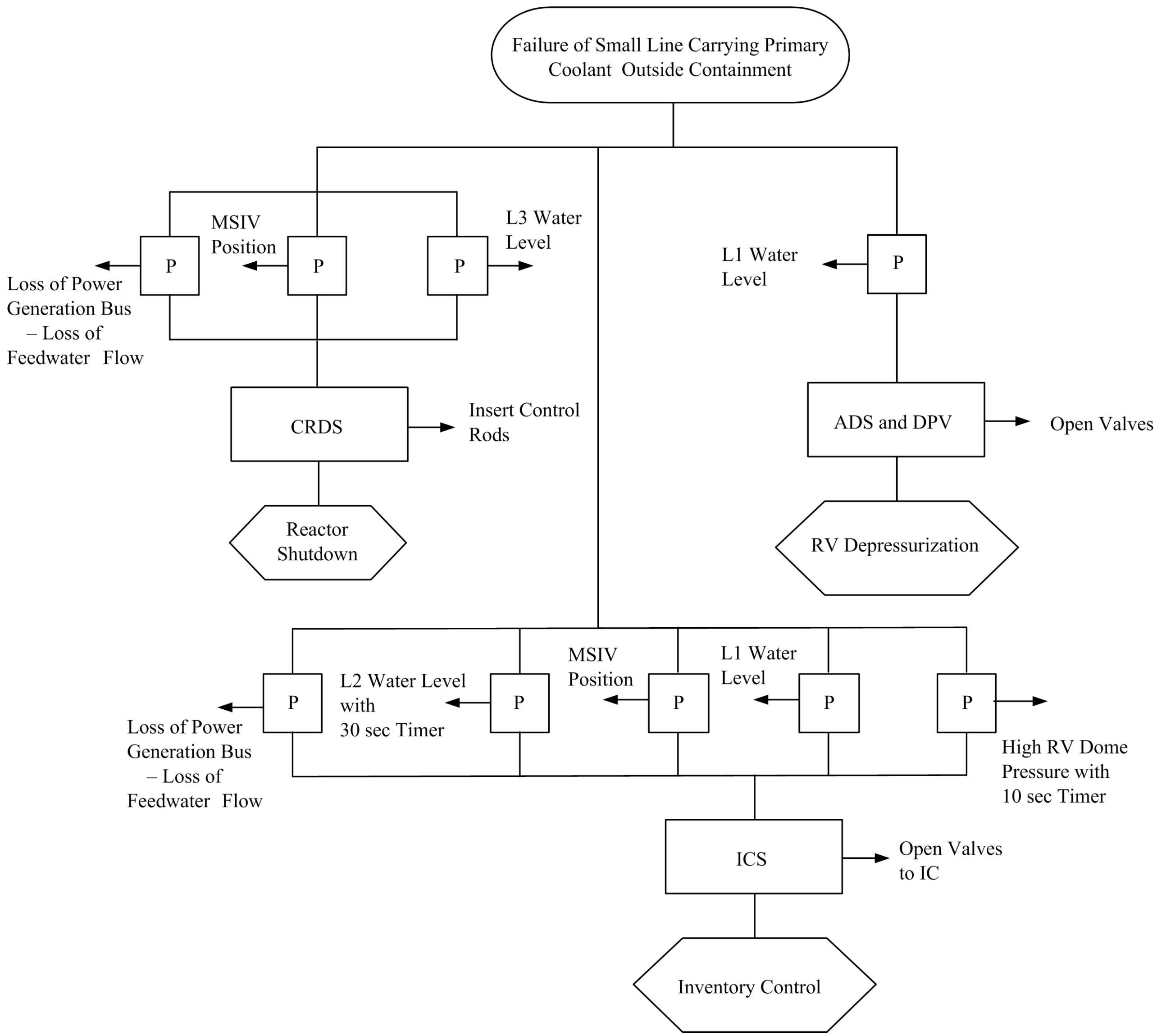


Figure 15.1-38a. Event Diagram – Failure of Small Line Carrying Primary Coolant Outside Containment

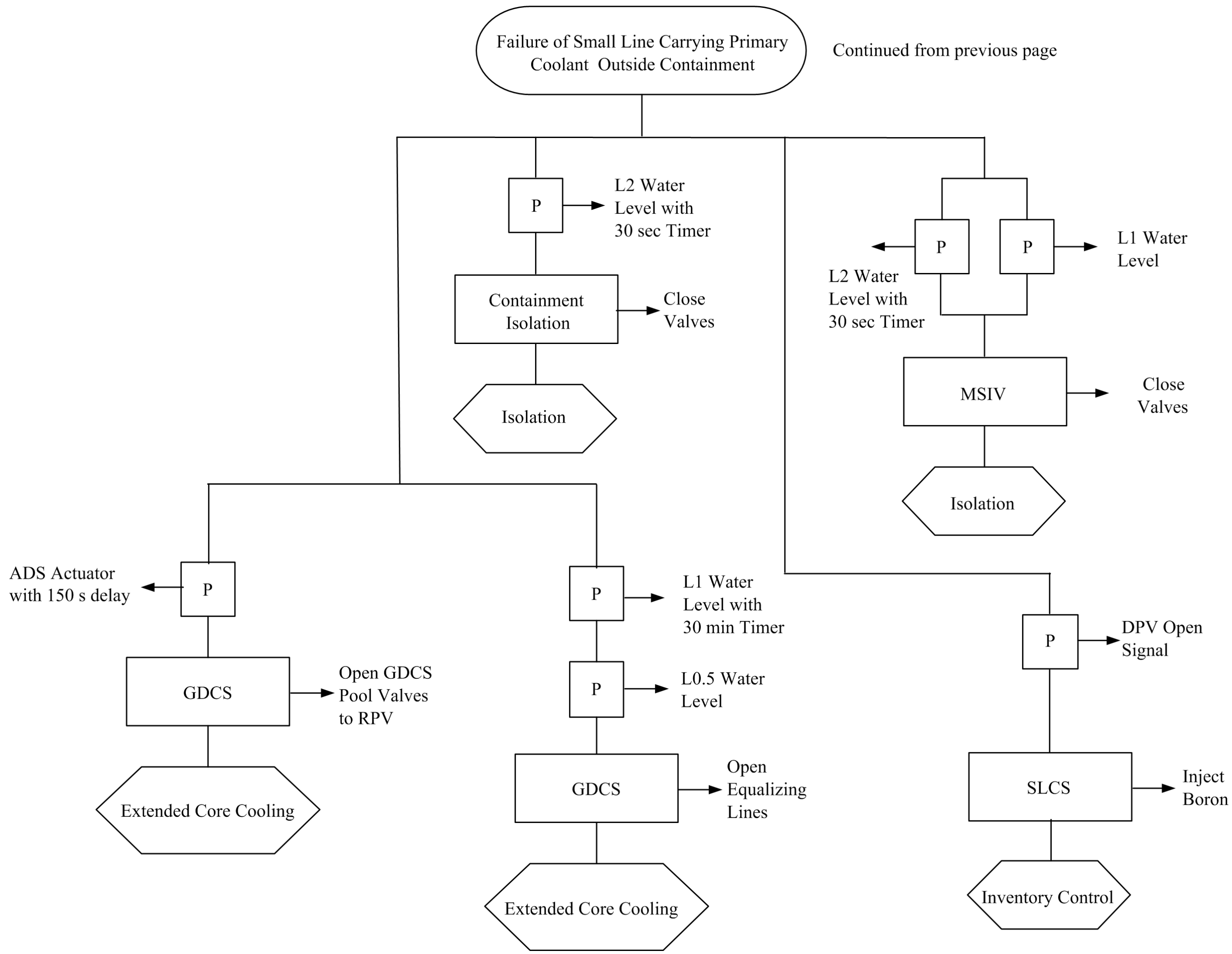


Figure 15.1-38b. Event Diagram – Failure of Small Line Carrying Primary Coolant Outside Containment

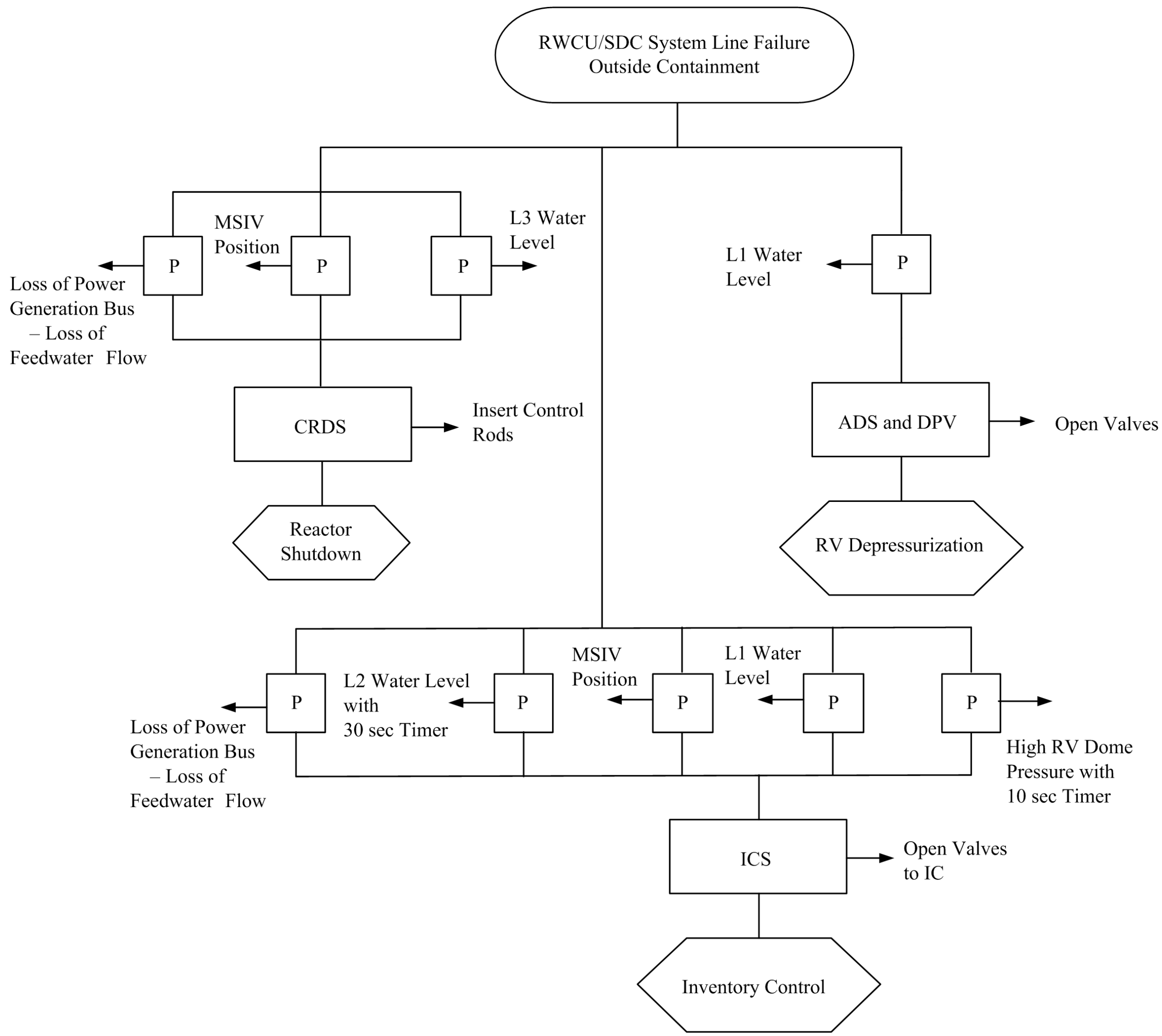


Figure 15.1-39a. Event Diagram – RWCU/SDC System Line Failure Outside Containment

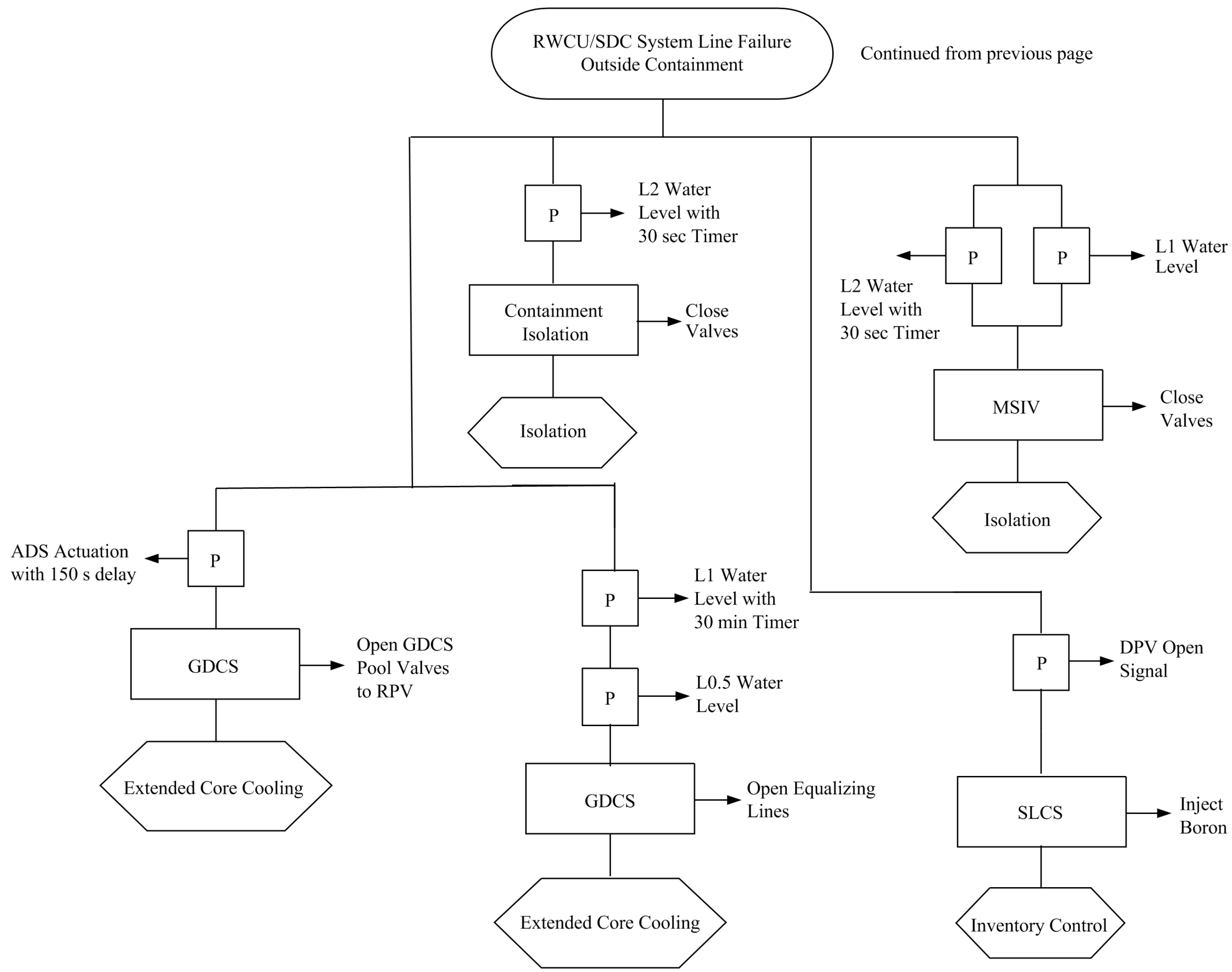


Figure 15.1-39b. Event Diagram – RWCU/SDC System Line Failure Outside Containment

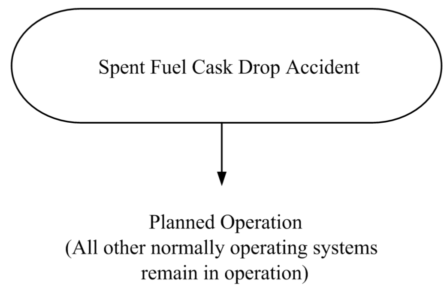


Figure 15.1-40. Event Diagram – Spent Fuel Cask Drop Accident

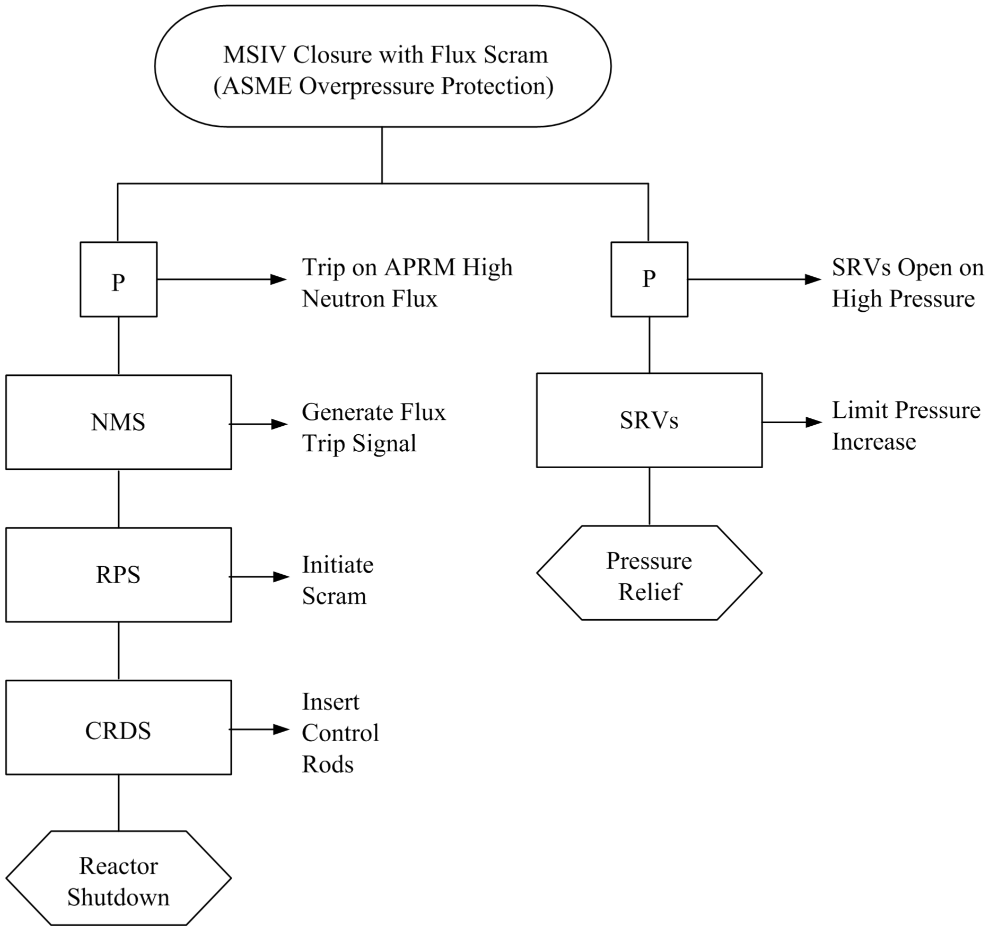


Figure 15.1-41. Event Diagram – MSIV Closure With Flux Scram (Overpressure Protection)

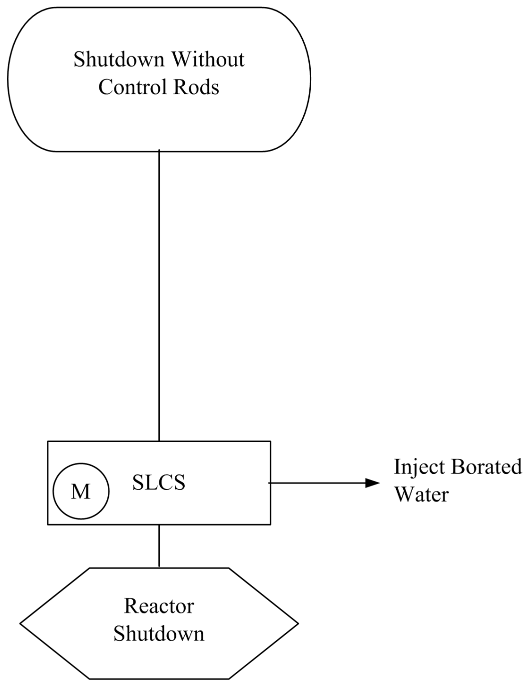


Figure 15.1-42. Event Diagram – Shutdown Without Control Rods (Standby Liquid Control System Capability)

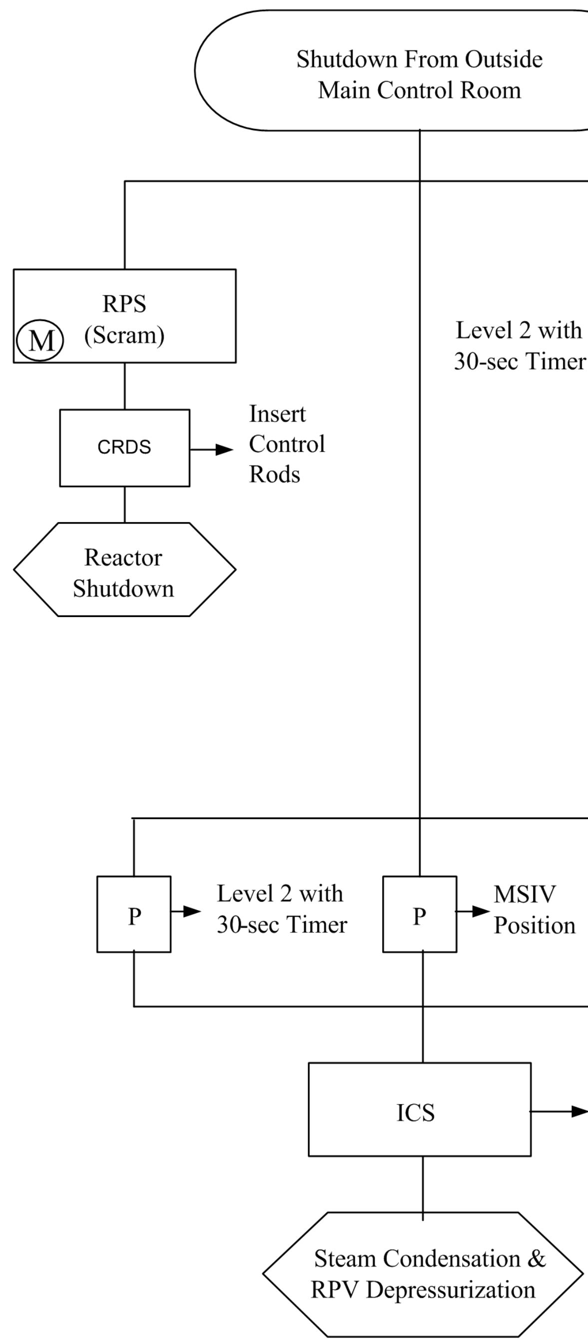


Figure 15.1-43. Event Diagram – Shutdown from Outside Main Control Room

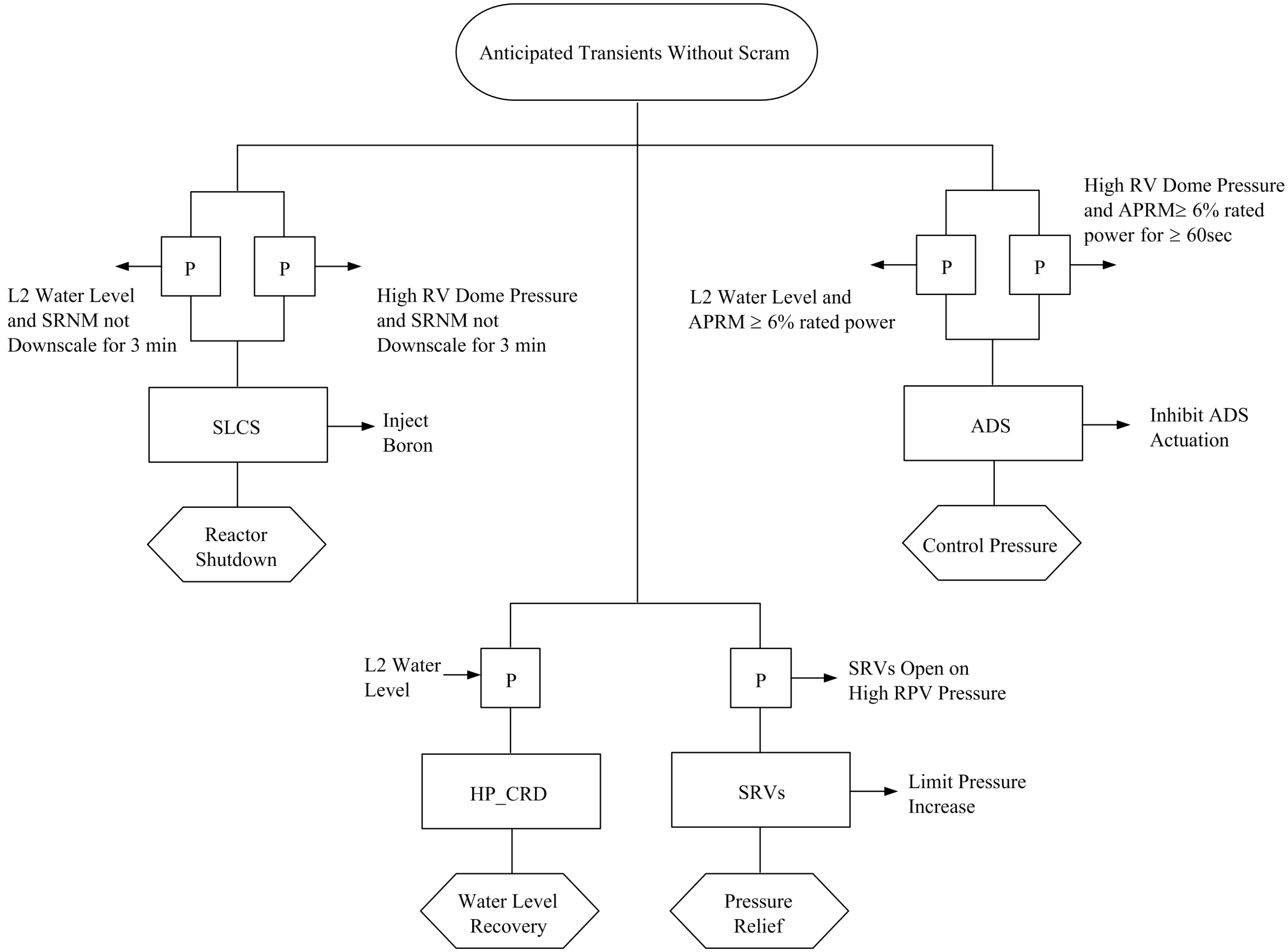


Figure 15.1-44a. Event Diagram – Anticipated Transients Without Scram

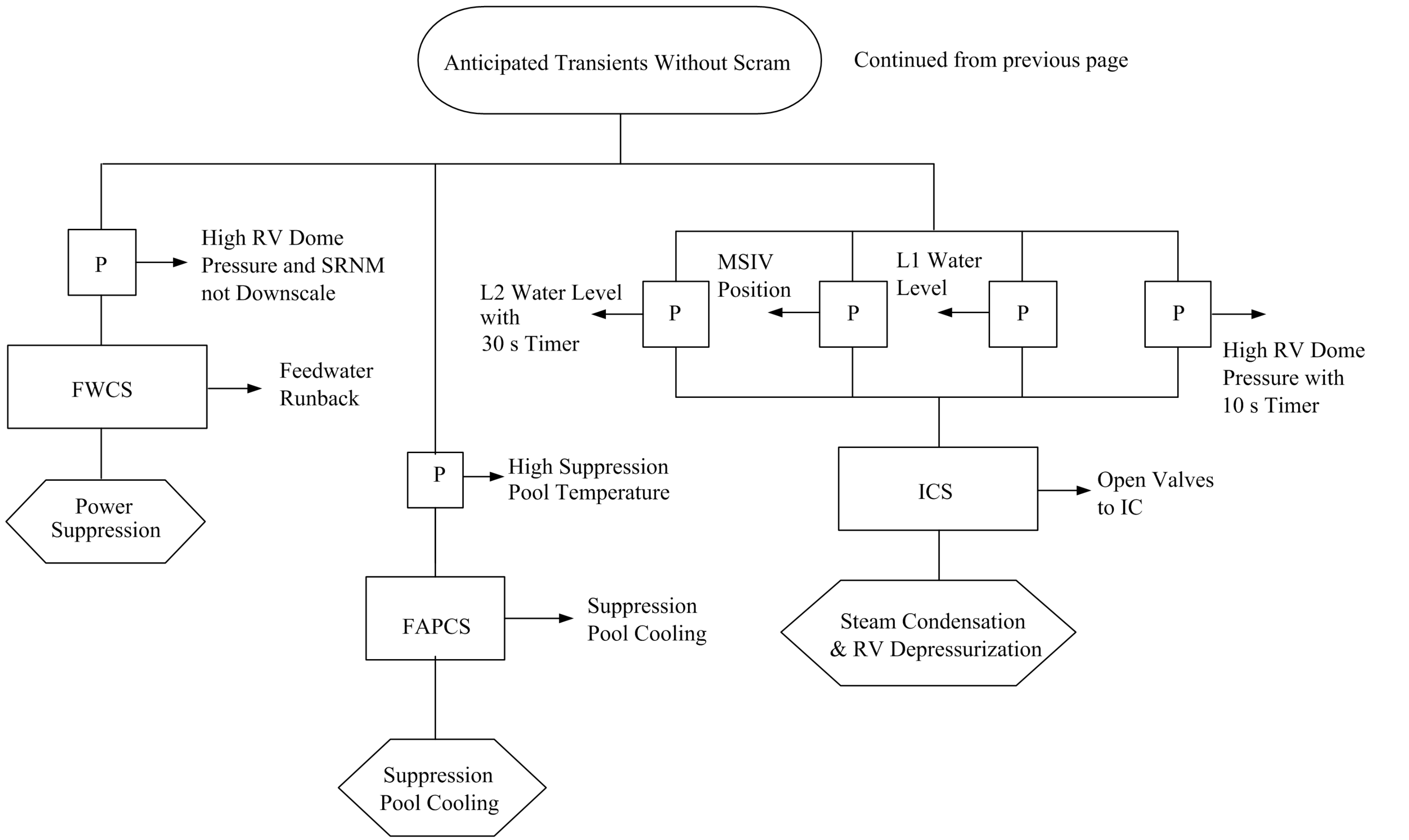


Figure 15.1-44b. Event Diagram – Anticipated Transients Without Scram

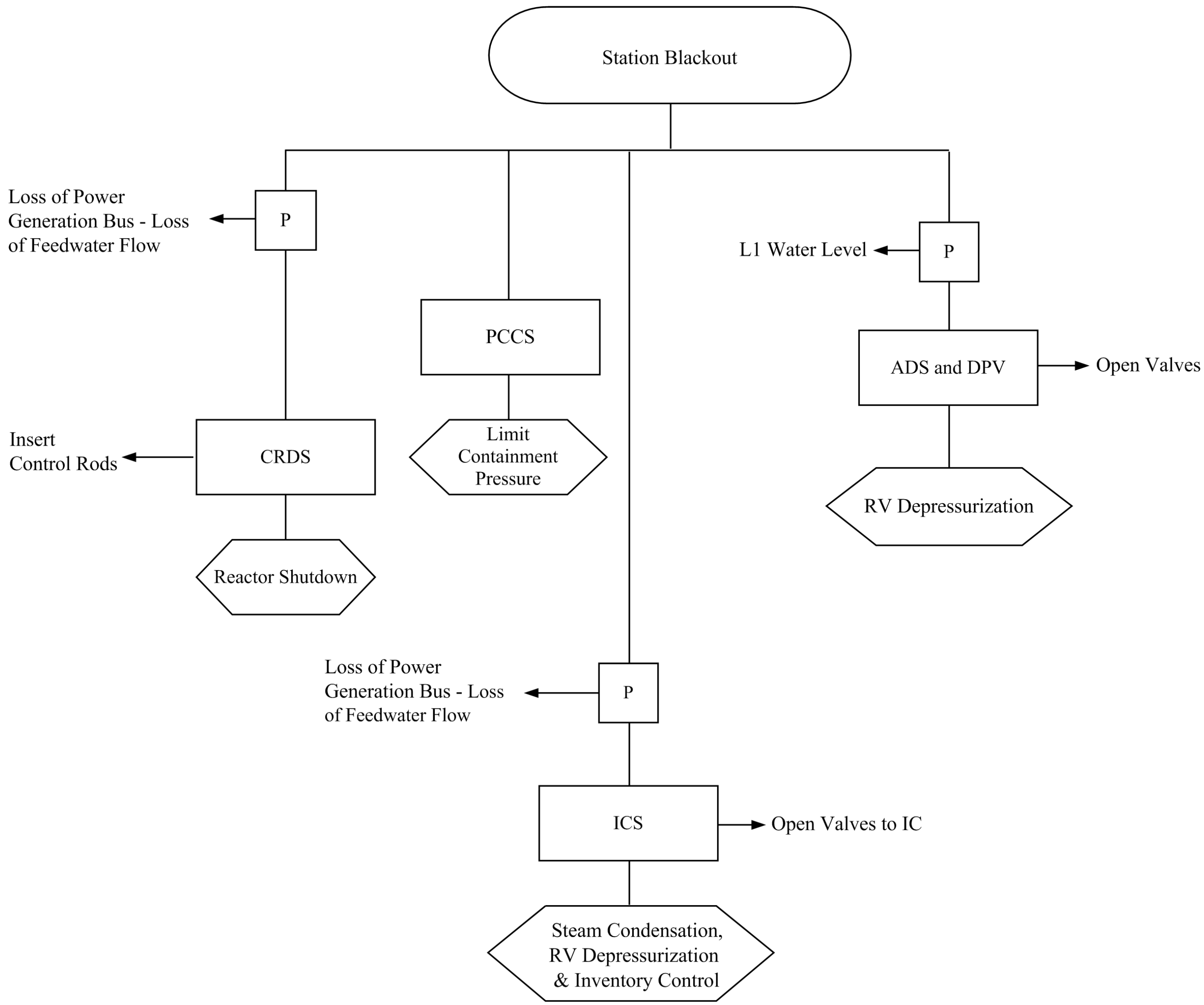


Figure 15.1-45a. Event Diagram – Station Blackout

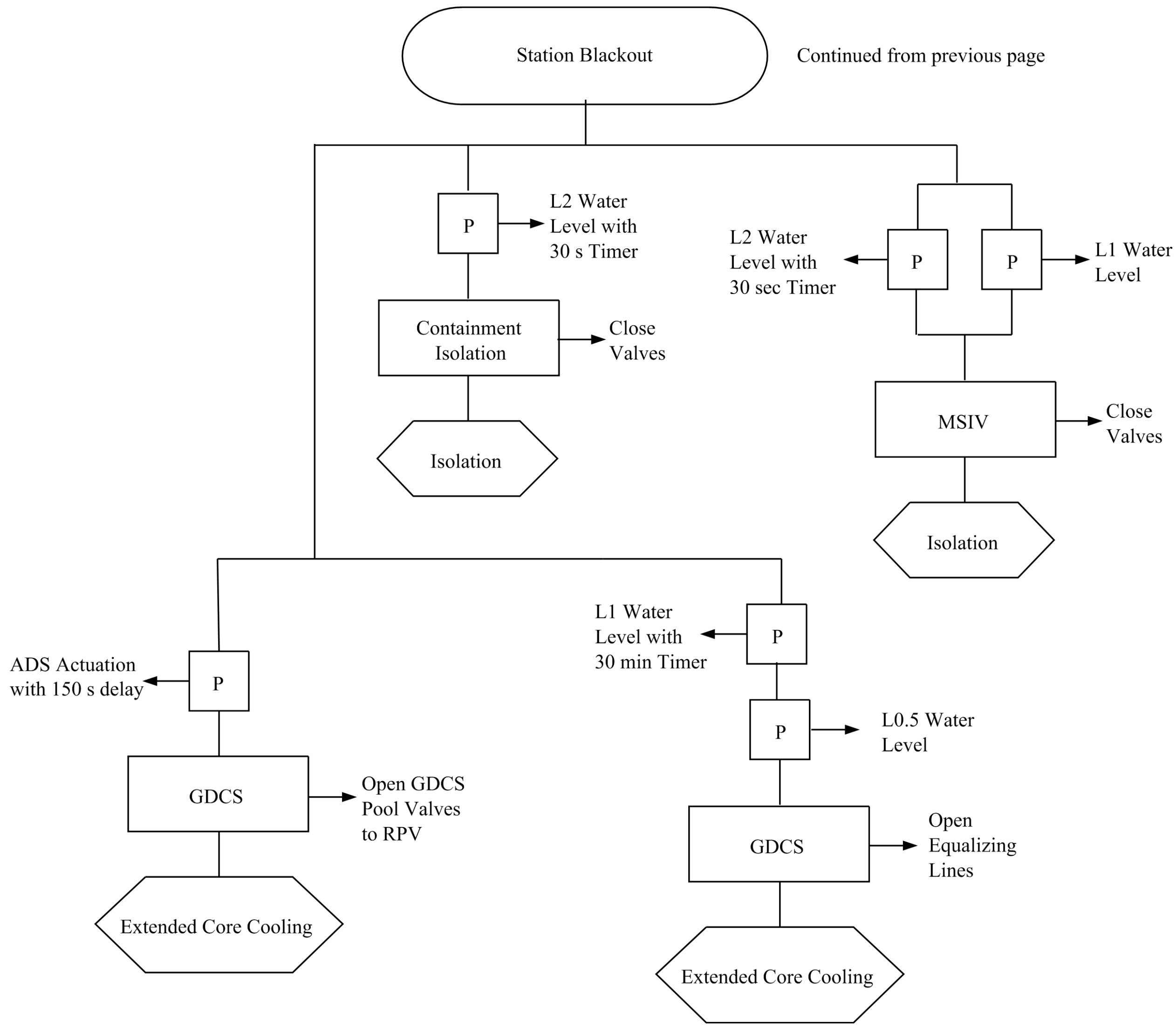


Figure 15.1-45b. Event Diagram – Station Blackout

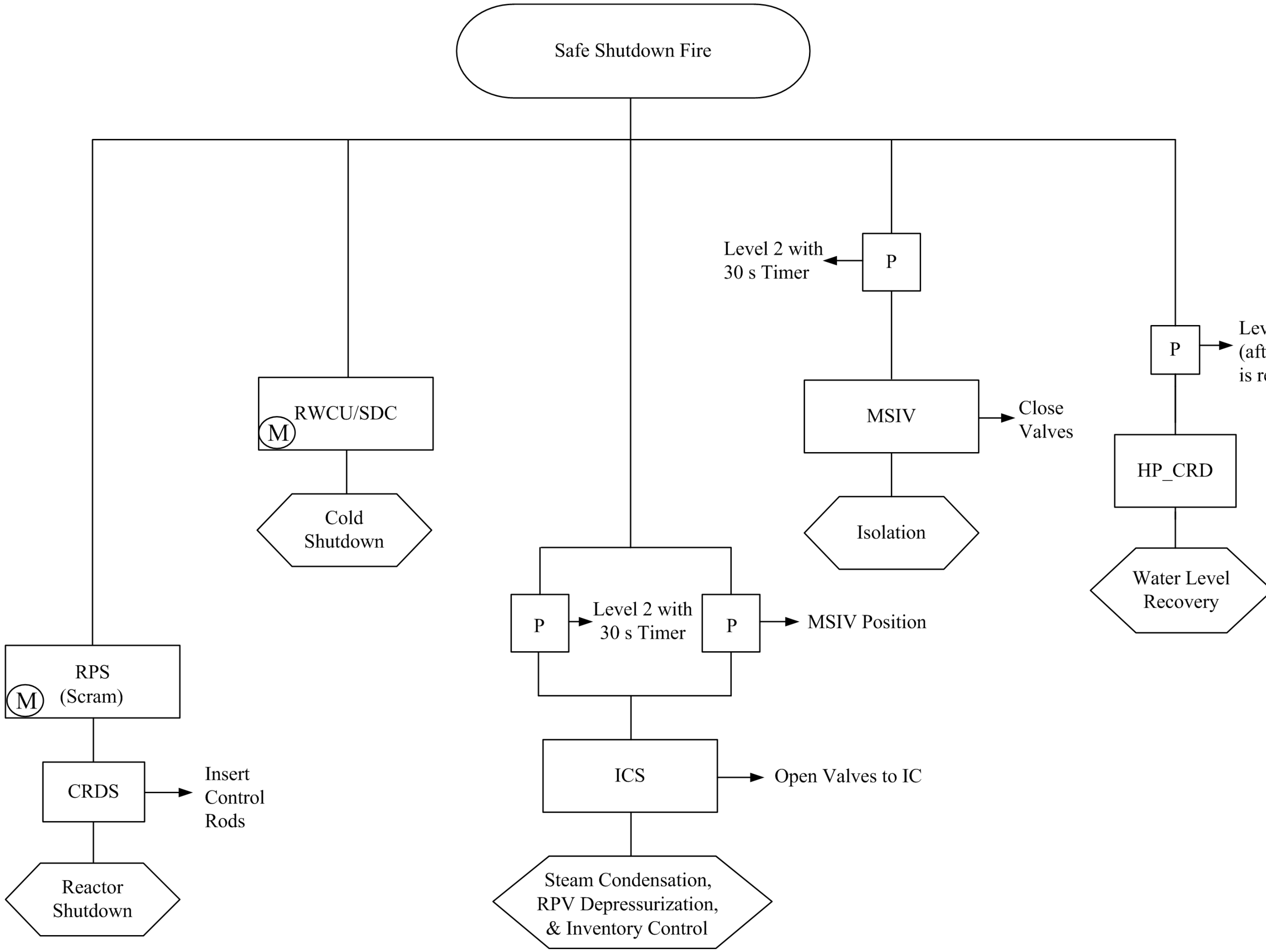
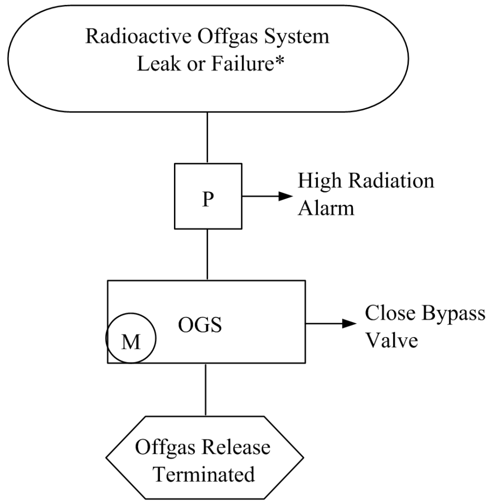


Figure 15.1-46. Event Diagram – Safe Shutdown Fire



* Leak is assumed to result from inadvertent opening of a valve bypassing the charcoal adsorber tanks

Figure 15.1-47. Event Diagram – Waste Gas System Leak or Failure

15.2 ANALYSIS OF ANTICIPATED OPERATIONAL OCCURRENCES

Each of the anticipated operational occurrences (AOOs) addressed in the Section 15.1, "Nuclear Safety Operations Analysis" (NSOA), is evaluated in the following subsections. Appendix 15A provides a determination of event frequency to categorize AOOs as defined in 10 CFR 50 Appendix A. Tables 15.2-1, 15.2-2, and 15.2-3 provide the important input parameters and initial conditions used/assumed in the AOO analyses.

In the analysis of AOOs and Infrequent Events in Section 15.3 nonsafety-related systems or components are considered to be operational in the following situations:

- When assumption of a nonsafety-related system results in a more limiting event;
- When a detectable and nonconsequential random, independent failure must occur in order to disable the system; and
- When nonsafety-related systems or components are used as backup protection (i.e. not the primary success path, included to illustrate the expected plant response to the event).

15.2.0 Assumptions

Assumptions are listed in the event discussions and Table 15.2-1.

15.2.1 Decrease In Core Coolant Temperature

15.2.1.1 Loss Of Feedwater Heating

15.2.1.1.1 Identification of Causes

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

- Steam extraction line to heater is closed; or
- FW is bypassed around heater.

The first case produces a gradual cooling of the FW. In the second case, the FW bypasses the heater and no heating of the FW occurs. In either case, the reactor vessel receives colder FW. The maximum number of FW heaters that can be tripped or bypassed by a single event represents the most severe event for analysis considerations.

The ESBWR is designed such that no single operator error or equipment failure causes a loss of more than 55.6°C (100°F) FW heating.

The loss of FW heating causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. However, the power increase is slow.

The Feedwater Control System (FWCS) includes logic to mitigate the effects of a loss of FW heating capability. The system is constantly monitoring the actual FW temperature and comparing it with a reference temperature. When a loss of FW heating is detected (i.e., when the difference between the actual and reference temperatures exceeds a ΔT setpoint)**,** the FWCS sends an alarm to the operator and sends a signal to the Nonsafety-Related Distributed Control and Information System (N-DCIS) to initiate the Selected Control Rod Run-In (SCRRI) and Select Rod Insert (SRI) functions to automatically reduce the reactor power. This prevents the reactor from violating any thermal limits. These functions are also collectively referred to as SCRRI/SRI.

Control rod insertion is conservatively assumed to start only when the temperature difference setpoint is reached in the FW nozzle. The SCRRI/SRI is able to suppress the neutron power increase and ensure that the minimum critical power ration (MCPR) reduction is small.

The SCRRI/SRI function reduces the core power and limits the change in MCPR after a Loss of Feedwater Heating. The SCRRI/SRI rod pattern depends on the fuel cycle exposure and initiating event. A typical rod pattern is analyzed in this event. The rod pattern analyzed is divided in five control rod groups. Four SRI groups, with scattered insertion times (a separation of 10 seconds between each subgroup) and a SCRRI group with a total insertion time of 110 seconds, activates simultaneously with the first SRI group.

Events may exist where the SCRRI/SRI is not activated because the loss of feedwater temperature is less than 16.67°C (30°F). These events have a ΔCPR/ICPR similar to the event studied here, however none are limiting.

15.2.1.1.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.2-4 lists the sequence of events for Figure 15.2-1

Because no scram occurs during this event, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. Also, the operator should determine the cause of failure prior to returning the system to normal.

Systems Operation

In establishing the expected sequence of events and simulating plant performance, the plant instrumentation and controls, plant protection and reactor protection systems are assumed to function normally. A failure of a single HCU is assumed.

15.2.1.1.3 Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by programming a change in FW enthalpy corresponding to the assumed loss in FW heating, shown in Table 15.2-1.

Results

Because the power increase during this event is controlled by the SCRRI/SRI insertion, the reduction of the MCPR is very small and is turned around when the SCRRI/SRI function takes effect. The results are summarized in Table 15.2-5.

No scram is assumed in this analysis. Nuclear system pressure does not significantly change and consequently, the Reactor Coolant Pressure Boundary (RCPB) is not threatened.

This event is potentially limiting with respect to operating limit minimum critical power ration (OLMCPR). This event is analyzed for each fuel cycle and SCRRI/SRI rod pattern. The SCRRI/SRI requirements are documented in the Core Operating Limits Report (COLR) in accordance with Technical Specifications. The OLMCPR is established for the limiting event and documented in the COLR in accordance with Technical Specifications.

15.2.1.1.4 Barrier Performance

As noted previously, the effects of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel (as stated in the Section 8.3 of Reference 15.2-1 regarding the centerline melt protection discussion with the TRACG methodology), pressure vessel or containment are designed. Therefore, these barriers maintain their integrity and function as designed.

15.2.1.1.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2 Increase In Reactor Pressure

15.2.2.1 Closure of One Turbine Control Valve

15.2.2.1.1 Identification of Causes

The ESBWR Steam Bypass and Pressure Control (SB&PC) system uses a triplicated digital control system. This system is similar to the one used in the ABWR design. The SB&PC system controls the TCVs and turbine bypass valves to maintain reactor pressure. As presented in Subsection 15.2.4.2, no credible single failure in the control system results in a minimum demand to all Turbine Control Valves (TCVs) and bypass valves. A voter or actuator failure may result in an inadvertent closure of one turbine control valve or one turbine bypass valve if it is open at the time of failure. In this case, the SB&PC system senses the pressure change and commands the remaining control valves or bypass valves, if needed, to open, and thereby automatically mitigates the transient to maintain reactor power and pressure.

Because turbine bypass valves are normally closed during normal full power operation, it is assumed for purposes of this transient analysis that a single failure causes a single turbine control valve to fail closed. Should this event occur at full power, the remaining control valves opening may not be sufficient to maintain the reactor pressure, depending on the turbine design. Neutron flux would increase, in this case, due to void collapse resulting from the pressure increase. A reactor scram would be initiated if the high flux or high pressure scram setpoint is reached.

15.2.2.1.2 Sequence of Events and System Operation

Sequence of Events

Postulating an actuator failure of the SB&PC system causes one TCV to close. The pressure increases because the reactor is still generating the initial steam flow. The SB&PC system opens the remaining control valves and some bypass valves. This sequence of events is listed in Table 15.2-6 for Figure 15.2-2, for a fast closure with partial arc, and in Table 15.2-7 for Figure 15.2-3, for a slow closure with partial arc.

Systems Operation

Normal plant instrumentation and control are assumed to function. After a closure of one turbine control valve, the steam flow rate that can be transmitted through the remaining three TCVs depends upon the turbine configuration. For plants with full-arc turbine admission, the steam flow through the remaining three TCVs is at least 95% of rated steam flow. This capacity drops to about 85% of rated steam flow for plants with partial-arc turbine admission. Therefore, this transient is less severe for plants with full-arc turbine admission. In this analysis, the case with partial-arc turbine admission is analyzed to cover all plants.

Table 15.2-1 provides the following data for the TCV:

- Design full stroke closure time, from fully open to fully closed;
- Bounding closure time assumed in the fast closure analysis;
- Closure time assumed in the slow closure analysis; and
- Percent of rated steam flow that can pass through three TCVs.

15.2.2.1.3 Core and System Performance

A simulated fast closure of one TCV is presented in Figure 15.2-2. Neutron flux increases, because of the void reduction caused by the pressure increase. However, the sensed neutron flux does not reach the high neutron flux scram setpoint. When the sensed reactor pressure increases, the pressure regulator opens the bypass valves, keeping the reactor pressure at a constant level. The calculated peak thermal flux is provided in Table 15.2-5. The number of rods in boiling transition for this transient remains within the acceptance criterion for AOOs. Therefore, the design basis is satisfied.

A slow closure of one TCV is also analyzed as shown in Figure 15.2-3. As in the fast closure case, the neutron flux increase does not reach the high neutron flux scram setpoint. Also, a reactor scram on high reactor pressure may also be generated. The results of this event are very similar to the fast closure event discussed above. During the transient, the number of rods in boiling transition remains within the acceptance criterion for AOOs. Therefore, the design basis is satisfied.

This event is potentially limiting with respect to OLMCPR. This event is analyzed for each fuel cycle. The OLMCPR is established for the limiting event and documented in the COLR in accordance with Technical Specifications.

15.2.2.1.4 Barrier Performance

Peak pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak vessel bottom pressure is below the upset pressure limit.

15.2.2.1.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.2 Generator Load Rejection With Turbine Bypass

15.2.2.2.1 Identification of Causes

Fast closure of the 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 over-speed of the turbine generator (TG) rotor. Closure of the TCVs causes a sudden reduction in steam flow. To prevent an increase in system pressure, sufficient bypass capacity is provided to pass steam flow diverted from the turbine.

After sensing a significant loss of electrical load on the generator, the TCVs are commanded to close rapidly. At the same time, the turbine bypass valves are signaled to open in the "fast" opening mode by the SB&PC system, which uses a triplicated digital controller. As presented in Subsection 15.2.4.3, no single failure can cause all turbine bypass valves to fail to open on demand.

Assuming no single failure, the plant has the full steam bypass capability available and the Reactor Protection System (RPS) verifies that the bypass valves are open. The fast closure of the TCVs produces a pressure increase that is negligible, because all the steam flow is bypassed through the turbine bypass valves. The reactor power decreases when the SCRRI/SRI actuates.

The SCRRI/SRI function reduces the core power and limits the change in MCPR after a generator load rejection with turbine bypass. The SCRRI/SRI rod pattern depends on the fuel cycle exposure and initiation event. A typical rod pattern is analyzed in this event. The rod pattern analyzed is divided in five control rod groups. Four SRI groups, with scattered insertion times (a separation of 10 seconds between each subgroup) and a SCRRI group with a total insertion time of 110 seconds, activate simultaneously with the first SRI group.

15.2.2.2.2 Sequence of Events and System Operation

Sequence of Events

A loss of generator electrical load from high power conditions produces the sequence of events listed in Table 15.2-8.

Identification of Operator Actions

Relatively small changes in plant conditions are experienced. The operator should, after checking that the SCRRI/SRI system has been activated, check reactor water level, reactor pressure and MSIV status. If conditions are normal, no further operator actions are needed.

System Operation

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 unless stated otherwise.

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

15.2.2.2.3 Core and System Performance

Input Parameters and Initial Conditions

The turbine electro-hydraulic control system (EHC) detects load rejection before a measurable turbine speed change takes place.

The closure characteristics of the TCVs are assumed such that the valves operate in the full arc (FA) mode. For this event, Table 15.2-1 provides the worst case full stroke closure time (from fully open to fully closed) for the TCVs, which is assumed in the analysis.

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

Results

Figure 15.2-4 shows the results of the generator trip from the 100% rated power conditions and with the turbine bypass system operating normally. Although the peak neutron flux and average simulated thermal heat flux increase**,** the number of rods expected in boiling transition remains within the acceptance criterion for AOOs.

This event is potentially limiting with respect to OLMCPR. This event is analyzed for each fuel cycle and SCRRI/SRI rod pattern. The SCRRI/SRI requirements are documented in the COLR in accordance with Technical Specifications. The OLMCPR is established for the limiting event and documented in the COLR in accordance with Technical Specifications.

15.2.2.2.4 Barrier Performance

Peak pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak vessel bottom pressure remains below the upset pressure limit.

15.2.2.2.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.3 Generator Load Rejection With a Single Failure in the Turbine Bypass System

15.2.2.3.1 Identification of Causes

Fast closure of the 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 TG rotor. Closure of the TCVs causes a sudden reduction in steam flow, which results in an increase in system pressure and reactor shutdown if the available turbine steam bypass capacity is insufficient.

After sensing a significant loss of electrical load on the generator, the TCVs are commanded to close rapidly. At the same time, the turbine bypass valves are signaled to open in the "fast" opening mode by the SB&PC system, which uses a triplicated digital controller. As presented in Subsection 15.2.4.3, no single failure can cause all turbine bypass valves to fail to open on demand. It is assumed that half of the turbine bypass valves fail to open on demand in this analysis.

15.2.2.3.2 Sequence of Events and System Operation

Sequence of Events

A loss of generator electrical load with a single failure in the turbine bypass system from high power conditions produces the sequence of events listed in Table 15.2-9.

Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Verify proper bypass valve performance;
- Observe that the FW/level controls have maintained the reactor water level at a satisfactory value;
- Observe that the pressure regulator is controlling reactor pressure at the desired value; and
- Observe reactor peak power and pressure.

System Operation

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 unless stated otherwise.

Conservatively, and to cover all possible failures, it is assumed that the system with a single failure only opens to 50% of the total steam bypass capacity.

All plant control systems maintain normal operation unless specifically designated.

15.2.2.3.3 Core and System Performance

Input Parameters and Initial Conditions

The turbine EHC detects load rejection before a measurable turbine speed change takes place.

The closure characteristics of the TCVs are assumed such that the valves operate in the full arc (FA) mode. For this event, Table 15.2-1 provides the design full stroke closure time (from fully open to fully closed) for the TCVs and the worst-case closure time is assumed in the analysis.

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

The pressurization and/or the reactor scram may compress the water level to the low level trip setpoint (Level 2) and initiate the CRD high pressure makeup function, and if the low level signal remains for 30 seconds, MSIV closure, and isolation condenser (IC) operation. Should this occur, it would follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred.

Results

Figure 15.2-5 shows the results of the generator trip from the 100% rated power conditions assuming only 50% of the total turbine bypass system capacity. Although the peak neutron flux and average simulated thermal heat flux increase**,** the number of rods in boiling transition remains within the acceptance criterion for AOOs in combination with an additional single active component failure or operator error.

This event is potentially limiting with respect to OLMCPR. This event is analyzed for each fuel cycle. The OLMCPR is established for the limiting event and documented in the COLR in accordance with Technical Specifications. .

15.2.2.3.4 Barrier Performance

Peak pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak vessel bottom pressure remains below the upset pressure limit.

15.2.2.3.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.4 Turbine Trip With Turbine Bypass

15.2.2.4.1 Identification of Causes

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

After the main turbine is tripped, turbine bypass valves are opened in their fast opening mode by the SB&PC system. The reactor power decreases when the SCRRI/SRI actuates.

The SCRRI/SRI function reduces the core power and limits the change in MCPR after a turbine trip with turbine bypass. A typical rod pattern is analyzed in this event. The SCRRI/SRI rod pattern used in the turbine trip with turbine bypass is the same as the one used in the generator load rejection with turbine bypass discussed in Subsection 15.2.2.2.1.

15.2.2.4.2 Sequence of Events and Systems Operation

Sequence of Events

Turbine trip at high power produces the sequence of events listed in Table 15.2-10.

Identification of Operator Actions

Relatively small changes in plant conditions are experienced. The operator should, after checking that the SCRRI/SRI system has been activated, check reactor water level, reactor pressure and MSIV status. If conditions are normal, no further operator actions are needed.

Systems Operation

All plant control systems maintain normal operation unless specifically designated to the contrary. Credit is taken for successful operation of the RPS.

15.2.2.4.3 Core and System Performance

Input Parameters and Initial Conditions

Table 15.2-1 provides the Turbine Stop Valve (TSV) full stroke closure time design range, and the worst case (bounding) TSV closure time assumed in the analysis.

Results

A turbine trip with the bypass system operating normally is simulated at rated power conditions as shown in Figure 15.2-6. Table 15.2-5 summarizes the analysis results. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the pressure increase is limited by the initiation of the steam bypass operation. Peak simulated thermal heat flux does not significantly exceed (< 1%) of its initial value. After the control system verifies that the bypass capacity is adequate, the system activates the SCRRI/SRI to reduce the power to 60% and later proceed to a possible restart or a controlled shut-down. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs.

This event is potentially limiting with respect to OLMCPR. This event is analyzed for each fuel cycle and SCRRI/SRI rod pattern. The SCRRI/SRI requirements are documented in the COLR in accordance with Technical Specifications. The OLMCPR is established for the limiting event and documented in the COLR in accordance with Technical Specifications.

15.2.2.4.4 Barrier Performance

Peak pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak pressure at the vessel bottom remains below the upset pressure limit.

15.2.2.4.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.5 Turbine Trip With a Single Failure in the Turbine Bypass System

15.2.2.5.1 Identification of Causes

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

After the main turbine is tripped, turbine bypass valves are opened in their fast opening mode by the SB&PC system.

15.2.2.5.2 Sequence of Events and Systems Operation

Sequence of Events

Turbine trip with a single failure in the turbine bypass system at high power produces the sequence of events listed in Table 15.2-11.

Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Verify that the generator breaker is automatically open to allow electrical buses originally supplied by the generator to be supplied by the incoming power;
- Monitor reactor water level and pressure;
- Check turbine for proper operation of all auxiliaries during coastdown;
- Manually initiate ICs, if necessary, to control reactor pressure;
- Depending on conditions, maintain pressure for restart purposes, or initiate normal operating procedures for cooldown;
- Put the mode switch in the startup position before the reactor pressure decays to below 6 MPa (870 psig);

Systems Operation

All plant control systems maintain normal operation unless specifically designated to the contrary. Credit is taken for successful operation of the RPS.

Conservatively and to cover all possible failures it is assumed that the system with a single failure only opens to 50% of the total steam bypass capacity.

15.2.2.5.3 Core and System Performance

Input Parameters and Initial Conditions

Table 15.2-1 provides the TSV full stroke closure time design range, and the worst case (bounding) TSV closure time assumed in the analysis. A reactor scram occurs due to fast TSV closure, with inadequate availability of turbine bypass.

Results

A turbine trip, assuming only 50% of the total turbine steam bypass capacity available, is simulated at rated power conditions as shown in Figure 15.2-7. Table 15.2-5 summarizes the analysis results. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited by the partial actuation of the steam bypass system and the initiation of reactor scram. The peak simulated thermal heat flux does not significantly increase (< 10%) above its initial value. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs in combination with an additional single active component failure or operator error. This event is reanalyzed for each specific initial core configuration.

15.2.2.5.4 Barrier Performance

Peak pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak pressure at the vessel bottom remains below the upset pressure limit.

15.2.2.5.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.6 Closure of One Main Steamline Isolation Valve

15.2.2.6.1 Identification of Causes

Protection system logic permits the test closure of one MSIV without initiating scram from the position switches. An inadvertent closure of one MSIV may cause an immediate closure of all other MSIVs, depending on reactor conditions. Closure of all MSIVs is discussed in Subsection 15.2.2.7.

15.2.2.6.2 Sequence of Events and Systems Operation

When a single MSIV is closed in conformance with normal testing procedures, no reactor scram occurs and the reactor settles into a new steady state operating condition. Closure of a single MSIV at power levels above those of the normal testing procedure may cause closure of all other MSIVs.

Table 15.2-12 lists the sequence of events for Figure 15.2-8.

15.2.2.6.3 Core and System Performance

The neutron flux increases slightly while the simulated thermal heat flux shows no increase. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs. The effects of closure of a single MSIV are considerably milder than the effects of closure of all MSIVs. Therefore, this event does not need to be reanalyzed for any specific core configuration.

Inadvertent closure of one MSIV while the reactor is shut down produces no significant transient. Closures during plant heatup are less severe than closure from maximum power cases.

15.2.2.6.4 Barrier Performance

Peak pressure at the vessel bottom remains below the pressure limits of the reactor coolant pressure boundary. Peak pressure in the main steamline remains below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool.

15.2.2.6.5 Radiological Consequence

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.7 Closure of All Main Steamline Isolation Valves

15.2.2.7.1 Identification of Causes

Various steamline and nuclear system malfunctions, or operator actions, can initiate MSIV closure. Examples are low steamline pressure, high steamline flow, low water level or manual action.

To define this event as an initiating event and not the byproduct of another AOO, only the following are considered:

- Manual action (purposely or inadvertent);
- Spurious signals such as low pressure, low reactor water level, 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 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 if the valves in two or more main steamlines are less than that shown in Table 15.2-1 (except for interlocks which permit proper plant startup). Protection system logic, however, switches to two out of three of the remaining MSIVs which permits the test closure of one valve without initiating scram from the position switches.

15.2.2.7.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.2-13 lists the sequence of events for Figure 15.2-9.

The following is the sequence of operator actions expected during the course of the event, assuming no restart of the reactor. The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Check that ICs have initiated (i.e., drain valves open);
- Monitor reactor water level and pressure;
- Initiate Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System operation in the shutdown cooling mode appropriate to hot shutdown;
- Determine the cause of valve closure before resetting the MSIV isolation;
- Observe turbine coastdown and break vacuum before the loss of sealing steam (check turbine auxiliaries for proper operation); and
- Check that conditions are satisfactory prior to opening and resetting MSIVs.

Systems Operation

MSIV closure initiates a reactor scram trip via position signals to the RPS. The same signal also initiates the operation of ICs, which prevent the lifting of SRVs.

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

15.2.2.7.3 Core and System Performance

Input Parameters and Initial Conditions

The MSIV design closure time range and the worst case (bounding) closure time assumed in this analysis are provided in Table 15.2-1.

Position switches on the valves initiate a reactor scram, as addressed in Table 15.2-1. Closure of these valves causes the dome pressure to increase.

Results

Figure 15.2-9 shows the changes in important nuclear system variations for the simultaneous isolation of all main steamlines while the reactor is operating at rated power. The neutron flux increases slightly while the simulated thermal heat flux shows no increase. The FW injection and the IC operation terminate the pressure increase. The anticipatory scram prevents any change in the thermal margins. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs. Therefore, this event does not have to be reanalyzed for any specific core configurations.

Inadvertent closure of all of the MSIVs while the reactor is shut down produces no significant transient. Closures during plant heatup are less severe than the maximum power cases (maximum stored and decay heat) presented.

15.2.2.7.4 Barrier Performance

Peak pressure at the vessel bottom remains below the upset event pressure limit for the reactor coolant pressure boundary (RCPB). Peak pressure in the main steamline remains below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool.

15.2.2.7.5 Radiological Consequence

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.8 Loss of Condenser Vacuum

15.2.2.8.1 Identification of Causes

Various system malfunctions that can cause a loss of condenser vacuum due to some single equipment failure are designated in Table 15.2-14.

15.2.2.8.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.2-15 lists the sequence of events for Figure 15.2-10.

The Loss of Condenser Vacuum initially does not affect the vessel, when the turbine trip setpoint is reached it has a simultaneous scram with a bypass valve opening. Six seconds later (see Table 15.2-16), the low vacuum setpoint produces closure of the bypass valve: with a small delay the MSIV also closes.

Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Monitor reactor water level and pressure;
- Check turbine for proper operation of all auxiliaries during coastdown;
- Use ICs to control reactor pressure;
- Depending on conditions, maintain pressure for restart purposes, or initiate normal operating procedures for cooldown;
- Put the mode switch in the STARTUP position before the reactor pressure decays below 6 MPa (870 psig).

Systems Operation

In establishing the expected sequence of events and simulating the plant performance, the plant instrumentation and controls, plant protection and reactor protection systems are assumed to normally function.

Tripping functions incurred by sensing main turbine condenser vacuum are presented in Table 15.2-16.

15.2.2.8.3 Core and System Performance

Input Parameters and Initial Conditions

TSV full stroke closure time is as shown in Table 15.2-1.

A reactor scram is initiated on low condenser vacuum at the same time that the turbine trip signal is generated.

The analysis presented here is a hypothetical case with a conservative vacuum decay rate (see Table 15.2-1). Thus, the bypass system is available for several seconds because the bypass is signaled to close at a vacuum level that is less than the stop valve closure (see Table 15.2-16).

Results

As shown in Table 15.2-15, under the analysis vacuum decay condition, the turbine bypass valves and MSIV closure would follow main turbine trip. This AOO is similar to a normal turbine trip with bypass. The effect of MSIV closure tends to be minimal, because the reactor scram on low condenser vacuum precedes the isolation by several seconds. Figure 15.2-10 shows the transient expected for this event. It is assumed that the plant is initially operating at rated power conditions. Peak neutron flux is shown in Table 15.2-5, and the average simulated thermal heat flux peaks at < 110% of rated. The number of rods in boiling transition during this event remains within the acceptance criterion for AOOs. Therefore, this event does not have to be reanalyzed for any specific core configuration.

15.2.2.8.4 Barrier Performance

Peak nuclear system pressure remains below the ASME code upset limit. Peak pressure in the main steamline remains below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. A comparison of these values to those for turbine trip at high power shows the similarities between these two transients. The prime difference is the subsequent main steamline isolation.

15.2.2.8.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.2.9 Loss of Shutdown Cooling Function of RWCU/SDC

Although the RWCU/SDC system is nonsafety-related, it can perform high and low pressure core cooling. The RWCU/SDC system has two trains, each containing the necessary piping, pumps, valves, heat exchangers, instrumentation and electrical power for operation. Each train also has its own cooling water supply, connection to standby AC power, pump, and equipment room cooling system. For the shutdown cooling function, each train has its own suction line from the RPV and return line to the FW line. Thus, each of the RWCU/SDC trains is completely independent. The RWCU/SDC system, together with the main condenser, reduces the primary system temperature after plant shutdown.

If the single active failure criterion is applied to the analysis of the RWCU/SDC system, resulting scenarios have one RWCU/SDC trains inoperable. However, in these same scenarios, the remaining operable RWCU/SDC train allows for achieving cold shutdown within 36 hours after reactor shutdown.

Failure of offsite power is another case that could affect the shutdown cooling function. The plant has two independent offsite power supplies. If both offsite power supplies are lost, each RWCU/SDC train has its own standby AC power source (e.g., diesel generator) that permits operating that train at its rated capacity. Application of the single active failure criterion would still leave an RWCU/SDC train operational.

The RWCU/SDC system description and performance evaluation in Subsection 5.4.8 describes the models, assumptions and results for shutdown cooling with two RWCU/SDC trains operational.

15.2.3 Reactivity and Power Distribution Anomalies

Based on the probability for limiting reactivity and power distribution anomalies, there are no reactivity and power distribution anomaly AOOs identified for the ESBWR.

15.2.4 Increase in Reactor Coolant Inventory

15.2.4.1 Inadvertent Isolation Condenser Initiation

15.2.4.1.1 Identification of Causes

Manual startup of the four individual Isolation Condenser (IC) systems is postulated for this analysis (i.e., operator error).

15.2.4.1.2 Sequence of Events and System Operation

Sequence of Events

Table 15.2-17 lists the sequence of events for Inadvertent Isolation Condenser Initiation.

Identification of Operator Actions

Relatively small changes in plant conditions are experienced. The operator should, after hearing the alarm that the IC system has commenced operation, check reactor water level, reactor pressure and MSIV status. If conditions are normal, the operator should shut down the system.

System Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls. Specifically, the pressure regulation and the vessel level control that respond directly to this event.

Required operation of engineered safeguards other than what is described is not expected for this event.

15.2.4.1.3 Core and System Performance

Input Parameter and Initial Conditions

The assumed IC system water temperature and enthalpy, startup time and flow rate are provided in Table 15.2-1.

Inadvertent startup of all loops of the IC system was chosen as the limiting case for analysis because it provides the greatest auxiliary source of cold water into the vessel.

Results

Figure 15.2-11 shows the simulated transient event. It begins with the introduction of cold water into the downcomer region. Full IC loop flow is established. No delays are considered because these are not relevant to the analysis.

Addition of cooler water to the downcomer causes a reduction in inlet enthalpy, which results in a power increase. The flux level settles out slightly above its operating level. The variations in the pressure and thermal conditions are relatively small, and no significant effect is experienced. The number of rods in boiling transition remains within the acceptance criterion for AOOs, and the fuel thermal margins are maintained.

This event is potentially limiting with respect to OLMCPR. This event is analyzed for each fuel cycle. The OLMCPR is established for the limiting event and documented in the COLR in accordance with Technical Specifications.

Consideration of Uncertainties

Important analytical factors, including IC loop condensate water temperature, are assumed to be at the worst conditions so that any deviations in the actual plant parameters would produce a less severe transient.

15.2.4.1.4 Barrier Performance

Inadvertent Startup of the IC causes only a slight pressure decrease from the initial conditions; therefore, no further RCPB pressure response evaluation is required.

15.2.4.1.5 Radiological Consequences

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

15.2.4.2 Runout of One Feedwater Pump

15.2.4.2.1 Identification of Causes

The FW pumps (three normally operating) are driven by an adjustable-speed, induction motor controlled by an adjustable speed drive. This event is postulated on the basis of a single failure of a control device, specifically one that can directly cause an increase coolant inventory by increasing the speed of a single FW pump. The term “runout” is used in this section to describe this failure.

The ESBWR FWCS uses a triplicated digital control system, instead of a single-channel analog system that was originally provided in current BWR designs (BWR/2-6). The digital systems consist of a triplicated fault-tolerant digital controller, the operator control stations and displays. The digital controller contains three parallel processing channels, each containing the microprocessor-based hardware and associated software necessary to perform all the control calculations. The operator interface provides information regarding system status and the required control functions.

Redundant transmitters are provided for key process inputs, and input voting and validation are provided such that faults can be identified and isolated. Each system input is triplicated internally and sent to the three processing channels (Figure 15.2-12). The channels produce the same output during normal operation. Interprocessor communication provides self-diagnostic capability. A two-out-of-three voter compares the processor outputs to generate a validated output to the control actuator. A separate voter is provided for each actuator. A “ringback” feature feeds back the final voter output to the processors. A voter failure is thereby detected and alarmed. In some cases, a protection circuit locks the actuator into its existing position promptly after the failure is detected.

Table 15.2-18 lists the potential failure modes of a triplicated digital control system and outlines the effects of each failure. Because of the triplicated architecture, it is possible to take one channel out of service for maintenance or repair while the system is online. Modes 2 and 5 of Table 15.2-18 address a failure of a component while an associated redundant component is out of service. This type of failure could potentially cause a system failure. However, the probability of a component failure during servicing of a counterpart component is considered to be so low that these failure modes are not considered AOOs, but are considered infrequent events (see Appendix 15A.3.5).

Adverse effect minimization is mentioned in the effects of Mode 2. This feature stems from the additional intelligence of the system provided by the microprocessor. When possible, the system is programmed to take action in the event of some failure to reduce the severity of the event. For example, if the total steam flow or total FW flow signals fail, the FWCS detects this by the input reasonability checks and automatically switch to one-element mode (i.e., control by level feedback only). The level control would essentially be unaffected by this failure.

The only credible single failures that would lead to some adverse effect on the plant are Modes 6 (failure of the output voter) and 7 (control actuator failure). Either of these failures would lead to a loss of control of only one actuator (i.e., only one FW pump with increasing flow). A voter failure is detected by the ringback feature. The FWCS initiates a lockup of the actuator upon detection of the failure. The probabilities of failure of the variety of control actuators are very low based on operating experience. The worst single failure in the FWCS causes a runout of one FW pump to its maximum capacity. In the event of one pump run-out, the FWCS would then reduce the demand to the remaining pumps, thereby automatically compensating for the excessive flow from the failed pump. However, the demand to the remaining FW pump decreases to offset the increased flow of the failed pump. The effect on total flow to the vessel is not significant. The worst additional single failure would cause all FW pumps to run out to their maximum capacity. However, the probability of this occurrence is extremely low.

15.2.4.2.2 Sequence of Events and Systems Operation

Sequence of Events

With momentary increase in FW flow, the water level rises and then settles back to its normal level. Table 15.2-19 lists the sequencing of events for Figure 15.2-13.

Identification of Operator Actions

Because no scram occurs for runout of one FW pump, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. Also, the operator should determine the cause of failure prior to returning the system to normal.

Systems Operation

Runout of a single FW pump requires no protection system or Engineered Safety Feature (ESF) system operation. This analysis assumes normal functioning of plant instrumentation and controls.

15.2.4.2.3 Core and System Performance

Input Parameters and Initial Conditions

The total FW flow for all pumps runout is provided in Table 15.2-1.

Results

The simulated runout of one FW pump event is presented in Figure 15.2-13. When the increase of FW flow is sensed, the FW controller starts to command the remaining FW pumps to reduce its flow immediately. The vessel water level increases slightly [about 14 cm (6 inch)] and then settles back to its normal level. Vessel pressure increases insignificantly, and the number of rods in boiling transition remains within the acceptance criterion for AOOs.

15.2.4.2.4 Barrier Performance

As previously noted, the effect of this event does not result in any temperature or pressure transient in excess of the design criteria for the fuel, pressure vessel or containment. Therefore, these barriers maintain their integrity and function as designed.

15.2.4.2.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.5 Decrease in Reactor Coolant Inventory

15.2.5.1 Opening of One Turbine Control or Bypass Valve

15.2.5.1.1 Identification of Causes

The ESBWR SB&PC system uses a triplicated digital control system instead of an analog system as was originally provided in current BWR designs (BWR/2-6). The SB&PC system controls TCVs and turbine bypass valves to maintain reactor pressure. As presented in Subsection 15.2.4.2, no credible single failure in the control system results in a maximum demand to all actuators for all TCVs and bypass valves. A voter or actuator failure may result in an inadvertent opening of one TCV or one turbine bypass valve.

15.2.5.1.2 Sequence of Events and Systems Operation

The SB&PC system senses the pressure change and commands the remaining control valves to close, and thereby automatically mitigate the transient and maintain reactor power and pressure.

Table 15.2-20 lists the sequence of events for Figure 15.2-14.

15.2.5.1.3 Core and System Performance

Reactor power and pressure is maintained. Reactor scram does not occur.

15.2.5.1.4 Barrier Performance

The effects of this event do not result in any temperature or pressure transient in excess of the design criteria for fuel, pressure vessel or containment. The peak pressure in the bottom of the vessel remains below the ASME code upset limit. Peak steam line pressure near the SRVs remains below the setpoint of the SRVs.

15.2.5.1.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.5.2 Loss of Non-Emergency AC Power to Station Auxiliaries

This event bounds the Loss of Unit Auxiliary Transformer and Loss of Grid Connection events.

15.2.5.2.1 Identification of Causes

Causes for interruption or loss of power from the unit auxiliary transformer can arise from transformer (main and unit auxiliary) malfunction and isolated phase bus failures. Loss of grid connection can result from major shifts in electrical loads, loss of loads, lightning, storms, wind, etc., which contribute to electrical grid instabilities. These instabilities could cause equipment damage if unchecked. Protective relay schemes automatically disconnect electrical sources and loads to mitigate damage and regain electrical grid stability. The plant is designed to affect bus transfers and operate isolated from the electrical grid without scram. However in this analysis, it is assumed that concurrent with a load rejection, there is a simultaneous loss of power on the four power generation busses, causing the feedwater and circulating pumps to be lost. The bypass valves remain initially available. The loss of the power generation busses produces a scram signal. The loss of the circulating water pumps results in a loss of condenser vacuum over a period of time. As condenser vacuum drops the turbine trips, bypass valves close and the MSIVs close.

15.2.5.2.2 Sequence of Events and Systems Operation

Sequence of Events

For the Loss of Unit Auxiliary Power Transformer, Table 15.2-21 lists the sequence of events for Figure 15.2-15.

Identification of Operator Actions

The operator should maintain the reactor water level by use of the IC system and Control Rod Drive system and control reactor pressure using the Isolation Condenser System (ICS) and RWCU/SDC system. Verify that the turbine and generator DC oil pumps are operating satisfactorily to prevent turbine bearing damage. Also verify proper switching and loading of the standby diesel generators.

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

- Verify that all rods are inserted;
- Follow the scram procedure;
- Check that diesel generators start and carry their assigned loads;
- Monitor reactor water level and pressure; verify that Control Rod Drive flow is controlling water level;
- Use IC system to control pressure;
- Check turbine for proper operation of all auxiliaries during coastdown;
- Put the mode switch in the STARTUP position before the reactor pressure decays below 6 MPa (870 psig);
- Secure the IC when both reactor pressure and level are under control;

Systems Operation

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 provide the simulation sequence shown in Table 15.2-21.

15.2.5.2.3 Core and System Performance

Figure 15.2-15 shows graphically the simulated transient. The initial portion of the transient is similar to the load rejection transient. At two seconds, the loss of the power generation busses signal produces a scram and activation of the ICs. The load rejection initiation of the SCRRI/SRI function is not credited. At approximately 6 seconds the turbine bypass valves are assumed no longer available to bypass the steam to the main condenser. The MSIV closure is produced at 14 seconds due to low condenser vacuum signal. The CRD high pressure injection is initiated due to low water level (Level 2), but the HP_CRD flow is delayed until diesel power is available (145 seconds). In the case where HP_CRD is unavailable for level control, the system response is similar to the station blackout event described in Subsection 15.5.5, which demonstrates that level can be maintained above the top of active fuel with the ICS as the primary success path. In either case, there is no significant increase in fuel temperature. The number of rods in boiling transition remains within the acceptance criterion for AOOs. Hence, fuel thermal margins are not threatened and the design basis is satisfied. Consequently, this event does not need to be reanalyzed for specific core configurations.

15.2.5.2.4 Barrier Performance

Peak nuclear system pressure at the vessel bottom remains below the ASME code upset limit. Peak pressure in the main steamline remains below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool.

15.2.5.2.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.5.3 Loss of All Feedwater Flow

Identification of Causes

A loss of FW flow could occur from pump failures, operator errors, or reactor system variables such as a high vessel water level (Level 8) trip signal.

15.2.5.3.1 Sequence of Events and Systems Operation

Sequence of Events

Table 15.2-22 lists the sequence of events for Figure 15.2-16.

Identification of Operator Actions

The operator should ensure ICS actuation and CRD injection transfer to high pressure injection mode so that water inventory is maintained in the reactor vessel. The operator should also monitor reactor water level, the pressure control, and the TG auxiliaries during shutdown.

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

- Verify that all rods are inserted;
- Follow the scram procedure;
- Monitor reactor water level and pressure; verify that CRD flow is controlling water level;
- Verify IC system initiation; use the IC system to control pressure;
- Monitor turbine coastdown and break vacuum before the loss of steam seals (check turbine auxiliaries for proper operation);
- When desired, the RWCU/SDC system can be put into service.

Systems Operation

Loss of FW flow results in a reduction of vessel inventory, causing the vessel water level to drop. The first corrective action is the loss of the power generation busses scram trip actuation. This scram trip function meets the single-failure criterion.

15.2.5.3.2 Core and System Performance

The results of this transient simulation are presented in Figure 15.2-16. The initial water level is assumed at Level 4. Feedwater flow terminates, and the loss of the power generation busses scram signal is assumed (with activation of the ICS simultaneously). Subcooling decreases, causing a reduction in core power level and pressure. As the core power level is reduced, the turbine steam flow starts to drop off because of the action of the pressure regulator in attempting to maintain pressure. Water level continues to drop, and the vessel level (Level 3) scram trip setpoint is reached. Note that the reactor has been scrammed previously. The vessel water level continues to drop to Level 2. At that time, CRD high pressure injection and closure of all MSIVs are produced (with 30 second delay). In case HP_CRD is unavailable for level control, the system response is similar to the station blackout event described in Subsection 15.5.5, which demonstrates that level can be maintained above the top of active fuel with the ICS as the primary success path. In either case, the number of rods in boiling transition remains within the acceptance criterion for AOOs because increases in the heat flux are not experienced. Consequently, this event does not need to be reanalyzed for specific core configurations.

15.2.5.3.3 Barrier Performance

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

15.2.5.3.4 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.6 AOO Analysis Summary

The results of the system response analyses are presented in Table 15.2-5. Based on these results, the limiting AOO events have been identified. The potentially limiting events that establish the CPR operating limit are identified below. The results of the system response analyses for the initial core loading documented in Reference 15.2-3 are provided in Reference 15.2-4. System response analyses bounding operation in the feedwater temperature operating domain are documented in Reference 15.2-5.

For the core loading in Reference 15.2-2, the resulting OLMCPR is 1.30, using the methodologies listed in Subsections 4.4.3.1.3 and 4.4.2.1.3 and Reference 15.2-1. The operating limit for each fuel cycle is documented in the COLR in accordance with Technical Specifications. The following AOOs are potentially limiting with respect to OLMCPR:

- Loss of Feedwater Heating
- Closure of One Turbine Control Valve
- Generator Load Rejection with Turbine Bypass
- Generator Load Rejection with a Single Failure in the Turbine Bypass System
- Turbine Trip with Turbine Bypass
- Inadvertent Isolation Condenser Initiation

15.2.7 COL Information

15.2-1-A Initial Core Design AOOs – (Deleted)

15.2-2-H Reload Core Design AOOs – (Deleted)

15.2-3-A Assumptions – (Deleted)

15.2.8 References

15.2-1 GE Nuclear Energy, "TRACG Application for Anticipated Operational Occurrences Transient Analysis" NEDE-32906P-A, Revision 1, April 2003.

15.2-2 Global Nuclear Fuel, "GE14 for ESBWR Nuclear Design Report", NEDC-33239-P, Class III (Proprietary), Revision 2, April 2007, NEDO-33239, Class I (Non-proprietary), Revision 2, April 2007.

15.2-3 Global Nuclear Fuel, "ESBWR Initial Core Nuclear Design Report", NEDC-33326-P, Class III (Proprietary), Revision 0, July 2007, NEDO-33326, Class I (Non-proprietary), Revision 0, July 2007.

15.2-4 GE-Hitachi Nuclear Energy, "ESBWR Initial Core Transient Analyses", NEDO-33337 Class I, Revision 0, Scheduled September 2007.

15.2-5 GE-Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis", NEDO-33338 Class I, Revision 0, Scheduled September 2007.

Table 15.2-1

Input Parameters, Initial Conditions and Assumptions Used In AOO and Infrequent Event Analyses

Parameter	Value
Thermal Power Level, MWt	4500
Core Flow, kg/s (Mlbm/hr)	10130 (80.40)
Steam Flow, kg/s (Mlbm/hr) Analysis Value	 2435 (19.3)
Feedwater Flow Rate, kg/s (Mlbm/hr) Analysis Value Total Flow For All Pumps Runout, % of rated at 1065 psig (At rated dome pressure, 1025 psig) The condensate and feedwater system in conjunction with the feedwater control system provide inventory equivalent to 240 s of rated feedwater flow after MSIV isolation. The condensate and feedwater system in combination with feedwater control system limit the maximum feedwater flow for a single pump to 75% of rated flow following a single active component failure or operator error.	 2429 (19.3) 155 (164)
Feedwater Temperature Rated, °C (°F) FW Heating Temperature Loss, Δ°C (Δ°F) Loss of FW Heating Setpoint (SCRRI/SRI Initiation), Δ°C (Δ°F)	 216 (420) 55.6 (100) 16.67 (30)
Vessel Dome Pressure, MPaG (psig)	7.07 (1025)
Vessel Core Pressure, MPaG (psig)	7.17 (1040)
Turbine Bypass Capacity, % of rated	110
Total Delay Time from TSV or TCV to the start of BPV Main Disc Motion, s	0.02
Total Delay Time from TSV or TCV to 80% of Total Capacity	0.17
TCV Closure Times, s Fast Closure Analysis Value (Bounding) Assumed Slow Closure Analysis Value	 0.08 2.5

Table 15.2-1

Input Parameters, Initial Conditions and Assumptions Used In AOO and Infrequent Event Analyses

Parameter	Value
TSV Closure Times, s	0.100
% of Rated Steam Flow that can pass through 3 Turbine Control Valves	85 (Partial Arc)
Turbine Inlet Pressure, MPaG (psig)	6.57 (953)
Fuel Lattice	N
Core Leakage Flow, %	9.4
MCPR Operating Limit	1.30
Control Rod Drive Position versus Time	Table 15.2-2 & 3
Core Design used in TRACG Simulations Exposure:	Reference 15.2-2 Middle of Cycle and End of Cycle
Safety Relief Valve (SRV) capacity, %NBR (103% accumulation)(1) At design pressure, MPaG (psig) Quantity Installed	89.5 8.618 (1250) 18
Safety Function Delay, s	0.2
Safety Function Opening Time, s	1.5
Analysis values for SRV setpoints Low Setpoint, MPaG (psig) High Setpoint, MPaG (psig)	 8.618 (1250) 8.756 (1270)
Closure Scram Position of 2 or More MSIVs, % open Maximum delay time, s	85 0.06
MSIV Minimum Closure Time, s MSIV Maximum Closure Time, s	3.0 5.0

Table 15.2-1

Input Parameters, Initial Conditions and Assumptions Used In AOO and Infrequent Event Analyses

Parameter	Value
MSIV Closure Profile used to Bound Minimum Closure Time, s 100% open 100% open 1% open 0% open	 0.0 0.6 1.7 3.0
High Flux Trip, % NBR, Sensor Time Constant	125.0 0.03
TSV Closure Scram Position of 2 or more TSV, % open Trip Time delay, s	85 0.06
TCV Fast Closure Scram Trip, s	0.08
High Pressure Scram, MPaG (psig) Maximum scram delay, s	7.619 (1105) 0.7
High Suppression Pool Temperature Scram trip, °C (°F), Maximum Delay Time, s	48.9(120) 1.05
High Suppression Pool Temperature FAPCS actuation, °C (°F)	43.3 (110)
Vessel level Trips (above bottom vessel) Level 9 – (L9), m (in) Level 8 – (L8), m (in) Level 4 – (L4), m (in) Level 3 – (L3), m (in) Level 2 – (L2), m (in) Level 1 – (L1), m (in) Level 0.5 – (L0.5) m (in)	 22.39 (881.5) 21.89 (861.8) 20.60 (811.2) 19.78 (778.7) 16.05 (631.9) 11.50 (452.8) 8.45 (332.7)
APRM Simulated Thermal Power Trip Scram, % NBR Time Constant, s	 115 7
Total Steamline Volume, m^3 (ft^3)	135 (4767)
CRD Hydraulic System minimum capacity, m^3/hr (gpm), Capacity in kg/s (Mlbm/hr) for 990 kg/m^3 density (61.8 lbm/ft^3)	235.1 (1035) 64.6 (0.513)

Table 15.2-1

Input Parameters, Initial Conditions and Assumptions Used In AOO and Infrequent Event Analyses

Parameter	Value
Maximum time delay from Initiating Signal (Pump 1 & 2), s If offsite power is not available	10 & 25 145
Isolation Condensers Max Initial Temperature, °C (°F) Minimum Initial Temperature, °C (°F) Time To injection valve full open (Max), s(2) Heat Removal Capacity for 4ICs, MW (% Rated Power) Isolation Condensers volume, 4 Units, from steam box to discharge at vessel m^3 (ft^3)	 40 (104) 10 (50) 31 (1) 135 (3%) 56.1 (1981)

(1) The SRV capacity used in the analysis is less than the ASME rated capacity noted in Table 5.2-2.

(2) In the analysis, after 1 s logic delay, the IC opening valve curve began to open at 15 s for a total opening time of 30 s. For IICI the valve begins to open at 15 s with a opening time of 7.5 s.

Table 15.2-2

CRD Scram Times for Vessel Bottom Pressures Below 7.481 MPa gauge (1085 psig)

Rod Insertion (%)	Scram Time (seconds) (After De-energization (0.05 s)) Used in Analysis
0	0.0
0	0.2
10	0.34
40	0.80
60	1.15
100	2.23

Table 15.2-3

CRD Scram Times for Bottom Vessel Pressures Between 7.481 MPa gauge (1085 psig) and 8.618 MPa gauge (1250 psig)

Rod Insertion (%)	Scram Time (seconds) (After De-energization (0.05 s)) Used in Analysis
0	0.0
0	0.2
10	0.37
40	0.96
60	1.36
100	2.95

Table 15.2-4

Sequence of Events for Loss of Feedwater Heating

Time (s)	Event
0	Initiate a 55.6°C (100°F) temperature reduction in the FW system
22.7 (est)	RC&IS initiates Selected Control Rod Run-In and Selected Rod Insertion (SCRRI/SRI)
24 (est)	Initial effect of unheated FW starts to raise core power level
24.0	First SRI group inserts (one HCU, 2 control rods, fails to actuate) and SCRRI start insertion
34.0	Second SRI group inserts
44.0	Third SRI group inserts
54.0	Fourth SRI group inserts
90.0	Steam flow below 60% of rated
134.0	SCRRI/SRI groups totally inserted
140.0	Power below 60% of rated
250.0 (est.)	Reactor variables settle into new steady state

* See Figure 15.2-1. This Figure has 20 s of steady state, a time of 0 s on the table corresponds to 20 s on the figure.

Table 15.2-5

Results Summary of Anticipated Operational Occurrence Events

Sub-section I.D.	Description	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPaG (psig)	Max. Vessel Bottom Pressure, MPaG (psig)	Max. Steamline Pressure, MPaG (psig)	Max. Core Average Surface Heat Flux, % of Initial	ΔCPR/ICPR or Minimum Water Level (m over TAF)
15.2.1.1	Loss of Feedwater Heating	100.2	7.08 (1027)	7.21 (1046)	7.04 (1024)	100	0.04
15.2.2.1	Closure of One Turbine Control Valve. FAST/SLOW	124 112	7.20 (1043) 7.20 (1043)	7.33 (1063) 7.33 (1063)	7.16 (1038) 7.16 (1038)	102 102	0.04 0.03
15.2.2.2	Generator Load Rejection with Turbine Bypass	135	7.15 (1037)	7.29 (1057)	7.28 (1056)	102	0.03
15.2.2.3	Generator Load Rejection with a Single Failure in the Turbine Bypass System	168	7.39 (1072)	7.53 (1091)	7.39 (1070)	103	0.03
15.2.2.4	Turbine Trip with Bypass	120	7.12 (1033)	7.26 (1053)	7.20 (1043)	101	0.02
15.2.2.5	Turbine Trip with a Single Failure in the Turbine Bypass System	146	7.37 (1069)	7.50 (1088)	7.37 (1067)	102	0.02
15.2.2.6	Closure of One MSIV	114	7.16 (1038)	7.30 (1059)	7.13 (1033)	101	0.02
15.2.2.7	Closure of All MSIV	103	7.76 (1126)	7.89 (1143)	7.76 (1126)	100	≤ 0.01
15.2.2.8	Loss of Condenser Vacuum	110	7.11 (1031)	7.26 (1053)	7.20 (1044)	100	≤ 0.01
15.2.2.9	Loss of Shutdown Cooling Function of RWCU/SDC	{ - - }	{ - - }	{ - - }	{ - - }	{ - - }	{ - - }
15.2.4.1	Inadvertent Isolation Condenser Initiation	113	7.08 (1027)	7.22 (1047)	7.04 (1021)	109	0.08
15.2.4.2	Runout of One Feedwater Pump	103	7.08 (1027)	7.22 (1047)	7.05 (1023)	100	<0.01>
15.2.5.1	Opening of One Turbine Control or Bypass Valve	102	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	≤ 0.01
15.2.5.2	Loss of Non-Emergency AC Power to Station Auxiliaries	136	7.13 (1035)	7.28 (1056)	7.28 (10546)	102	5.40 m

Table 15.2-5

Results Summary of Anticipated Operational Occurrence Events

Sub-section I.D.	**Description**	**Max. Neutron Flux,** % NBR	**Max. Dome Pressure,** MPaG (psig)	**Max. Vessel Bottom Pressure,** MPaG (psig)	**Max. Steamline Pressure,** MPaG (psig)	**Max. Core Average Surface Heat Flux,** % of Initial	**ΔCPR/ICPR or Minimum Water Level** (m over TAF**)**
15.2.5.3	Loss of Feedwater Flow	100	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	5.28 m

Table 15.2-6

Sequence of Events for Fast Closure of One Turbine Control Valve

Time (s)	**Event***
0	Simulate one main TCV to fast close
0	Failed TCV starts to close
0.08	TCV closed
1.13	Turbine bypass valves start to open
30.0	New steady state is established

* See Figure 15.2-2.

Table 15.2-7

Sequence of Events for Slow Closure of One Turbine Control Valve

Time (s)	**Event***
0	Simulate one main TCV to slow close
0	Failed TCV starts to close
2.5	TCV closed
2.9	Turbine bypass valves start to open
30.0	New steady state is established

* See Figure 15.2-3.

Table 15.2-8

Sequence of Events for Generator Load Rejection with Turbine Bypass

Time (s)	Event*
-0.015	Turbine-generator detection of loss of electrical load
0.0	Turbine-generator load rejection sensing devices trip to initiate TCVs fast closure and main turbine bypass system operation
0.02	Turbine bypass valves start to open
0.08	Turbine control valves closed
0.17	Turbine bypass opened at 80%
0.20	SCRRI activated
1.0	First SRI group inserts and SCRRI start insertion
10.0	Second SRI group inserts
20.0	Third SRI group inserts
30.0	Fourth SRI group inserts
50.0	Steam flow below 60% of rated
111.0	SCRRI groups totally insertion
140	Core power reaches 60%
0.0-200.0	FW temperature is decreasing because of loss of turbine extraction steam to FW heaters
200	New steady state is established

* See Figure 15.2-4.

Table 15.2-9

Sequence of Events for Generator Load Rejection with a Single Failure in the Turbine Bypass System

Time (s)	**Event***
-0.015	Turbine-generator detection of loss of electrical load
0.0	Turbine-generator load rejection sensing devices trip to initiate TCVs fast closure and main turbine bypass system operation
0.02	Turbine bypass valves start to open (Half fail to open)
0.08	Turbine control valves closed
0.20	Not enough turbine bypass availability is detected and the plant is scrammed
0.45	Control Rods begin to enter in the core
2.90	L3 level is reached
Long term	L2 is not reached, new steady state

* See Figure 15.2-5.

Table 15.2-10

Sequence of Events for Turbine Trip with Turbine Bypass

Time (s)	**Event** *
0.0	Turbine trip initiates closure of main stop valves.
0.0	Turbine trip initiates bypass operation.
0.02	Turbine bypass valves start to open to regulate pressure.
0.10	Turbine stop valves closed
0.17	Turbine bypass opened at 80%
0.20	SCRRI/SRI activated
1.0	First SRI group inserts and SCRRI start insertion
10.0	Second SRI group inserts
20.0	Third SRI group inserts
30.0	Fourth SRI group inserts
50.0	Steam flow below 60% of rated
111.0	SCRRI groups totally inserted
140.	Core power reaches 60%
0.0-200.0	FW temperature is decreasing because of reduced turbine steam flow
200	New steady state is established

* See Figure 15.2-6.

Table 15.2-11

Sequence of Events for Turbine Trip with a Single Failure in the Turbine Bypass System

Time (s)	**Event ***
0.0	Turbine trip initiates closure of main stop valves.
0.0	Turbine trip initiates bypass operation.
0.02	Turbine bypass valves start to open to regulate pressure (Half fail to open).
0.1	Turbine stop valves closed.
0.20	Not enough turbine bypass availability is detected and the plant is scrammed
0.45	Control Rods begin to enter in the core.
Long term	L2 is not reached, new steady state (1) L2 is reached and HP_CRD is activated to recover the level

* See Figure 15.2-7.

Table 15.2-12

Sequence of Events for Closure of one MSIV

Time (s)	**Event** *
0.0	Closure of one MSIV.
2.0	Maximum neutron flux
2.8	Turbine Bypass open.
3.0	MSIV is closed
40.0	New steady state is reached

* See Figure 15.2-8.

Table 15.2-13

Sequence of Events for Closure of all MSIV

Time (s)	**Event** *
0.0	Closure of all MSIVs (MSIV).
0.78	MSIVs reach 85% open.
0.85	MSIVs position trip scram initiated.
1.82	IC initiated
2.84	L3 is reached
3.0	MSIVs are closed
4.30	Reactor pressure reaches its peak value.
20.1	L2 is reached
30.1	HP_CRD is initiated
31.82	The ICs valves are fully open .
Long term	The FW is available for a period of time to control the Water Level in vessel.

* See Figure 15.2-9.

Table 15.2-14

Typical Rates of Decay for Loss of Condenser Vacuum

Cause	Estimated Vacuum Decay Rate
Failure of Isolation of Steam Jet Air Ejectors	1 inch/minute
Loss of One or More Circulating Water Pumps	0.5 inch/s

Table 15.2-15

Sequence of Events for Loss of Condenser Vacuum

Time (s)	**Event ***
-3.0	Initiate simulated loss of condenser vacuum trip.
0.0	Low condenser vacuum main turbine trip and Scram actuated.
0.02	Turbine bypass valves start to open to regulate pressure.
0.1	Turbine stop valves close.
0.20	Scram initiated
2.4	Turbine Bypass initiates closure, because of low vessel pressure
6.0	Low condenser vacuum forces main turbine bypass valve closure.
6.5	Bypass valve is closed
8.0	Low condenser vacuum initiates MSIV closure.
8.8	MSIV closure initiates IC (85% position).
9.8	IC is activated (1 s delay from MSIV closure signal)
14.60	L2 water level is reached.
24.8	HP_CRD is activated
39.8	The ICs valves are fully open
Long term	The FW is available for a period of time to control the Water Level in vessel.

* See Figure 15.2-10.

Table 15.2-16

Trip Signals Associated With Loss of Condenser Vacuum

Vacuum (cm (in) of Hg)	**Protective Action Initiated**
5.2-9.1 (2-3.6)	Normal Vacuum Range
17.8-25.4 (7-10)	Main Turbine Trip (Stop Valve Closure) & Scram
50.8-58.4 (20-23)	MSIV Closure and Turbine Bypass Valve Closure

Table 15.2-17

Sequence of Events for Inadvertent Isolation Condenser Initiation

Time (s)	**Event** *
10	Simulate IC cold water injection.
25	IC drainage valve begins to open
32.5	IC drainage valve is fully open
≈50	Full power established for IC.
≈200	MCPR is recovered
≈300	Power increase effect stabilized.

* See Figure 15.2-11.

Table 15.2-18

Single Failure Modes for Digital Controls

Modes	Description	Effects
1	Critical input failure	None - Signal from redundant transmitter is utilized - Operator informed of failure
2	Input failure while one sensor out of service	Possible system failure. Adverse effects minimized when possible
3	Operator hardwired switch single contact failure for critical manual FTDC initiation functions (e.g. for the Manual Feedwater Runback initiation pushbuttons)	None - Triplicated contacts are used for hardwired switches associated with critical manual initiation functions
4	Processor channel failure	None - Redundant processors maintain control - Operator informed of failure
5	Operating channel processor failure while another channel is failed or has been taken out of service	System failure; operator alarm activated and if needed, mitigating function of another system is activated such as Main Turbine Trip initiation upon SBPC system failure detection
6	Hardware/Firmware Output module failure	Loss of one output associated with the control of an actuator (e.g. loss of FWCS Low Flow Control Valve position demand signal). When practical, actuator lockup output from a separate Hardware/Firmware Output module is activated to minimize the plant disturbance; or redundant FTDC actuator output signals are provided with internal voting logic in the actuator equipment itself so that there will be no system disturbance because each of the redundant actuator signals are provided from separate Hardware/Firmware Output modules.
7	Actuator failure	Loss of control of one actuator (e.g. loss of speed control of one FW Pump ASD only).

Table 15.2-19

Sequence of Events for Runout of One Feedwater Pump

Time (s)	**Events** *
0	Initiate simulated increase in speed of one FW pump. The maximum pump flow is 75% at rated conditions.
~0.1	Feedwater controller starts to reduce the FW flow from the FW pumps.
6.0	Vessel water level reaches its peak value and starts to return to its normal value.
21.0 (est.)	Vessel water level returns to its normal value.

* See Figure 15.2-13.

Table 15.2-20

Sequence of Events for Opening of one Turbine Control or Bypass Valve

Time (s)	**Events** *
0	One Turbine Bypass opens
~0.1	TCV closes slightly to control pressure
30.0	New steady state is established

* See Figure 15.2-14.

Table 15.2-21

Sequence of Events for Loss of Non-Emergency AC Power to Station Auxiliaries

Time (s)	**Event** *
0.0	Loss of AC power to station auxiliaries, which initiates a generator trip.
0.0	Additional Failure assumed in transfer to "Island mode", feedwater, condensate and circulating water pumps are tripped.
0.0	Turbine control valve fast closure is initiated.
0.0	Turbine control valve fast closure initiates main turbine bypass system operation.
0.0	Feedwater and condenser pumps are tripped.
0.02	Turbine bypass valves start to open.
0.08	Turbine control valves closed.
2.0 (1)	Loss of power on the four power generation busses is detected and initiates a reactor scram and activation of ICs with 1 s delay.
5.0	Feedwater flow decay to 0.
6.0	Low condenser vacuum setpoint is detected and initiates turbine bypass closure.
6.0	Loss of condenser vacuum rate is reduced due to bypass valve closure
6.5	Vessel water level reaches Level 3
10.0	Vessel water level reaches Level 2.
14.0	Low-Low condenser vacuum signal closes the MSI valves.
18.0	ICs begins to drop cold water inside the vessel
33.0	ICs drainage valve is fully open
145.0	HP_CRD injection mode is initiated.
≈100	The level recovers above 13 m (42.7 ft)
≈550-600	The level recovers above 15 m (49.2 ft)

* See Figure 15.2-15. This Figure has 50 s of steady state to change the initial water level to L4, a time of 0 s on the table corresponds to 50 s on the figure.

(1) Conservatively the insertion of the first SRI previous to the SCRAM is not credited.

Table 15.2-22

Sequence of Events for Loss of All Feedwater Flow

Time (s)	**Event** *
0	Trip of all FW pumps initiated.
2.0	Non FW flow availability initiates reactor scram and initiates IC with 1 s delay
5.0	Feedwater flow decays to zero.
10	Vessel water level reaches Level 2.
18.0	ICs begins to drop cold water inside the vessel
20.0	HP_CRD injection mode is initiated.
33.0	The ICs drainage valves are fully open
40.0	MSIV closure
≈80	The level recovers above 13 m (42.7 ft)
≈500	The level recovers above 15 m (49.2 ft)

* See Figure 15.2-16. This Figure has 50 s of steady state to change the initial water level to L4, a time of 0 s on the table corresponds to 50 s on the figure.

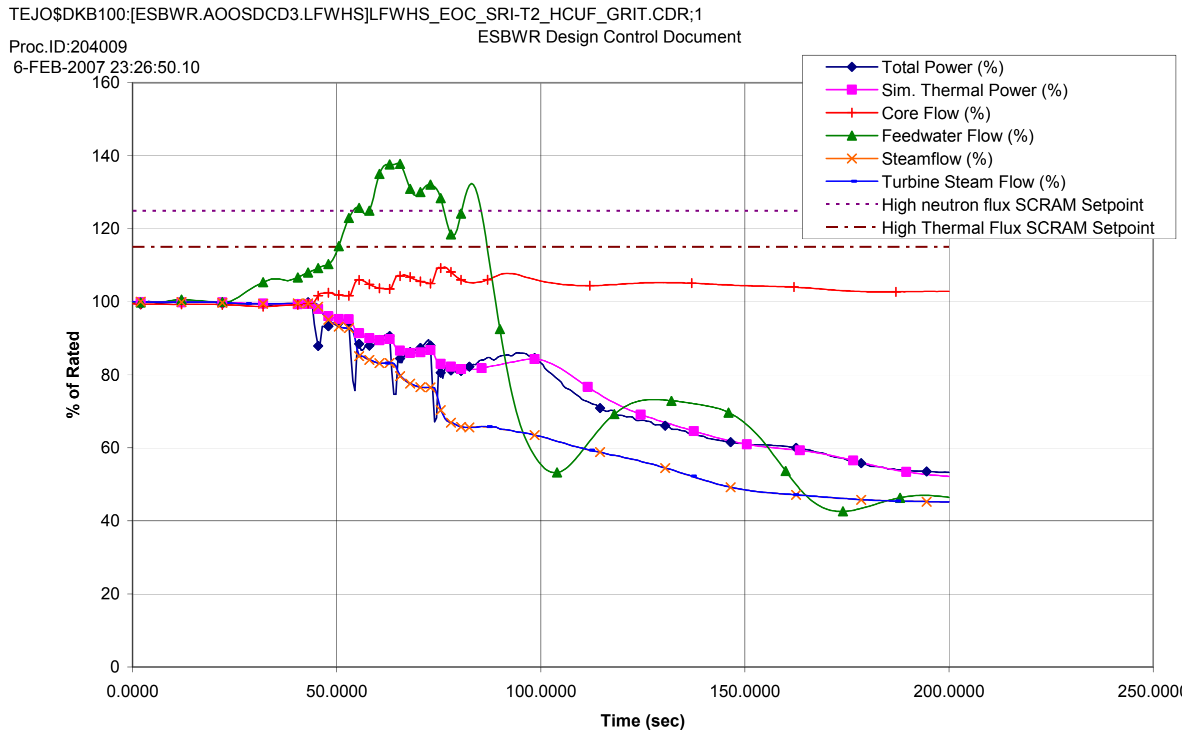


Figure 15.2-1a. Loss of Feedwater Heating

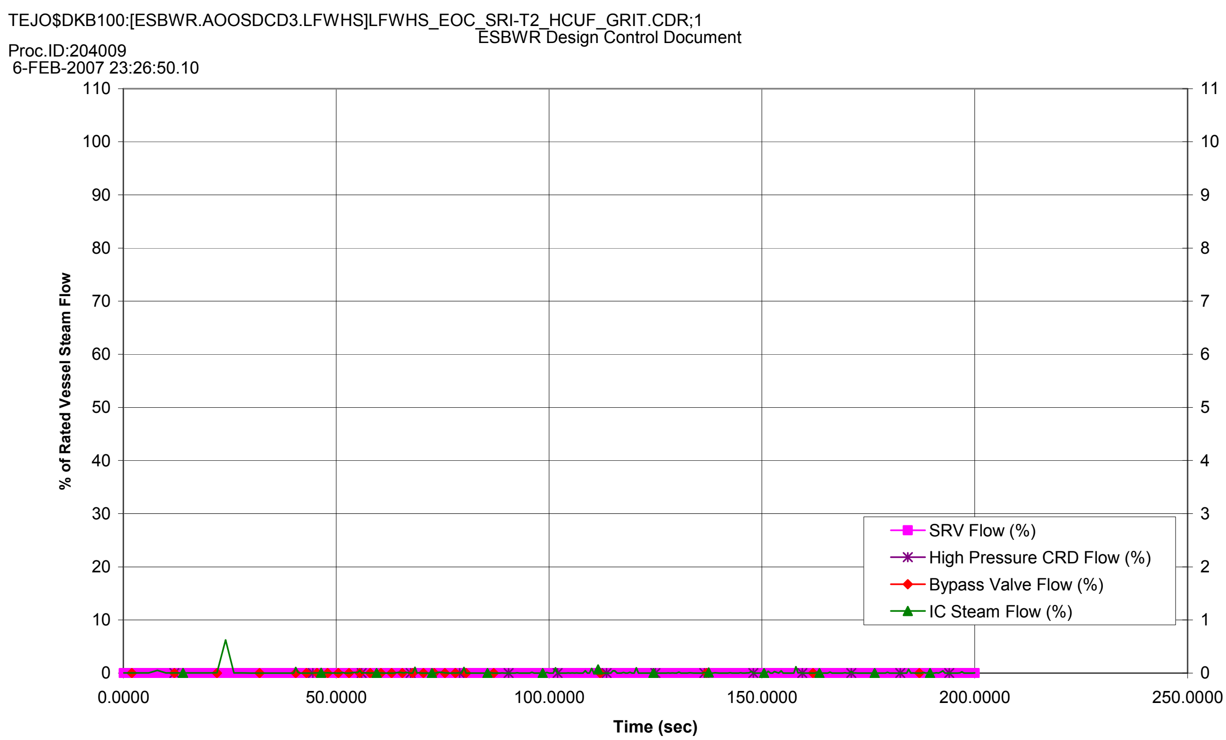


Figure 15.2-1b. Loss of Feedwater Heating

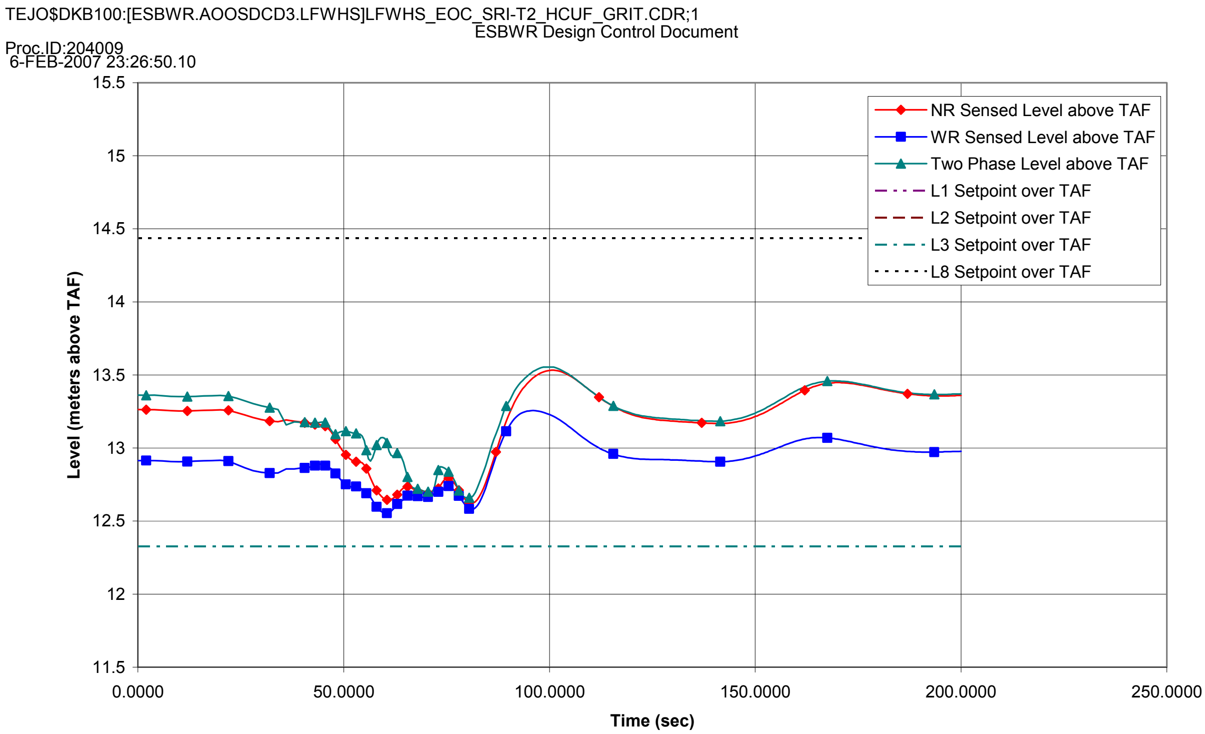


Figure 15.2-1c. Loss of Feedwater Heating

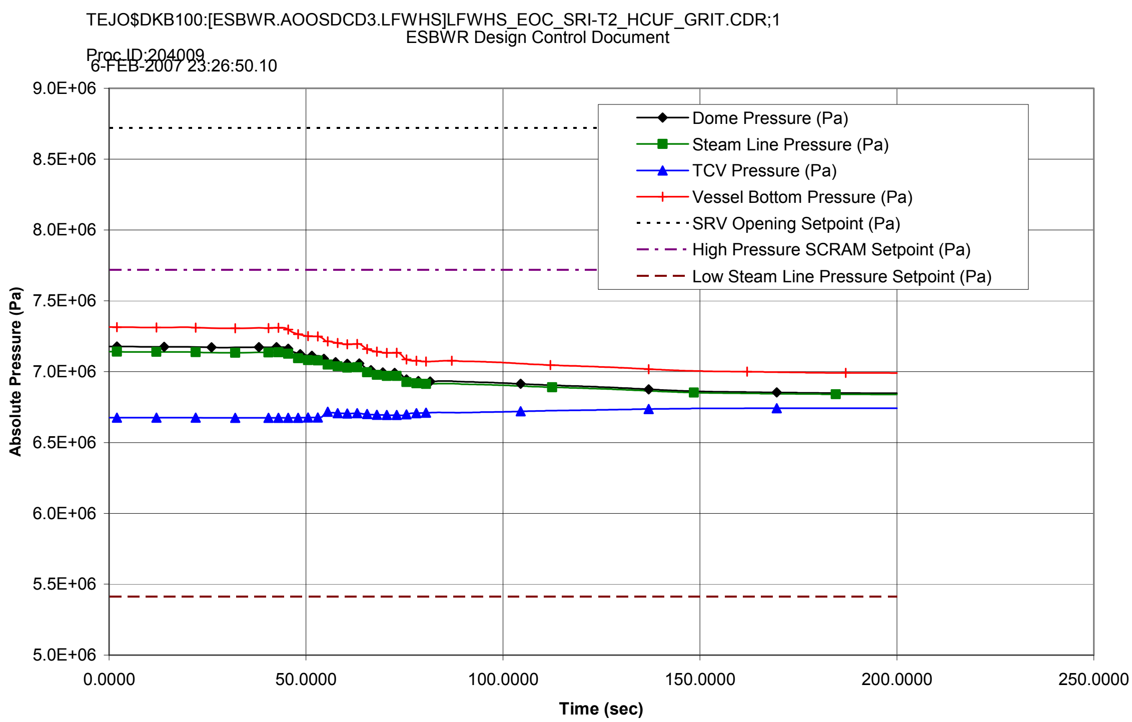


Figure 15.2-1d. Loss of Feedwater Heating

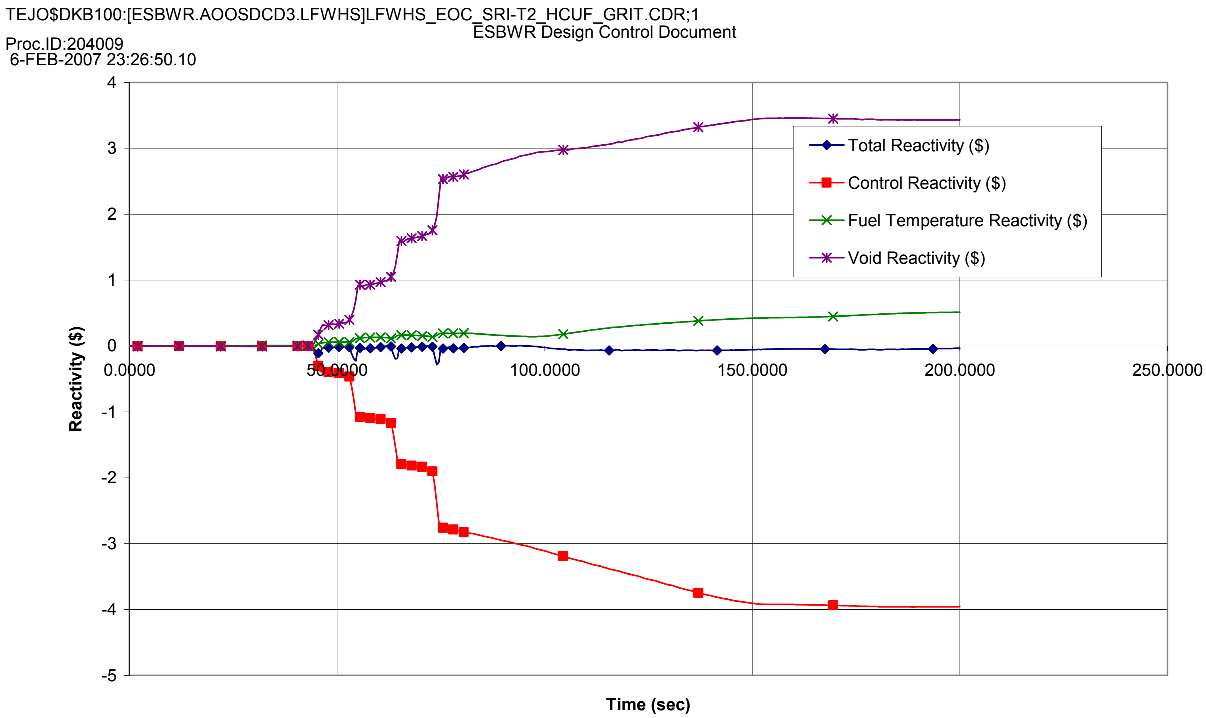


Figure 15.2-1e. Loss of Feedwater Heating

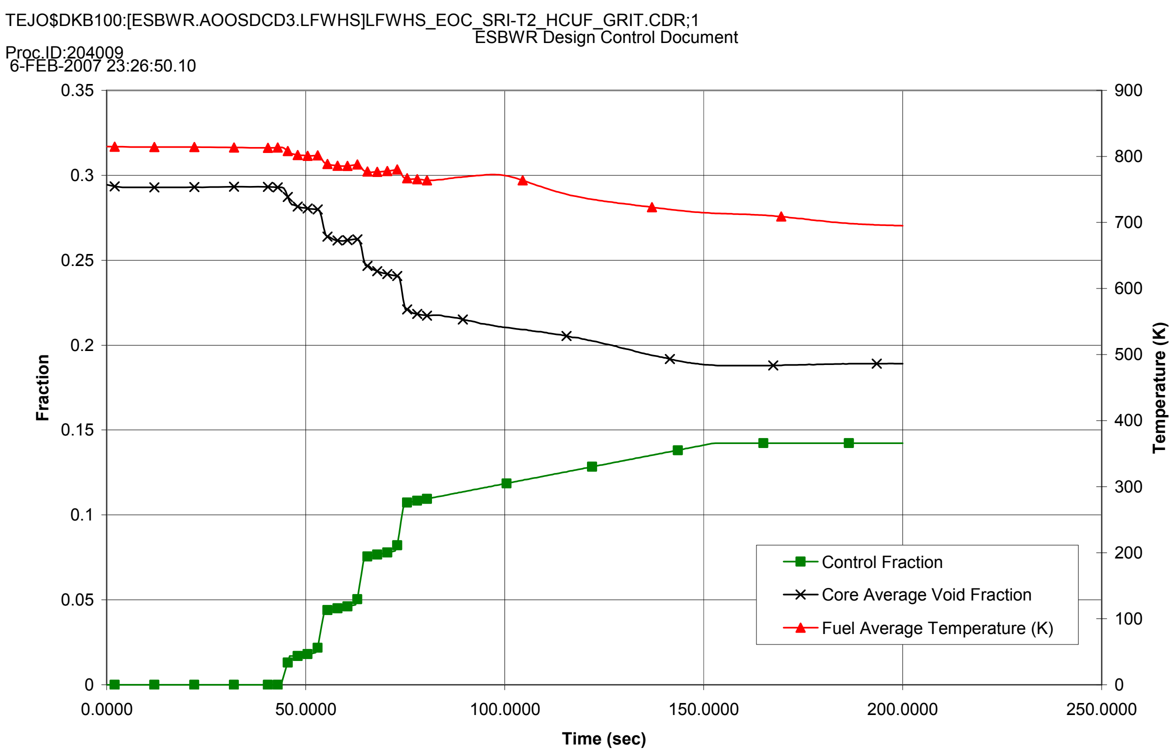


Figure 15.2-1f. Loss of Feedwater Heating

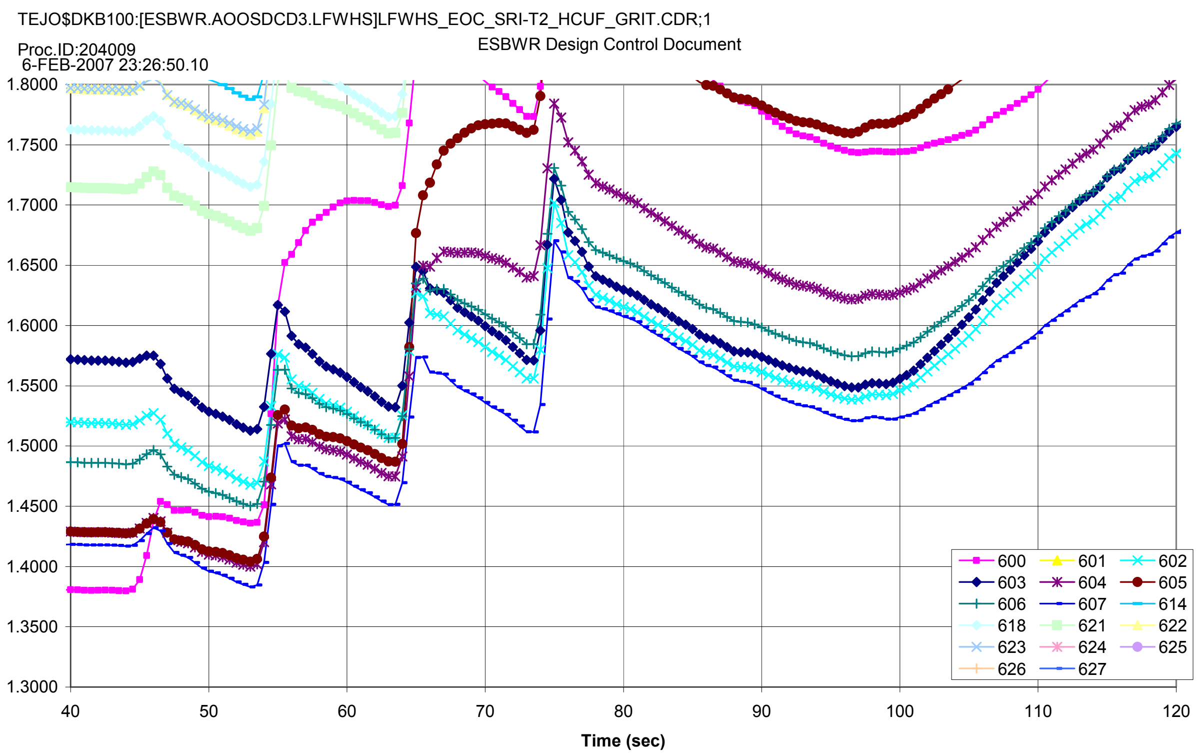


Figure 15.2-1g. Loss of Feedwater Heating

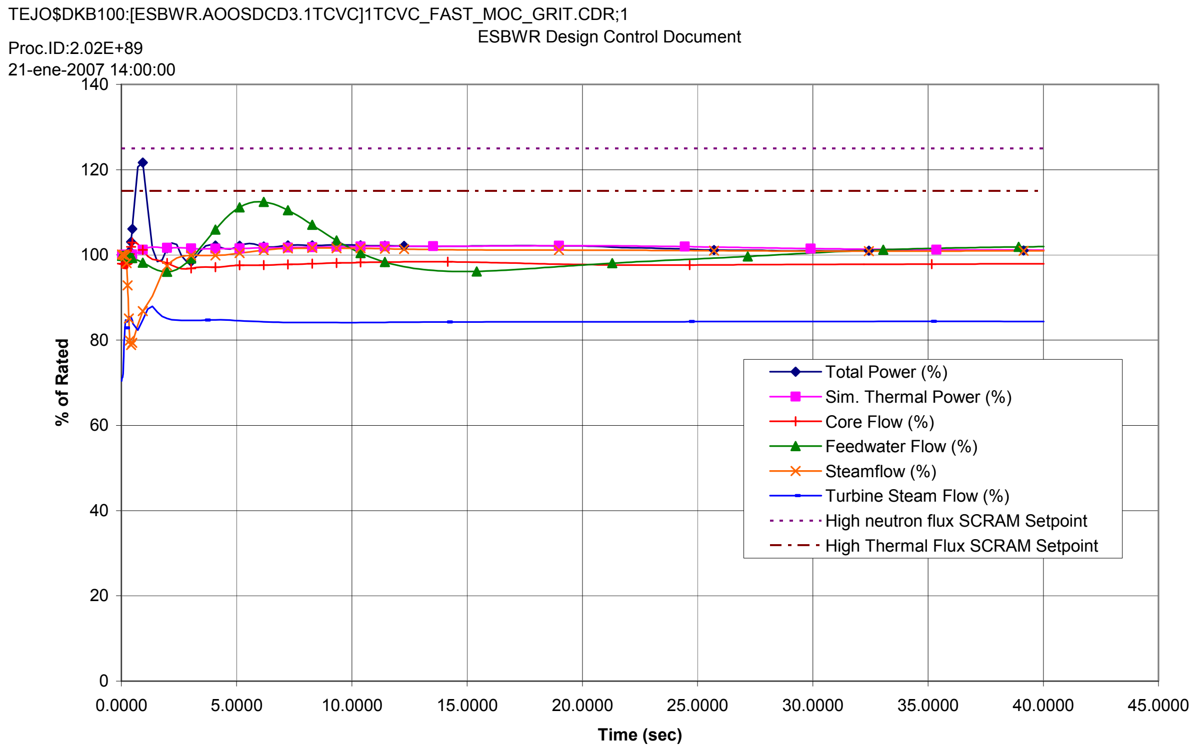


Figure 15.2-2a. Fast Closure of One Turbine Control Valve

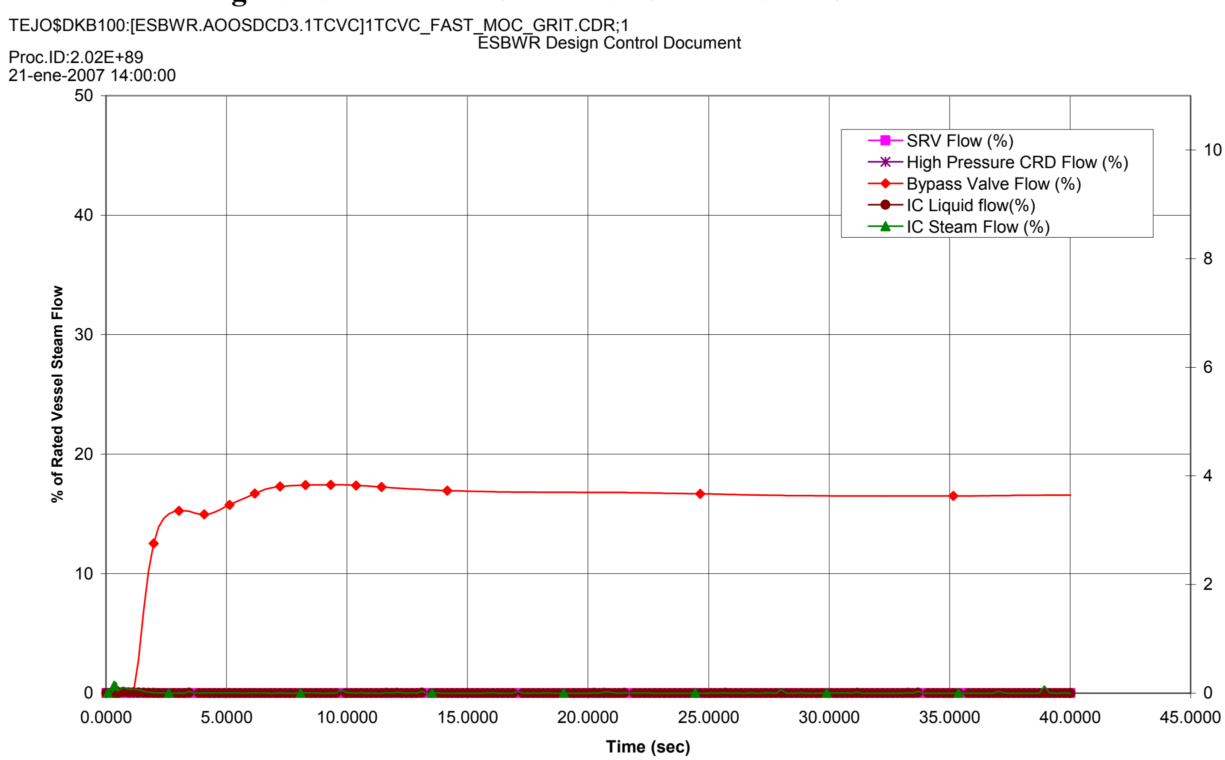


Figure 15.2-2b. Fast Closure of One Turbine Control Valve

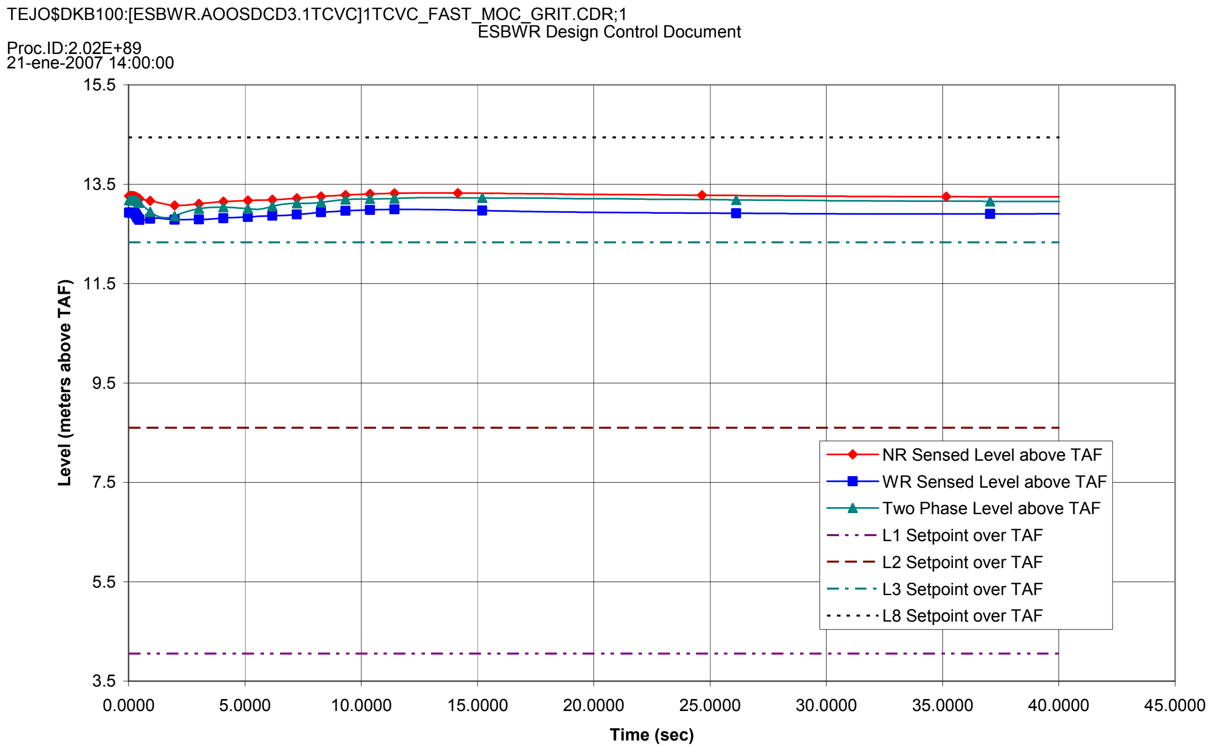


Figure 15.2-2c. Fast Closure of One Turbine Control Valve

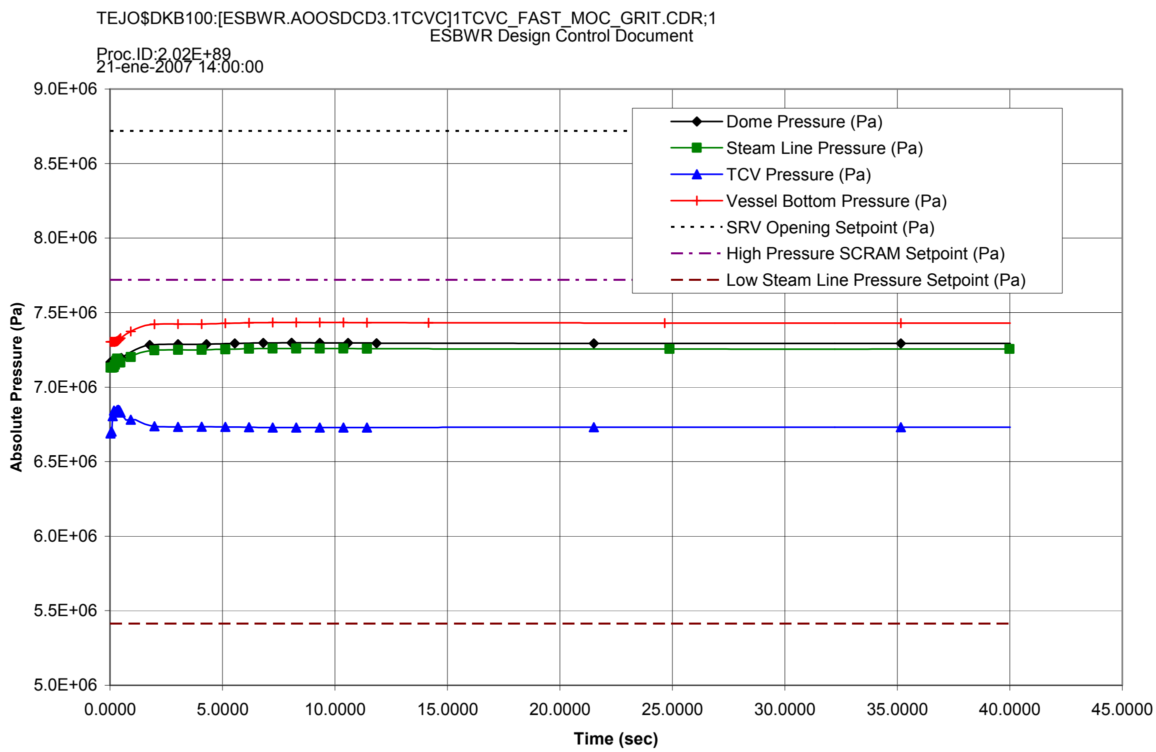


Figure 15.2-2d. Fast Closure of One Turbine Control Valve

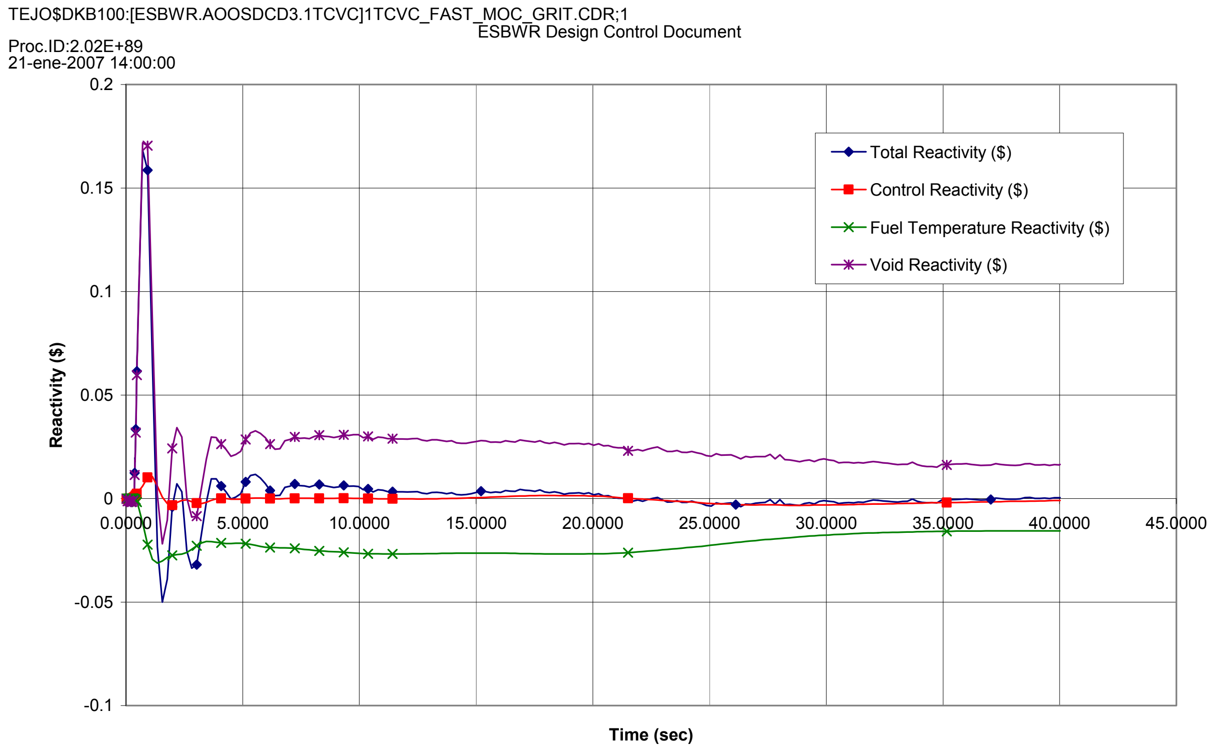


Figure 15.2-2e. Fast Closure of One Turbine Control Valve

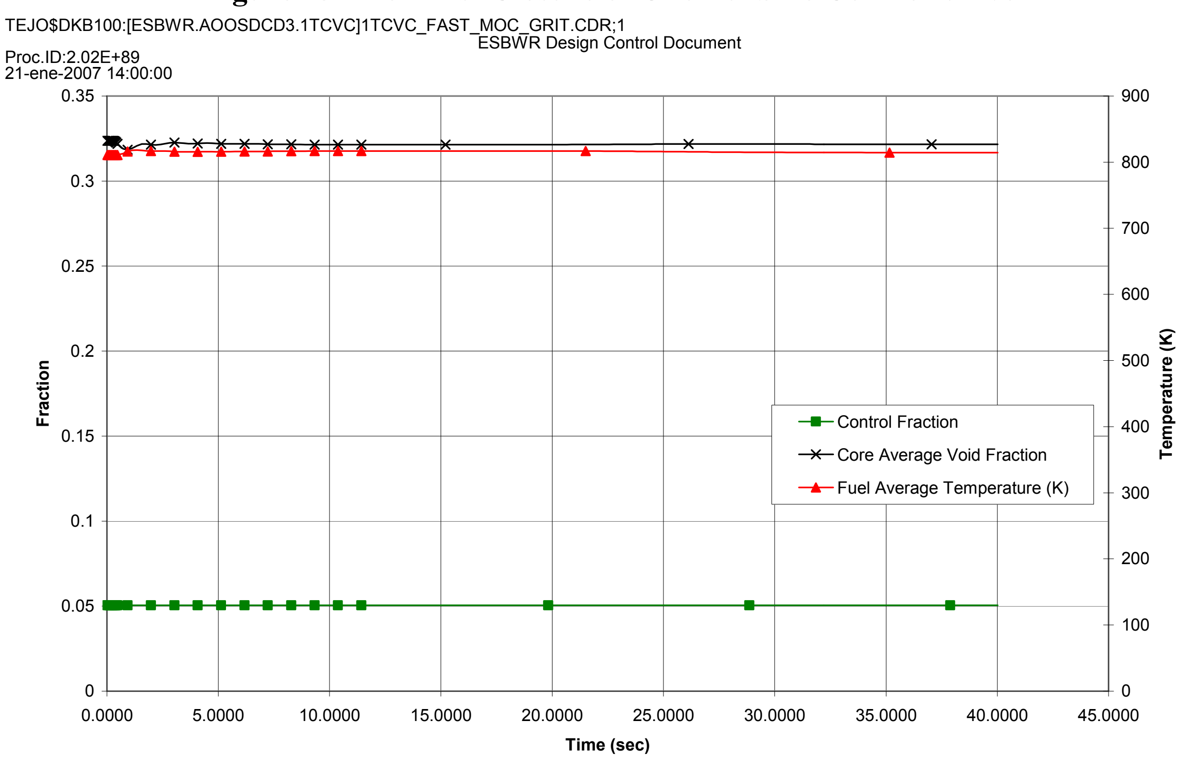


Figure 15.2-2f. Fast Closure of One Turbine Control Valve

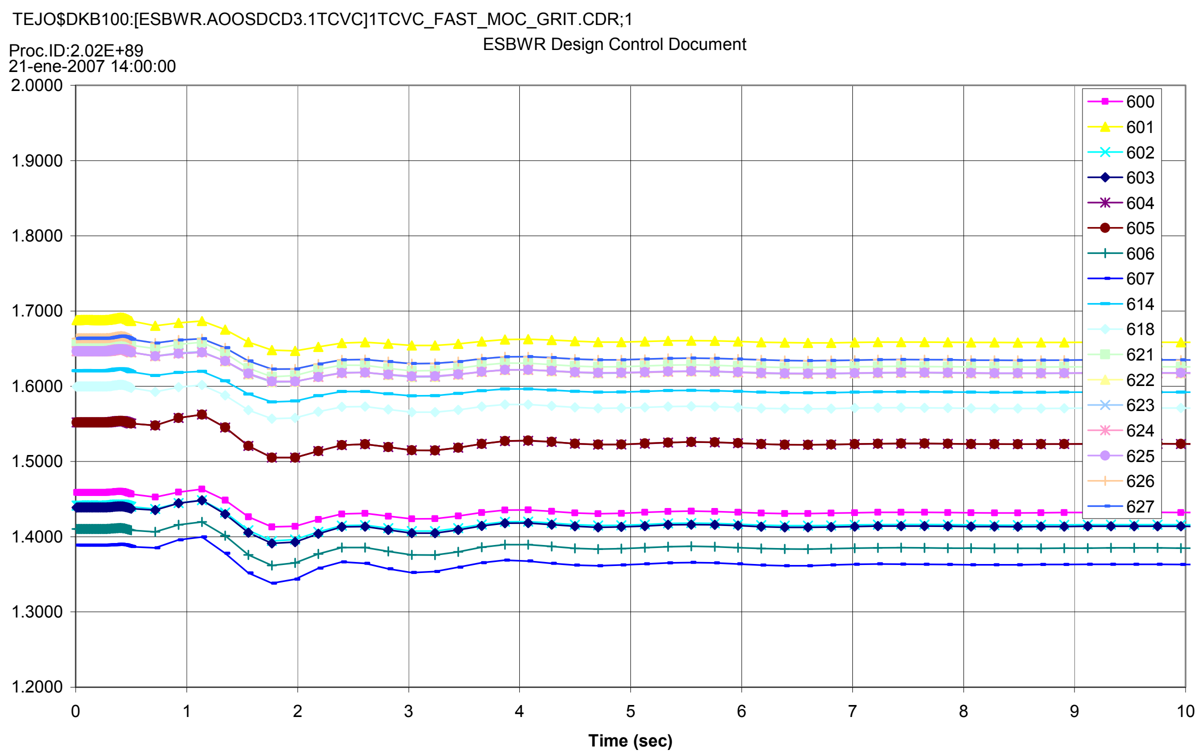


Figure 15.2-2g. Fast Closure of One Turbine Control Valve

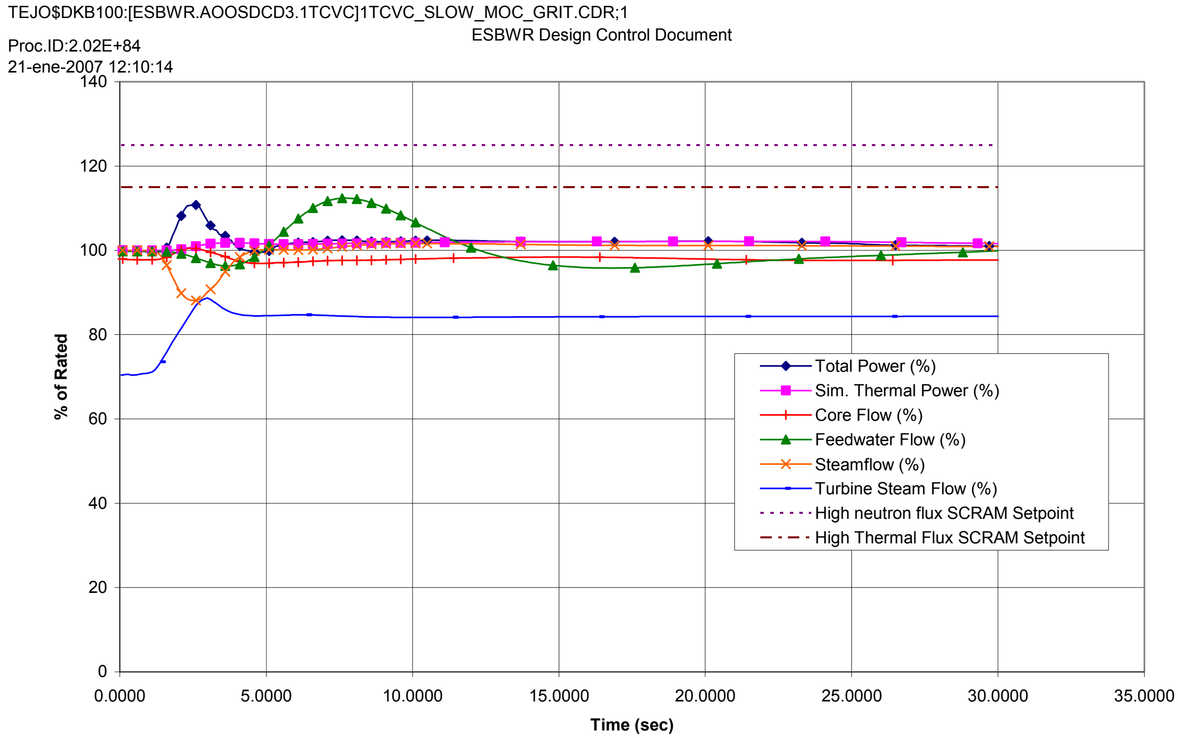


Figure 15.2-3a. Slow Closure of One Turbine Control Valve

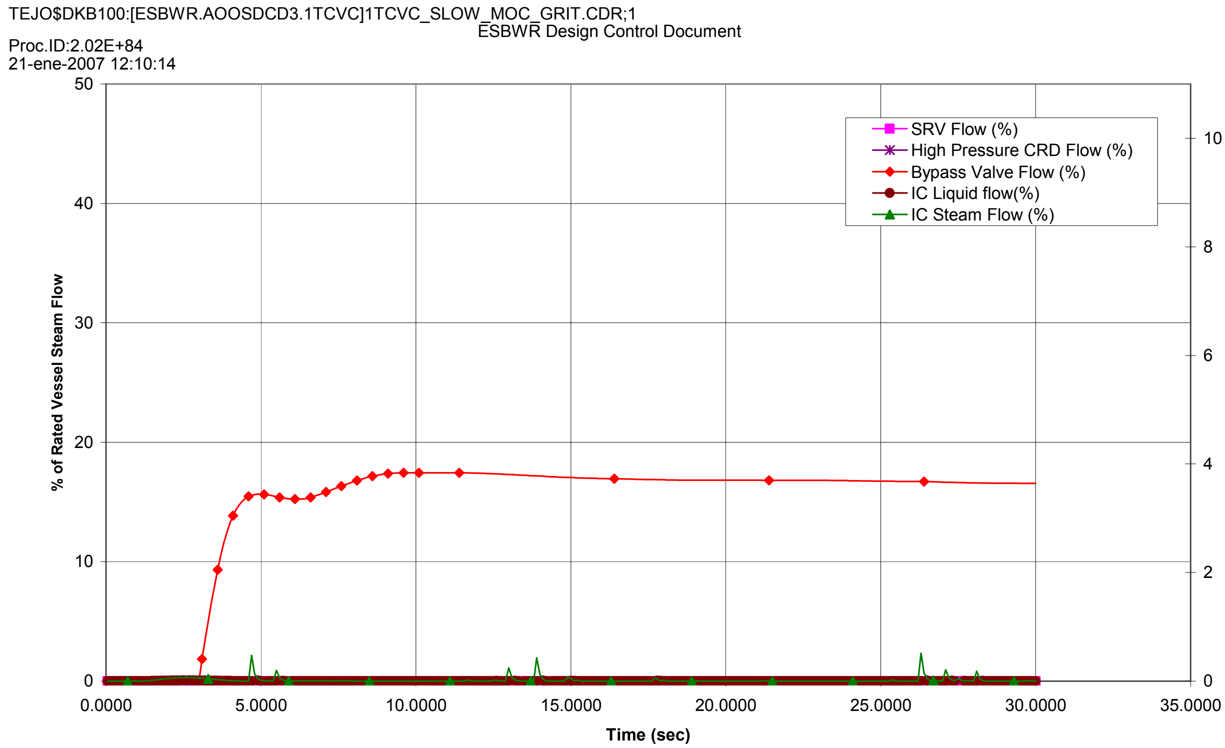


Figure 15.2-3b. Slow Closure of One Turbine Control Valve

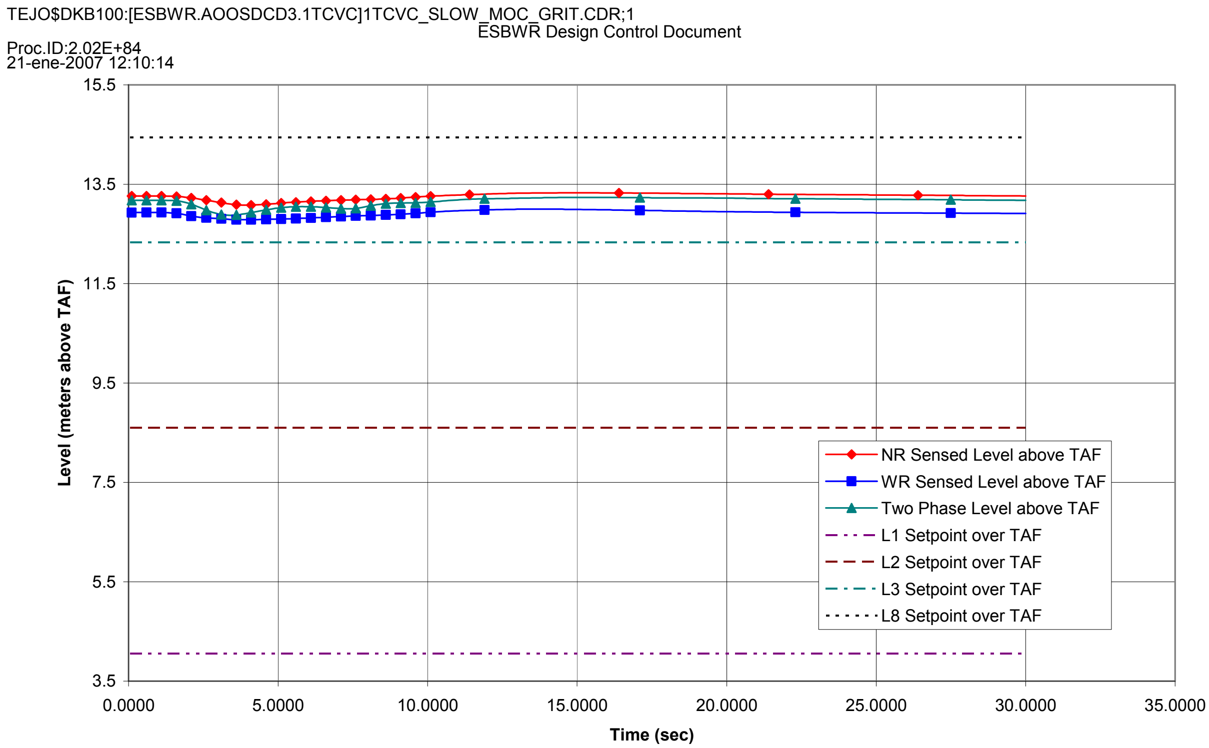


Figure 15.2-3c. Slow Closure of One Turbine Control Valve

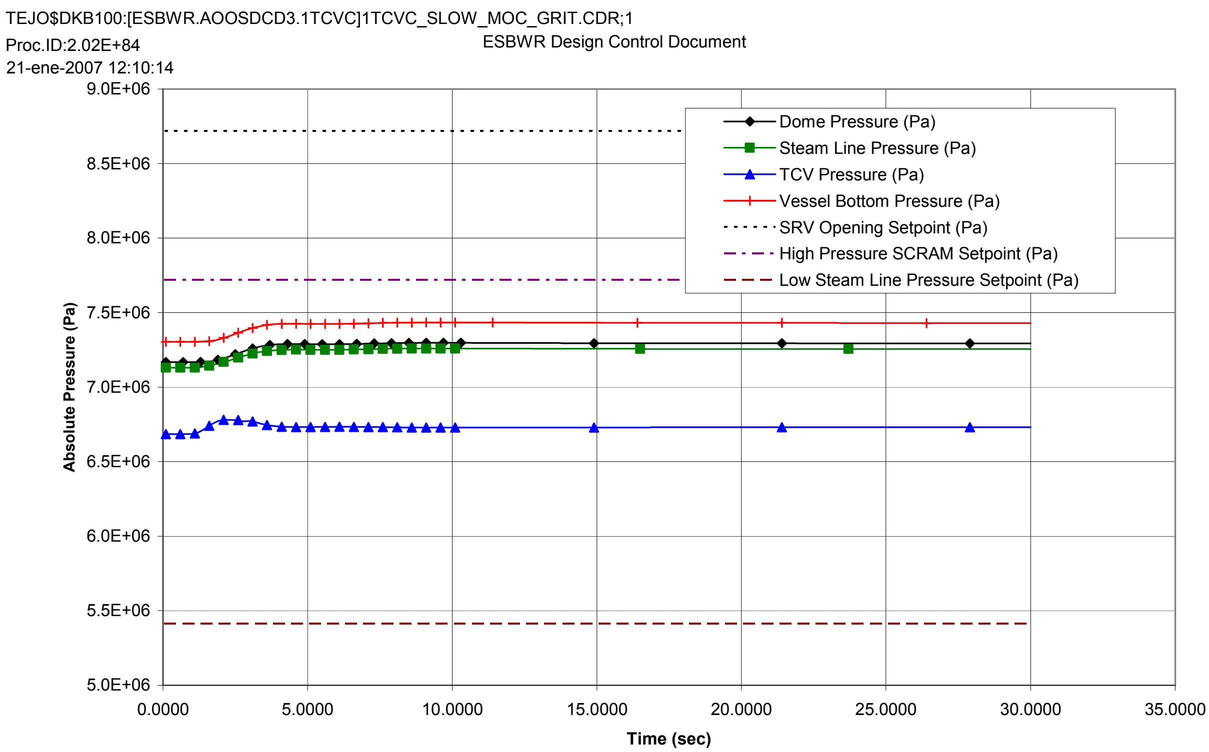


Figure 15.2-3d. Slow Closure of One Turbine Control Valve

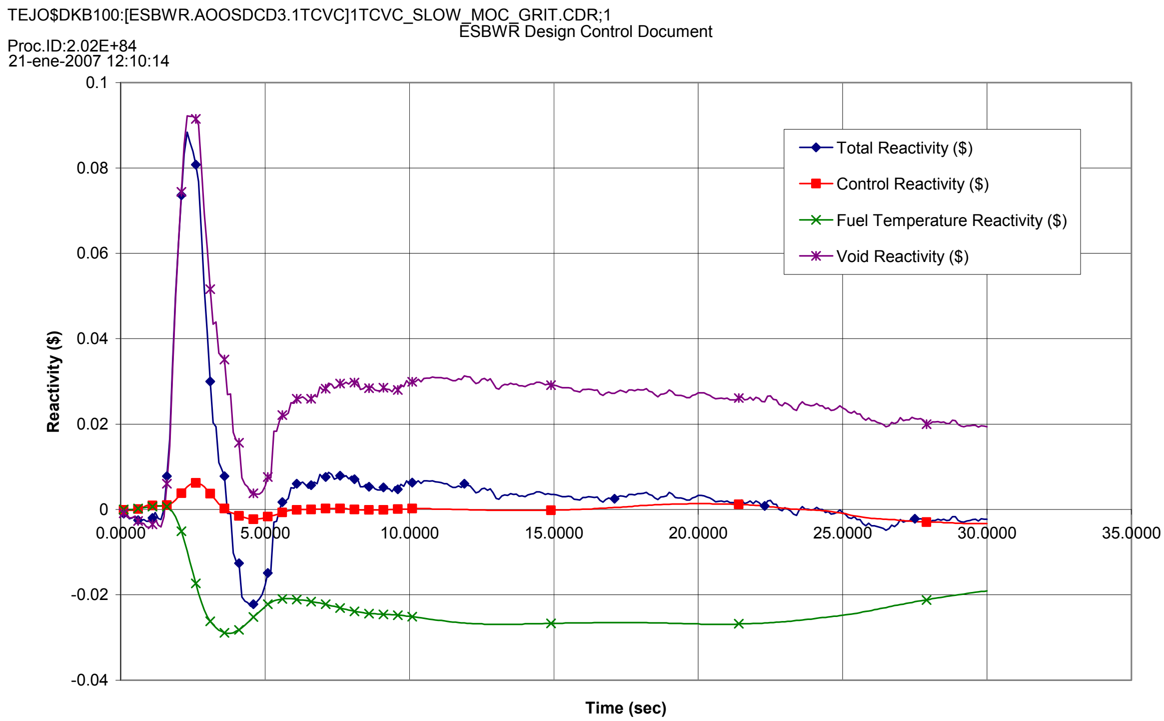


Figure 15.2-3e. Slow Closure of One Turbine Control Valve

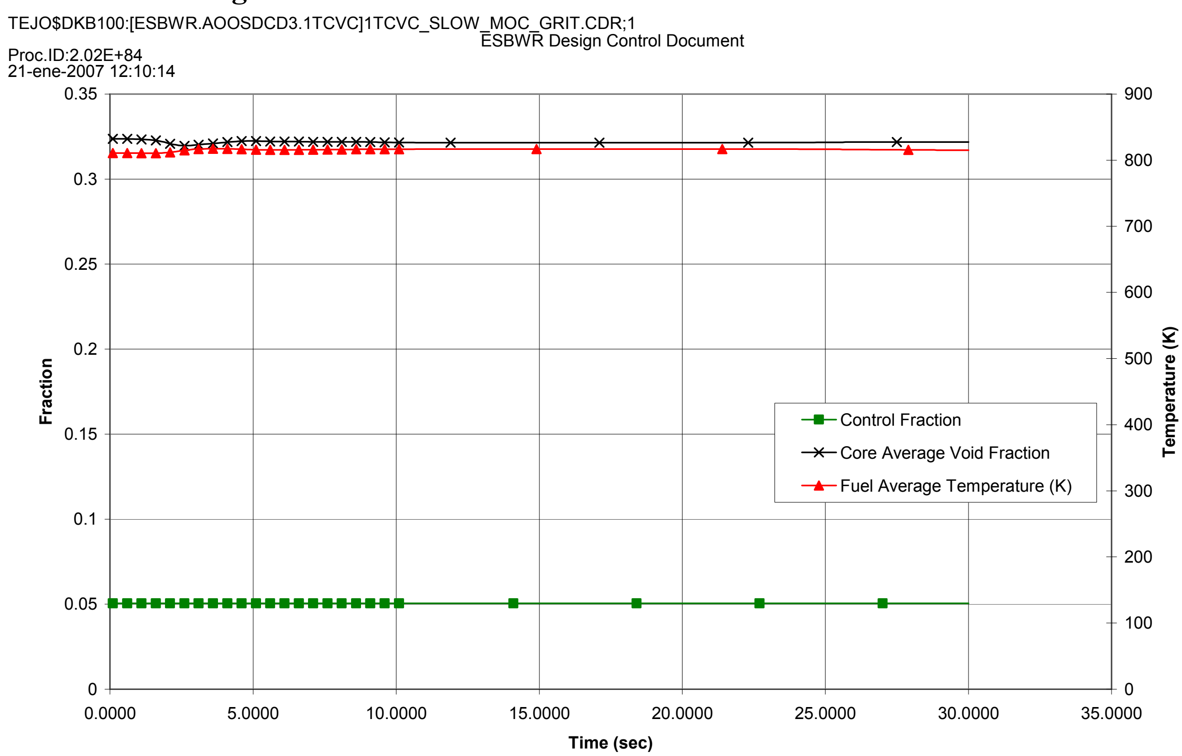


Figure 15.2-3f. Slow Closure of One Turbine Control Valve

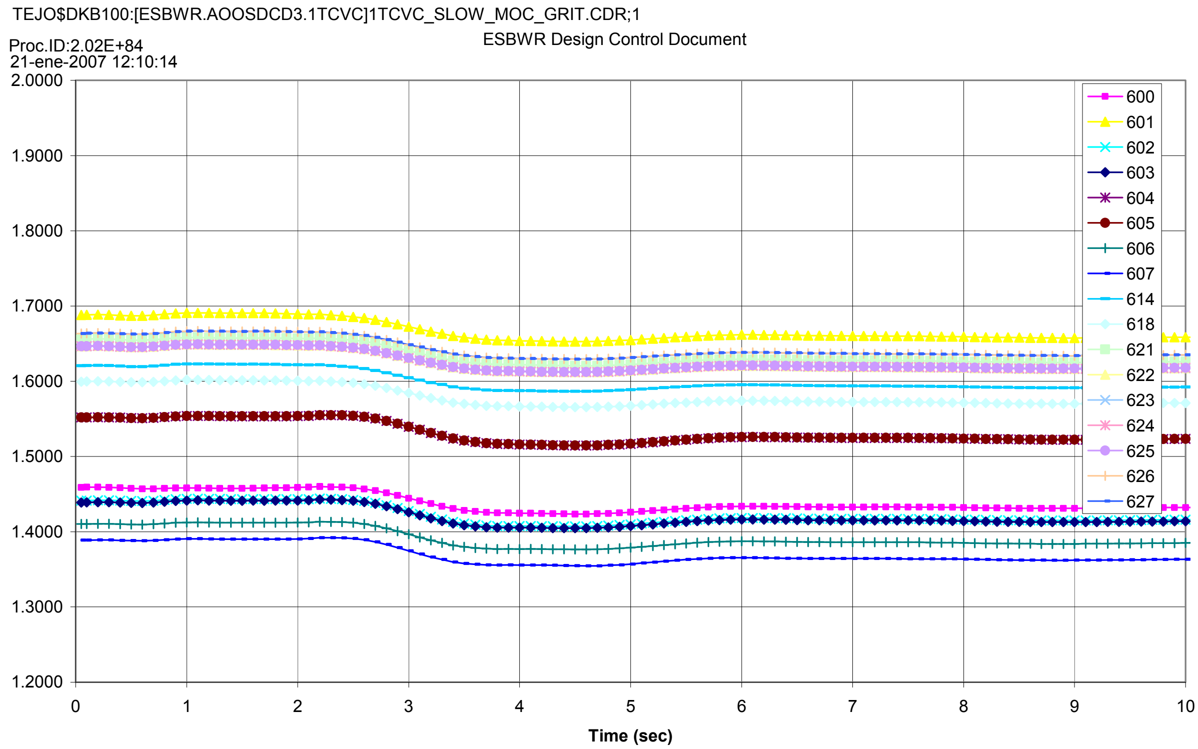


Figure 15.2-3g. Slow Closure of One Turbine Control Valve

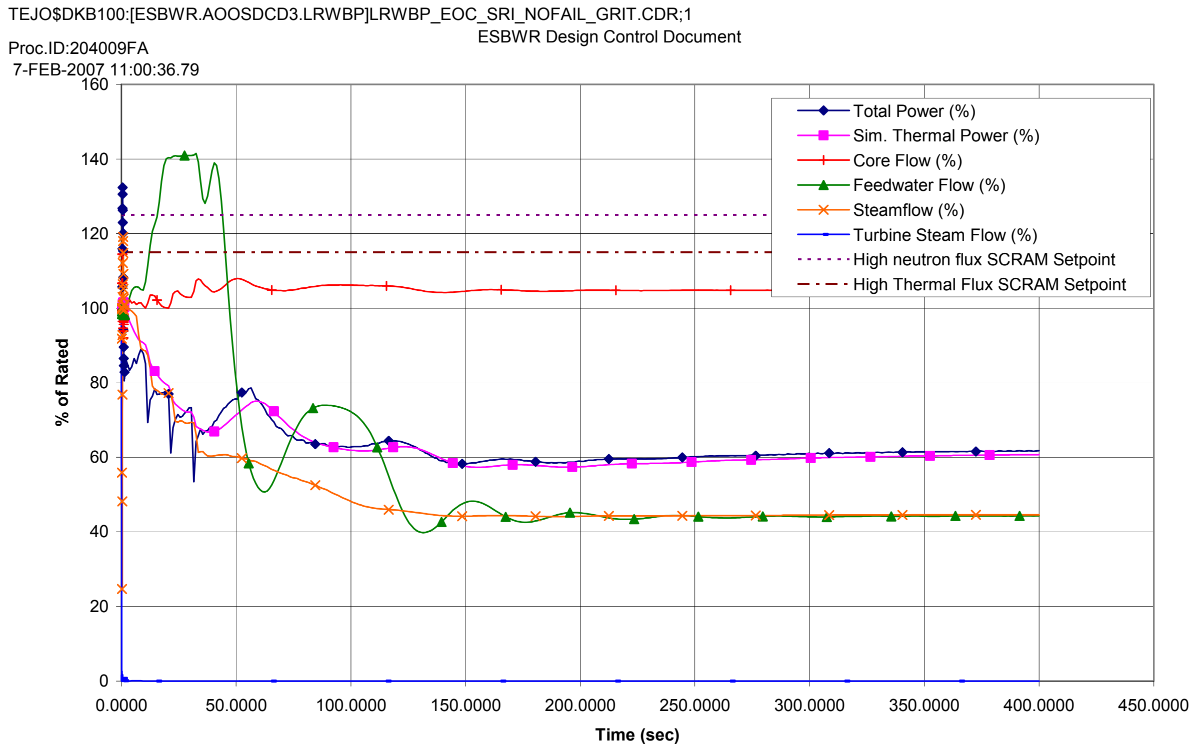


Figure 15.2-4a. Generator Load Rejection with Turbine Bypass

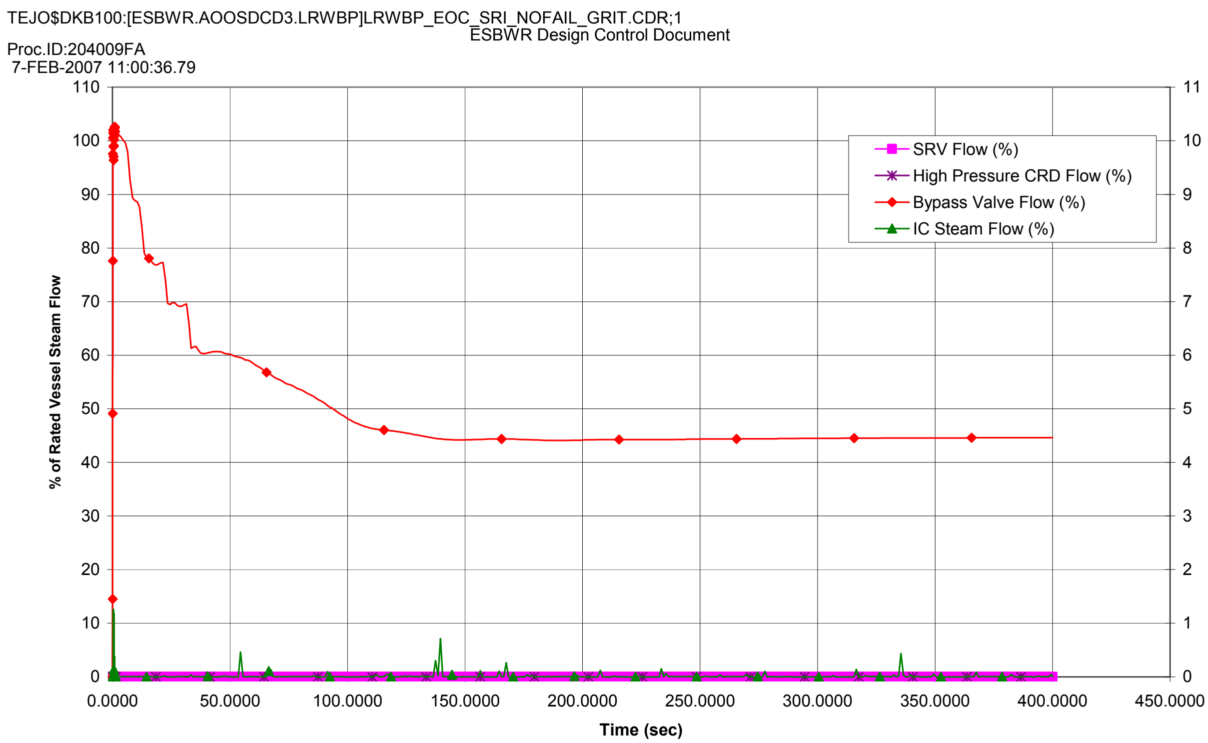


Figure 15.2-4b. Generator Load Rejection with Turbine Bypass

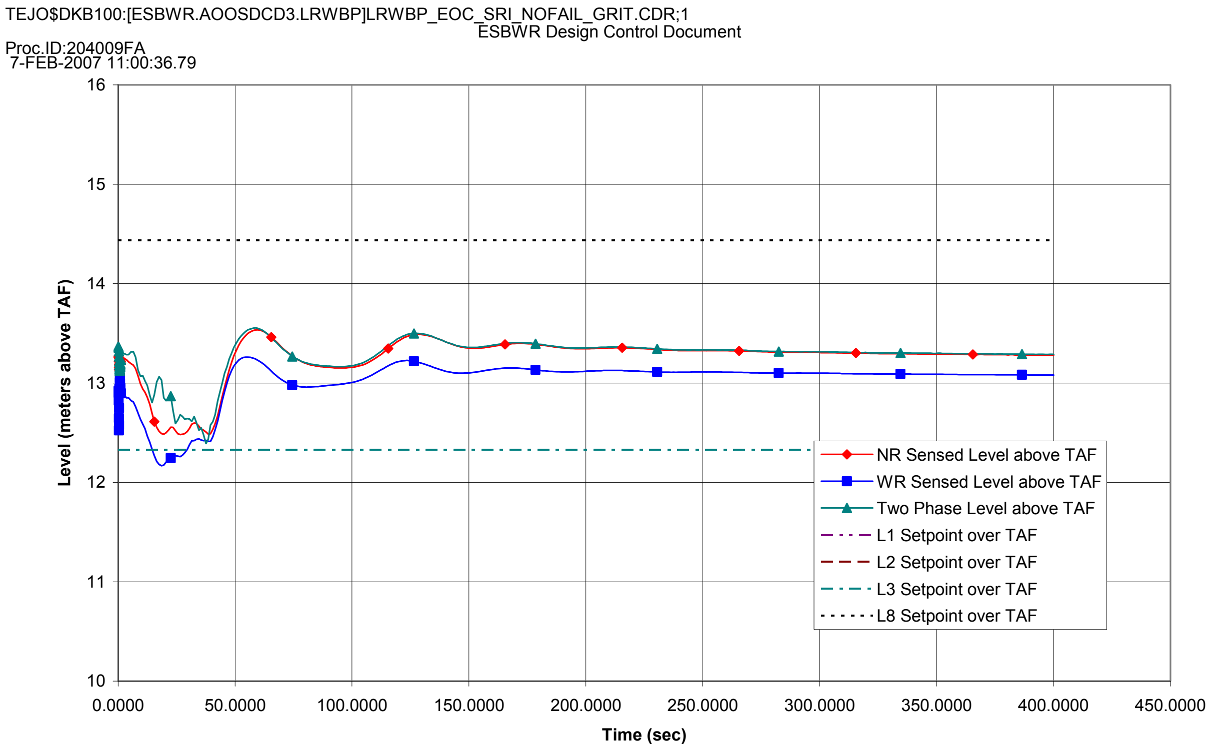


Figure 15.2-4c. Generator Load Rejection with Turbine Bypass

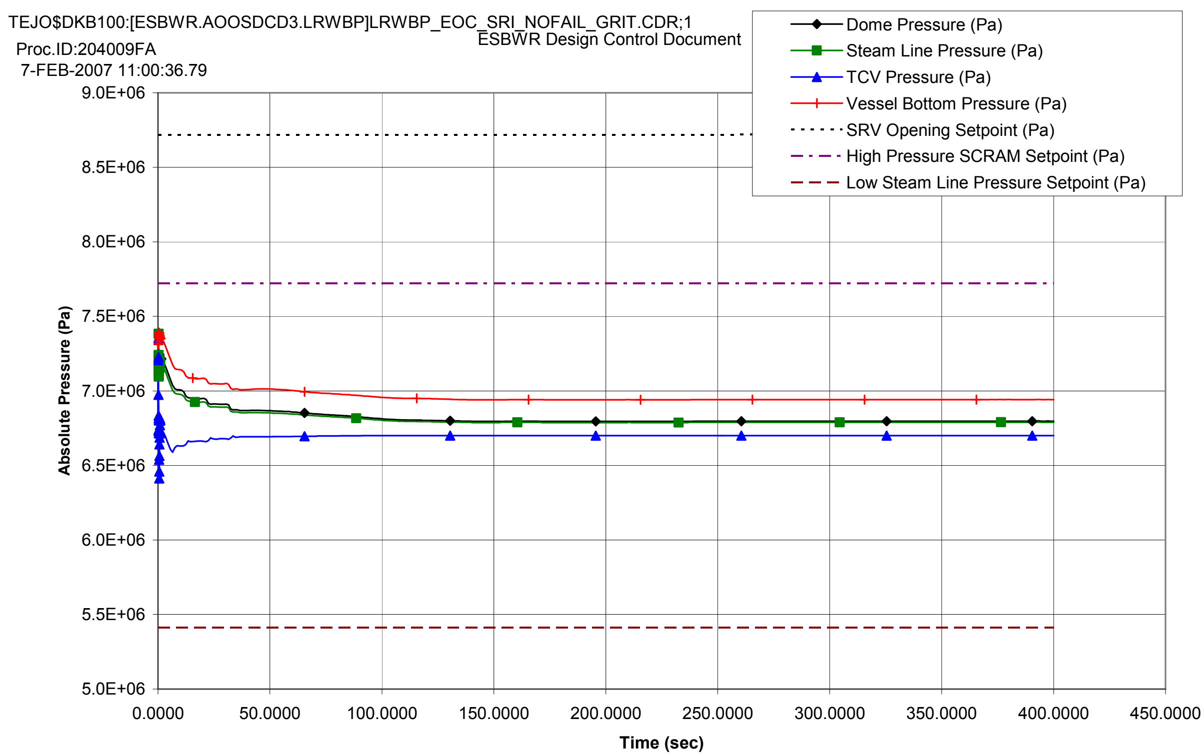


Figure 15.2-4d. Generator Load Rejection with Turbine Bypass

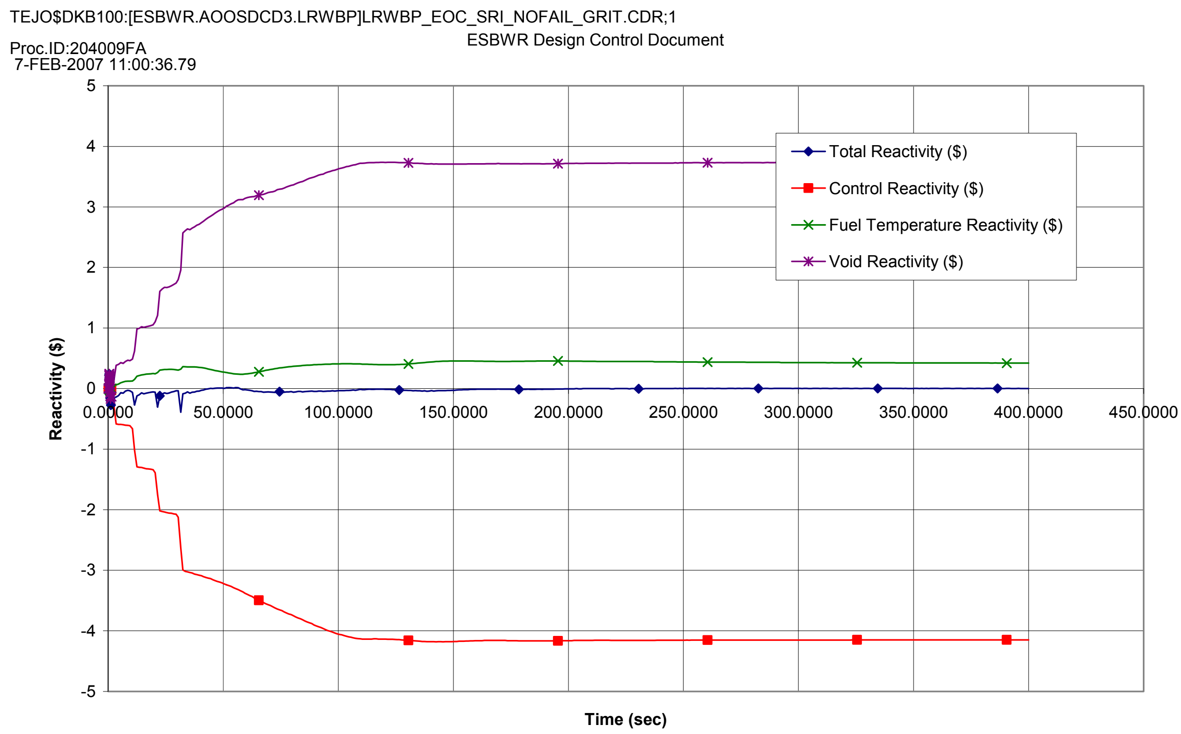


Figure 15.2-4e. Generator Load Rejection with Turbine Bypass

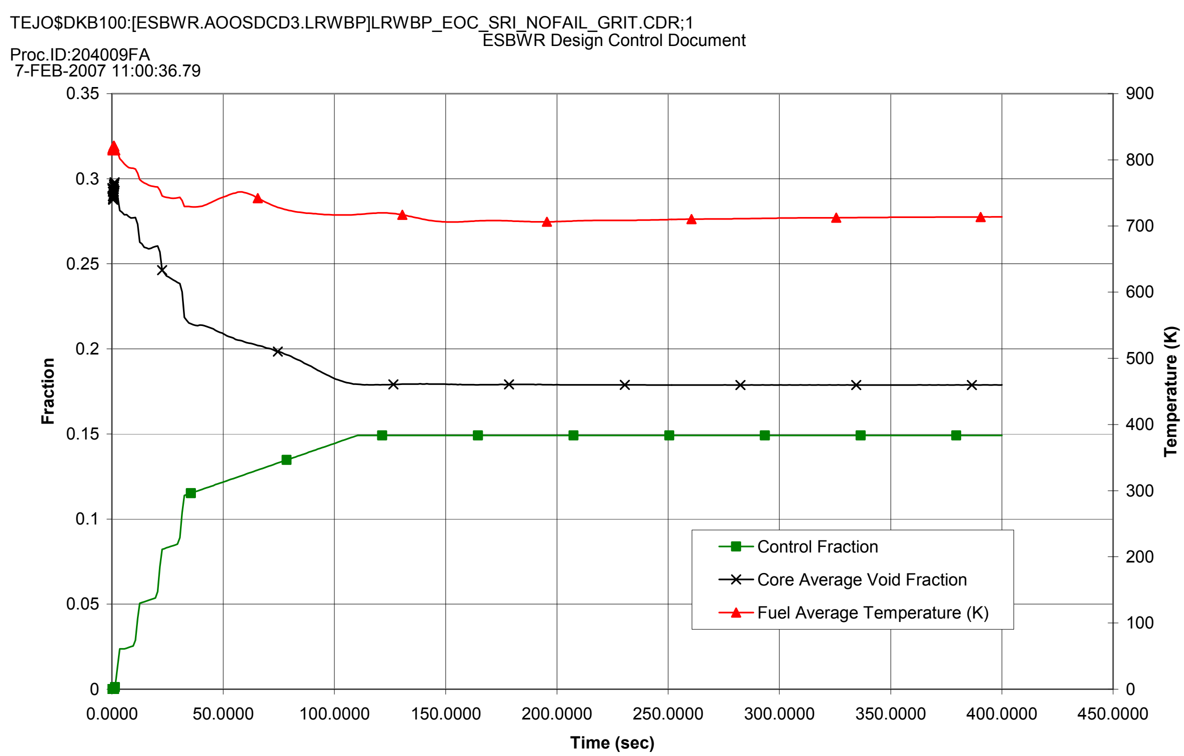


Figure 15.2-4f. Generator Load Rejection with Turbine Bypass

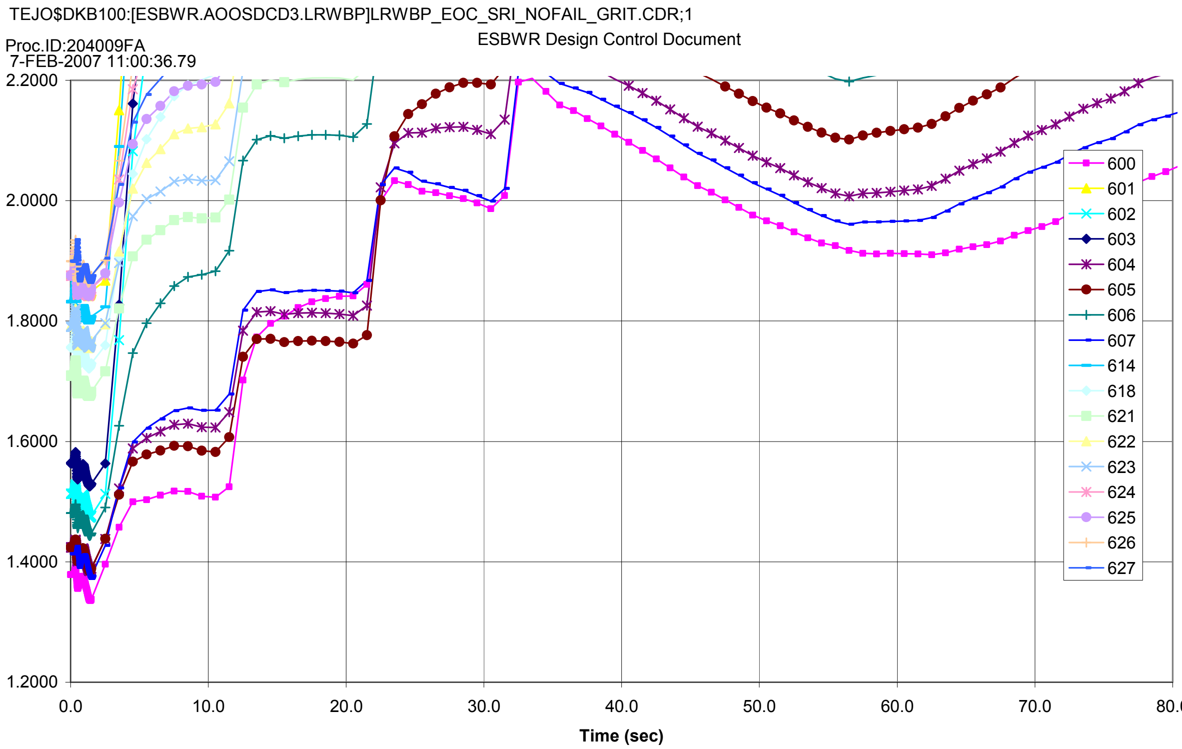


Figure 15.2-4g. Generator Load Rejection with Turbine Bypass

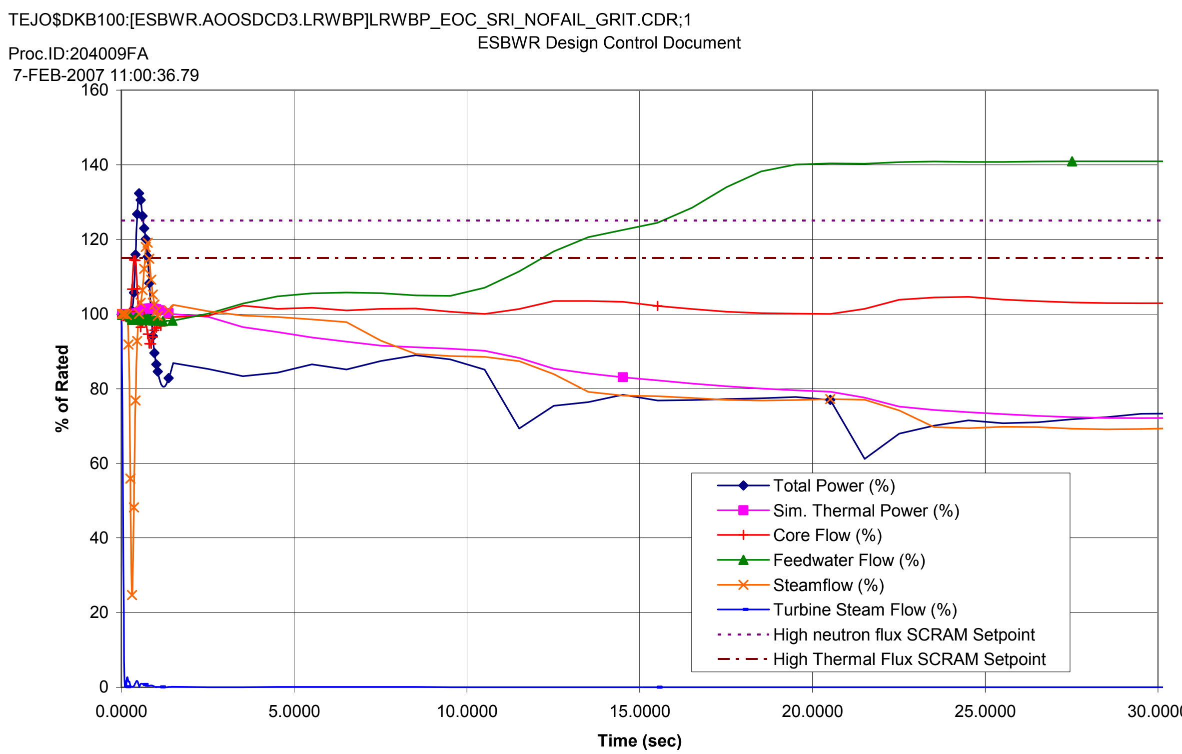


Figure 15.2-4h. Generator Load Rejection with Turbine Bypass
(Figure 15.2-4a from 0 to 30 s)

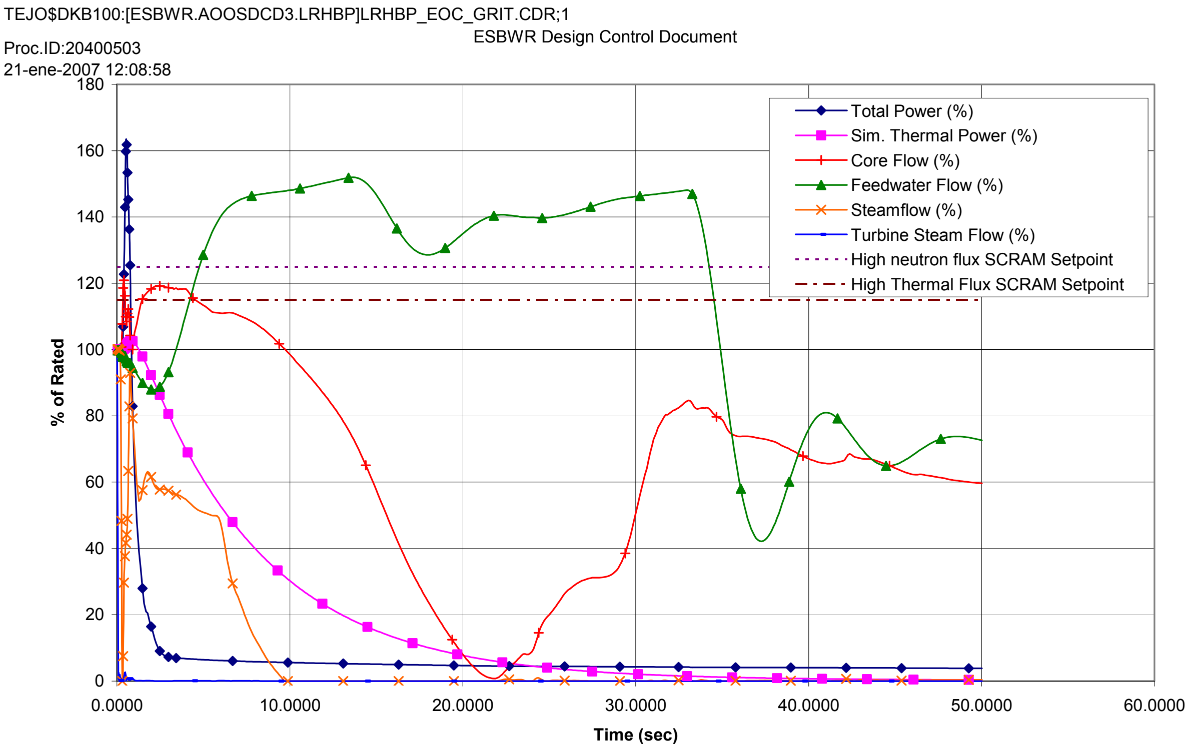


Figure 15.2-5a. Generator Load Rejection with a Single Failure in the Turbine Bypass System

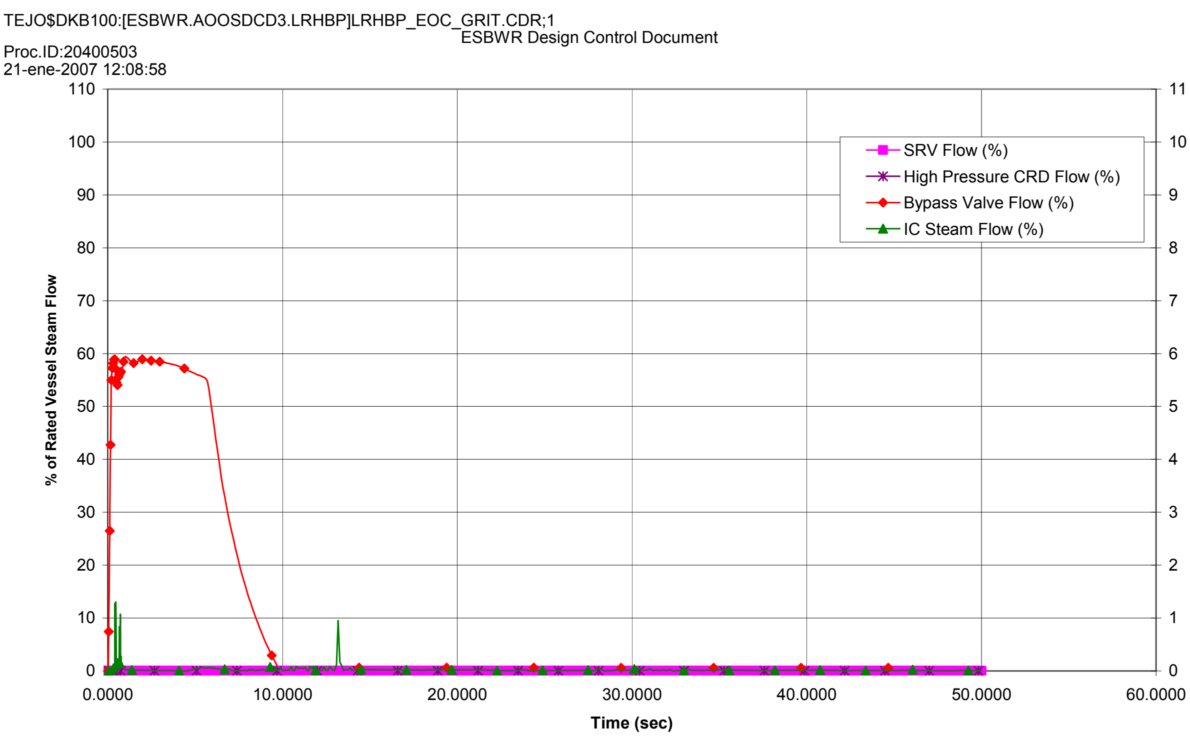


Figure 15.2-5b. Generator Load Rejection with a Single Failure in the Turbine Bypass System

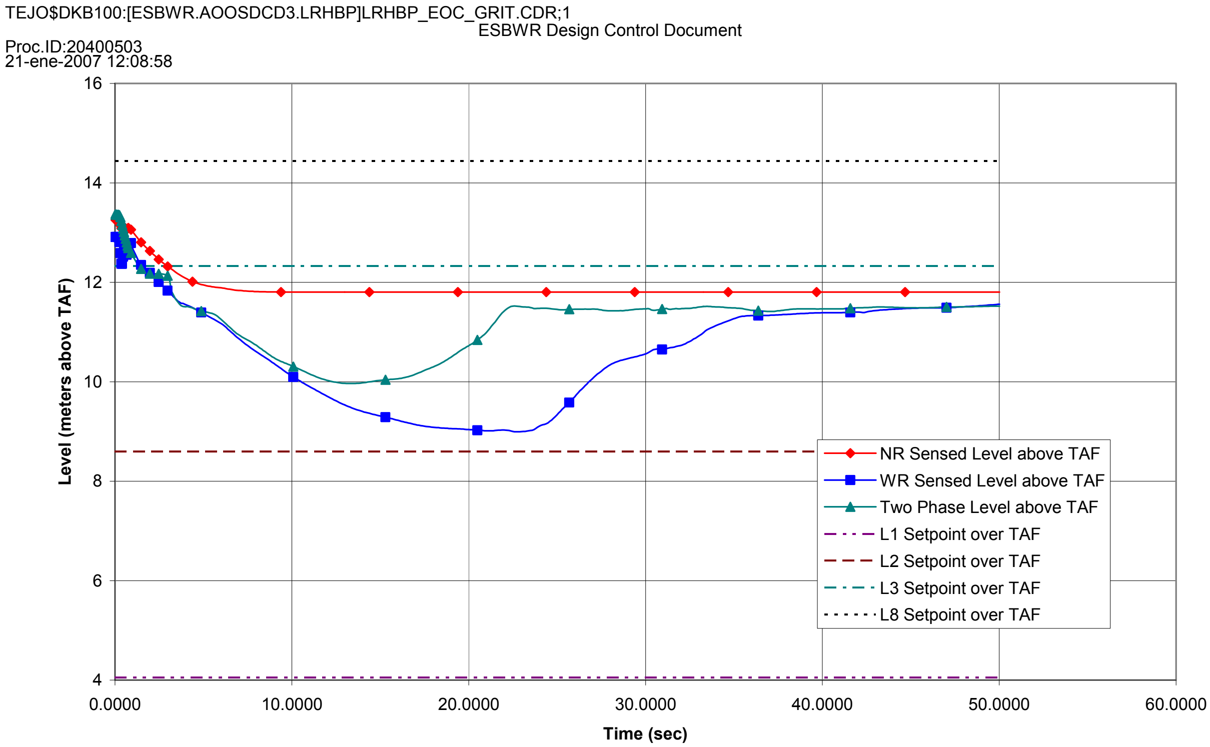


Figure 15.2-5c. Generator Load Rejection with a Single Failure in the Turbine Bypass System

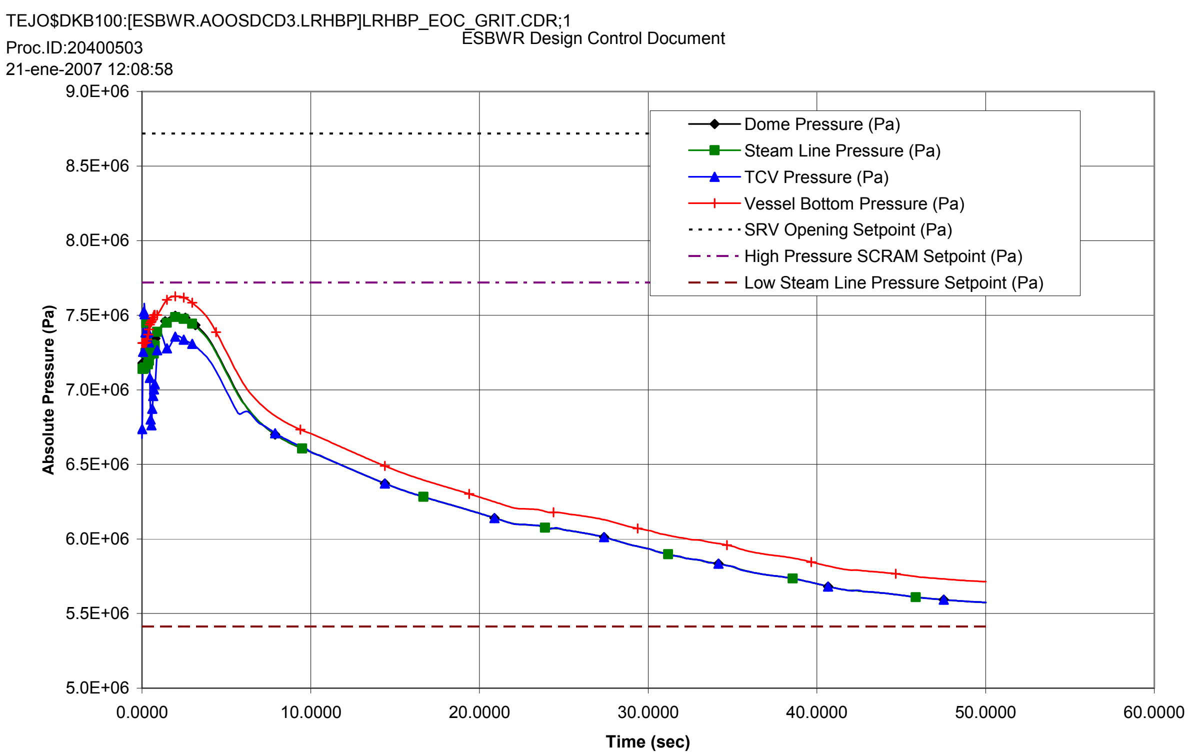


Figure 15.2-5d. Generator Load Rejection with a Single Failure in the Turbine Bypass System

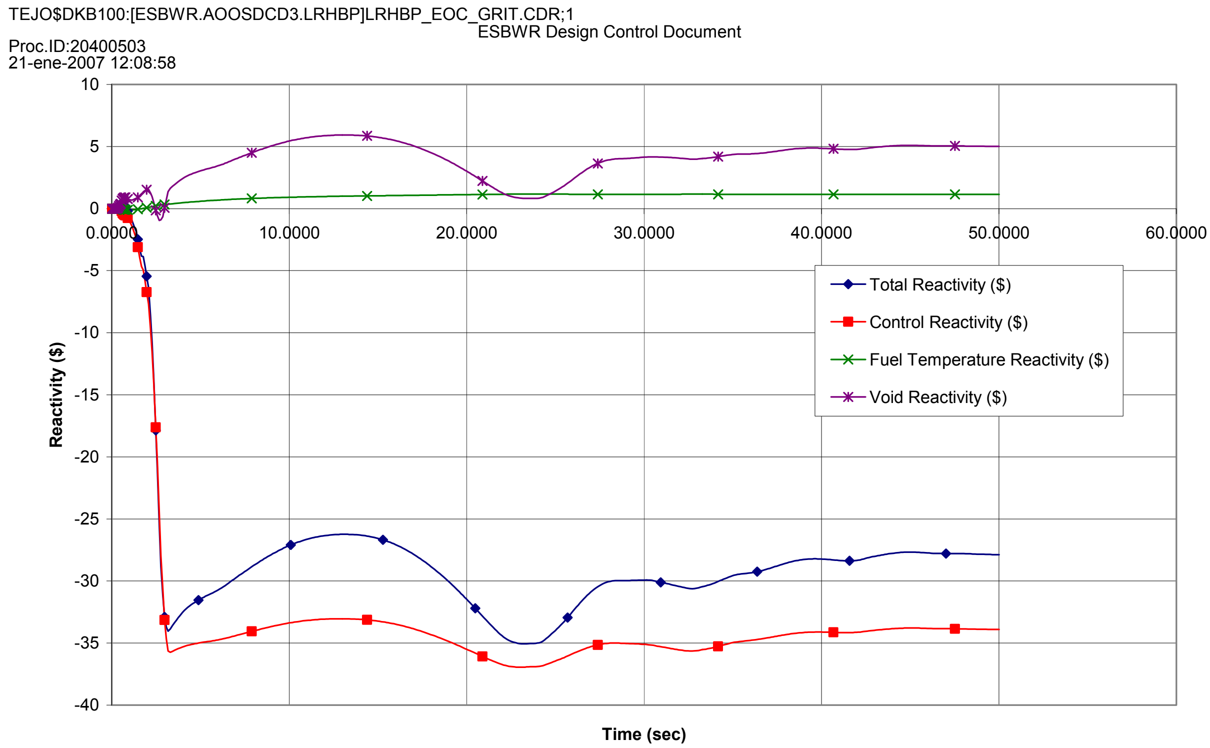


Figure 15.2-5e. Generator Load Rejection with a Single Failure in the Turbine Bypass System

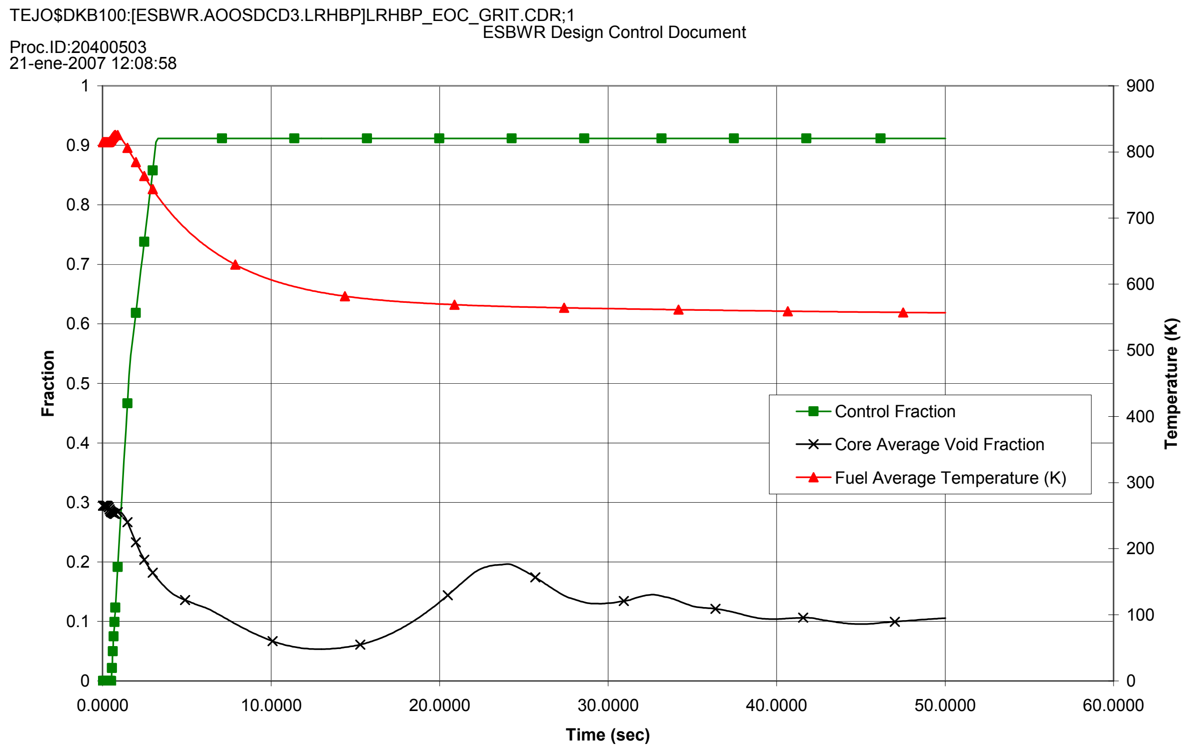


Figure 15.2-5f. Generator Load Rejection with a Single Failure in the Turbine Bypass System

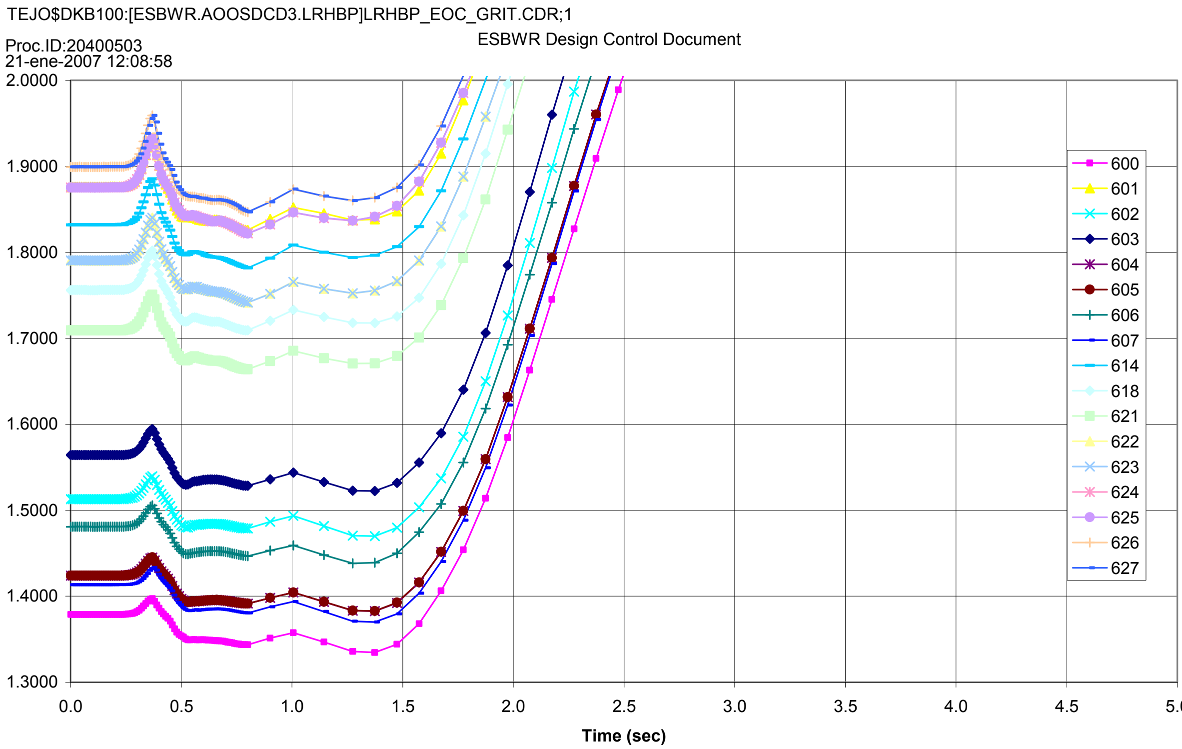


Figure 15.2-5g. Generator Load Rejection with a Single Failure in the Turbine Bypass System

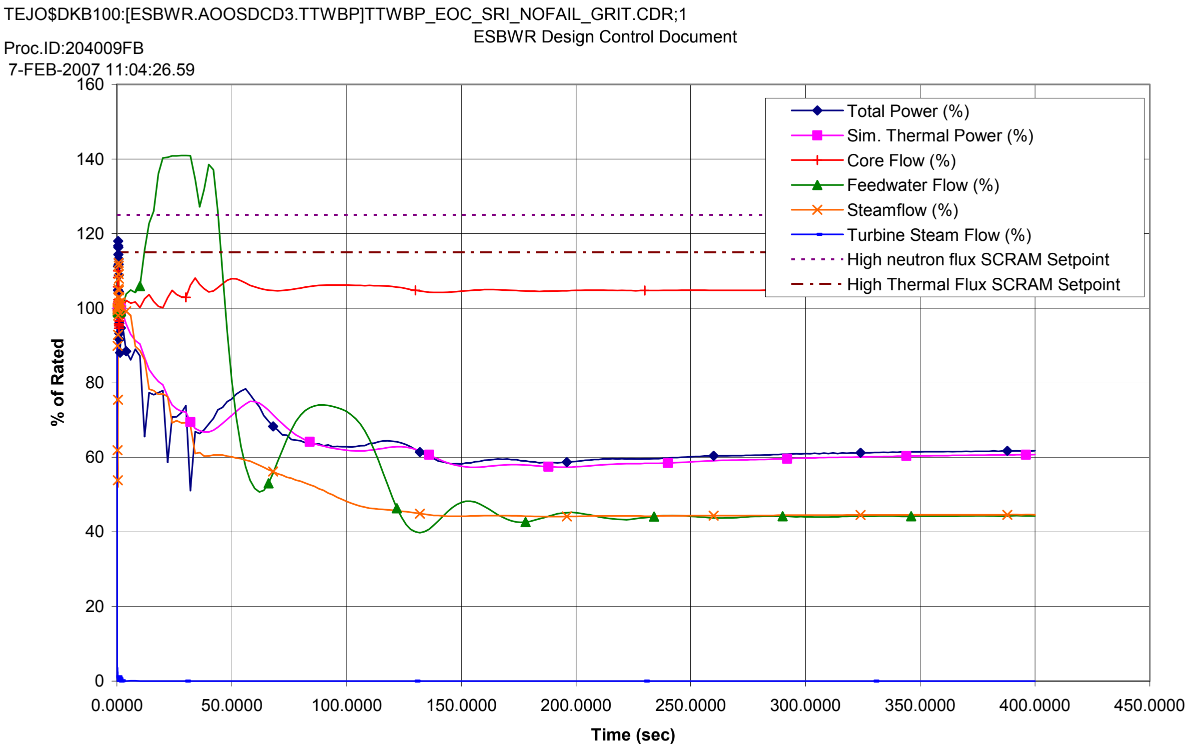


Figure 15.2-6a. Turbine Trip with Turbine Bypass

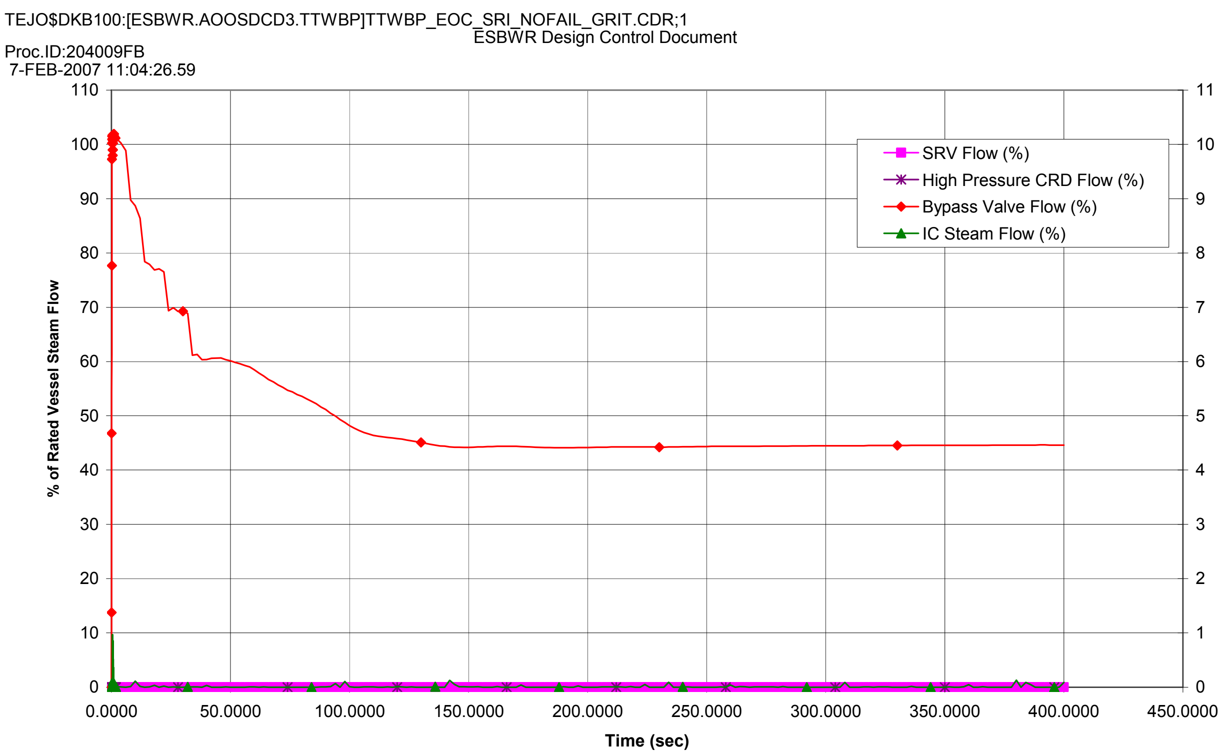


Figure 15.2-6b. Turbine Trip with Turbine Bypass

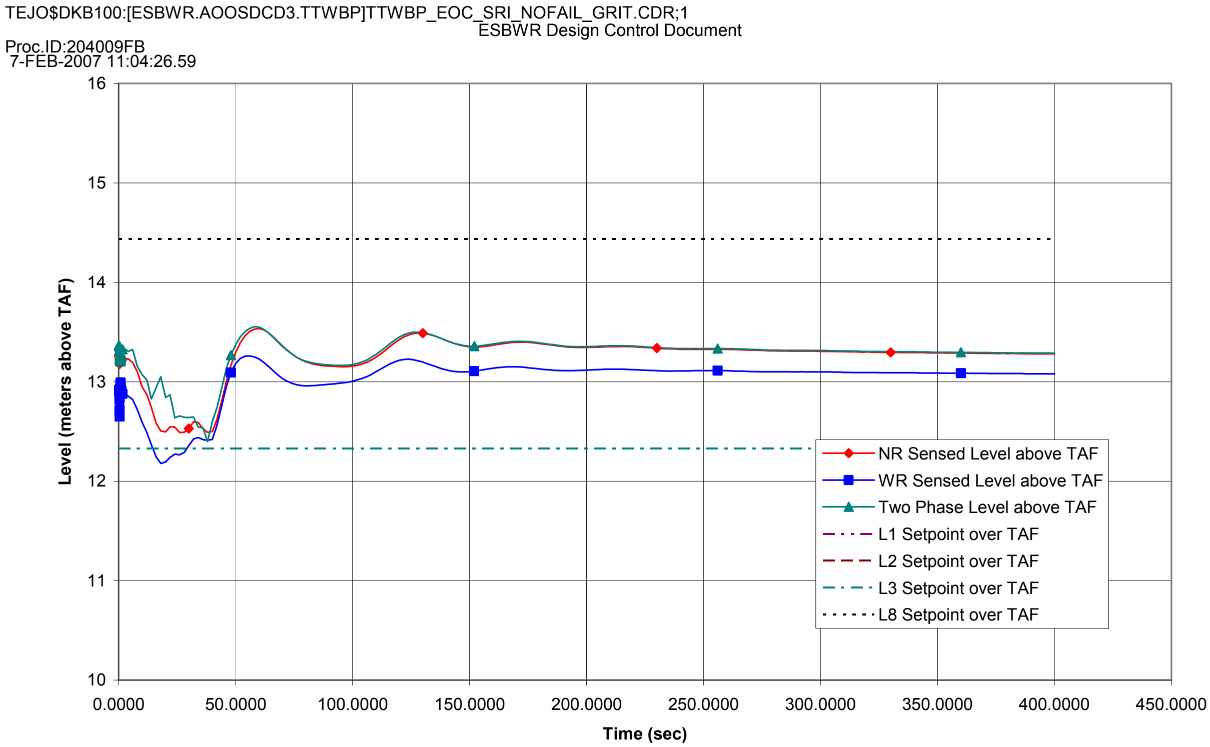


Figure 15.2-6c. Turbine Trip with Turbine Bypass

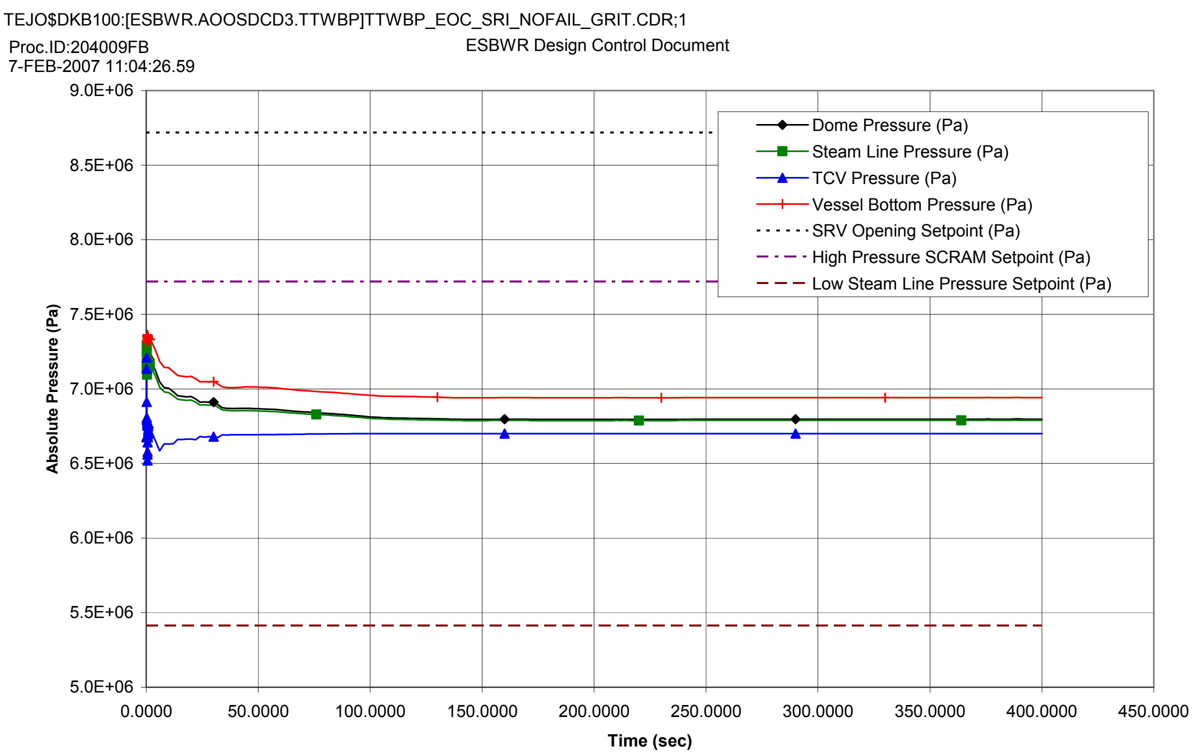


Figure 15.2-6d. Turbine Trip with Turbine Bypass

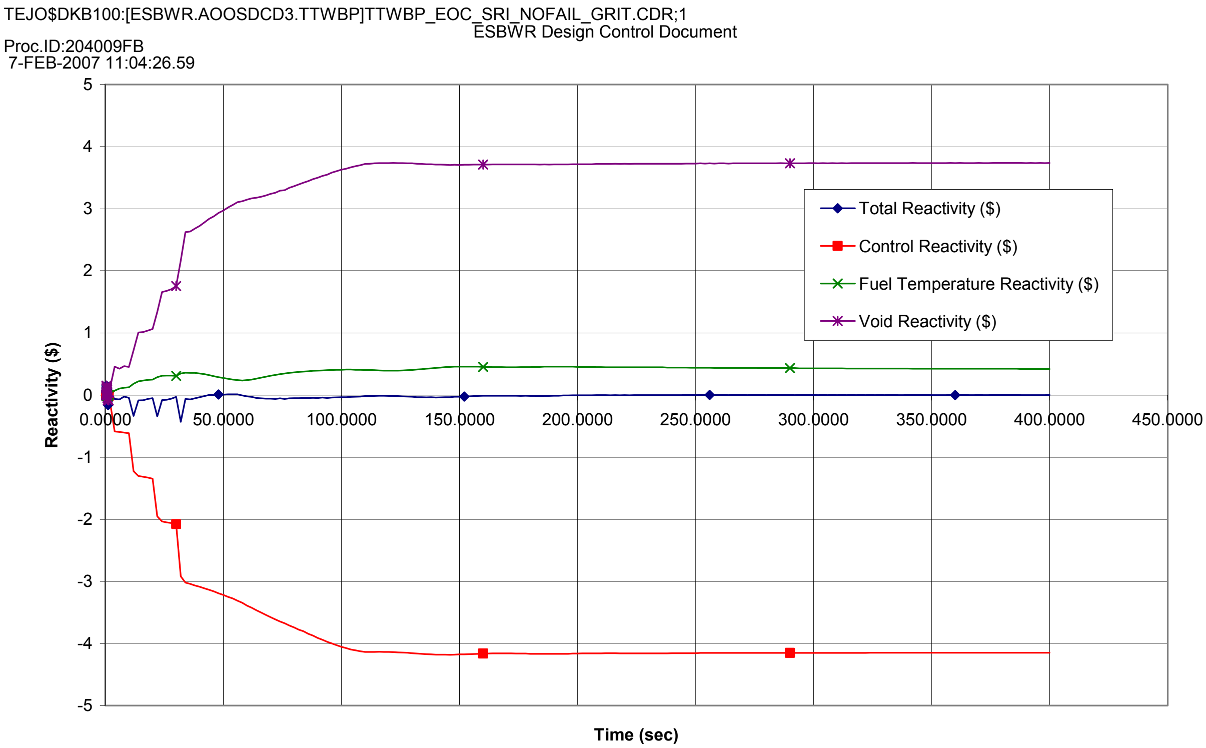


Figure 15.2-6e. Turbine Trip with Turbine Bypass

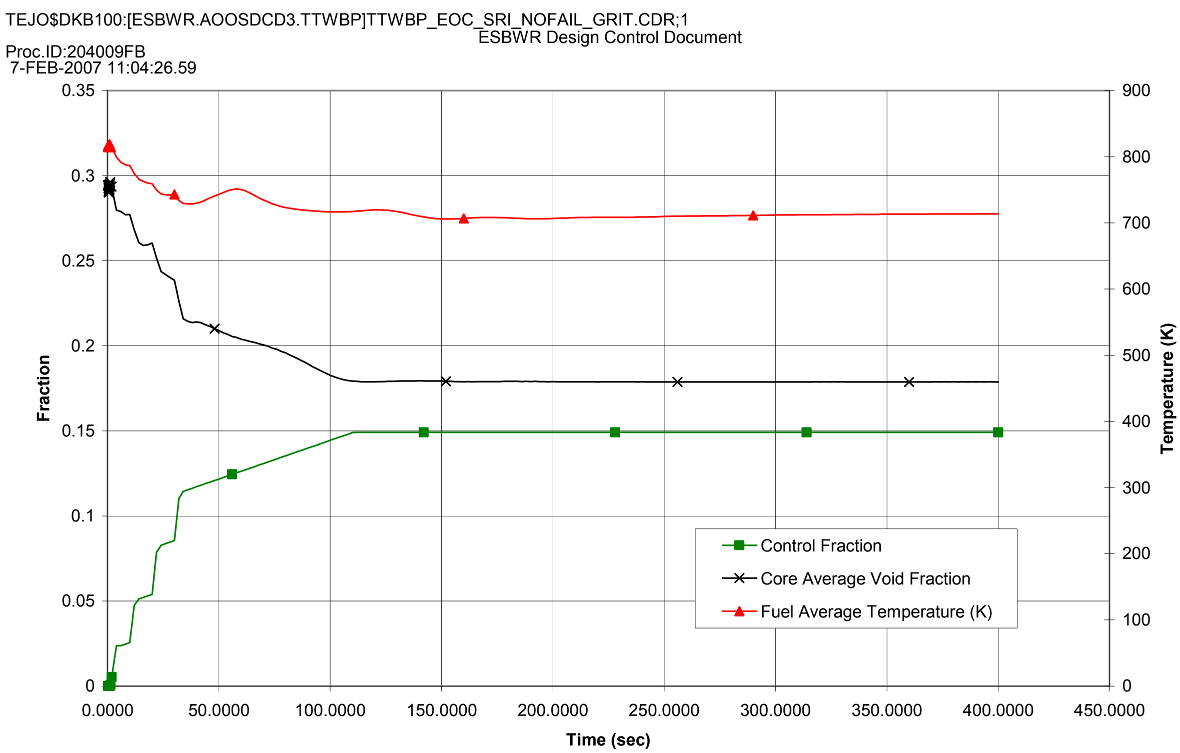


Figure 15.2-6f. Turbine Trip with Turbine Bypass

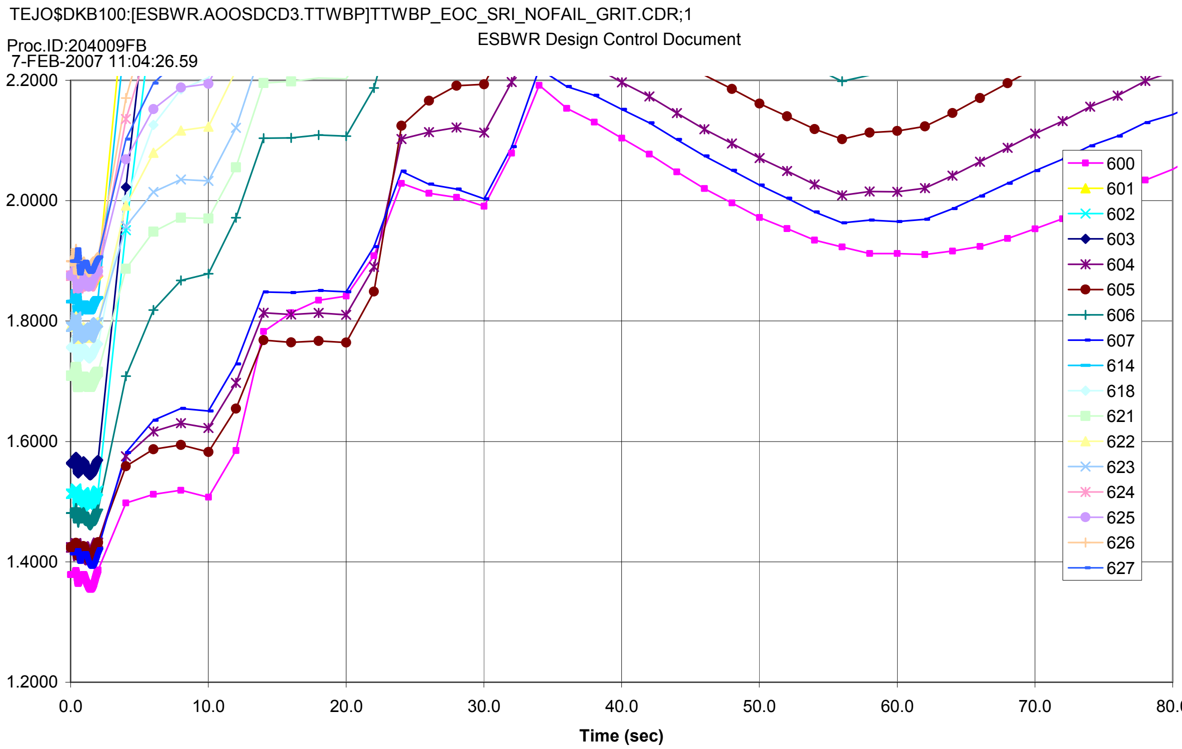


Figure 15.2-6g. Turbine Trip with Turbine Bypass

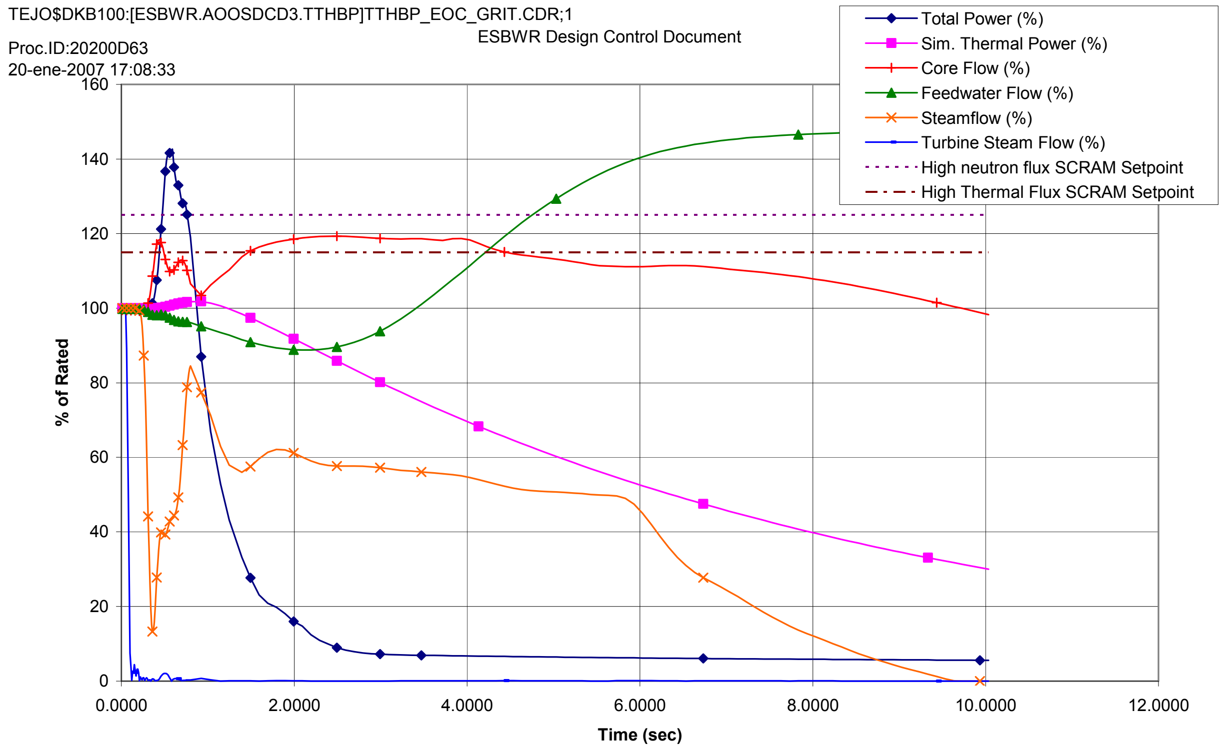


Figure 15.2-7a. Turbine Trip with a Single Failure in the Turbine Bypass System

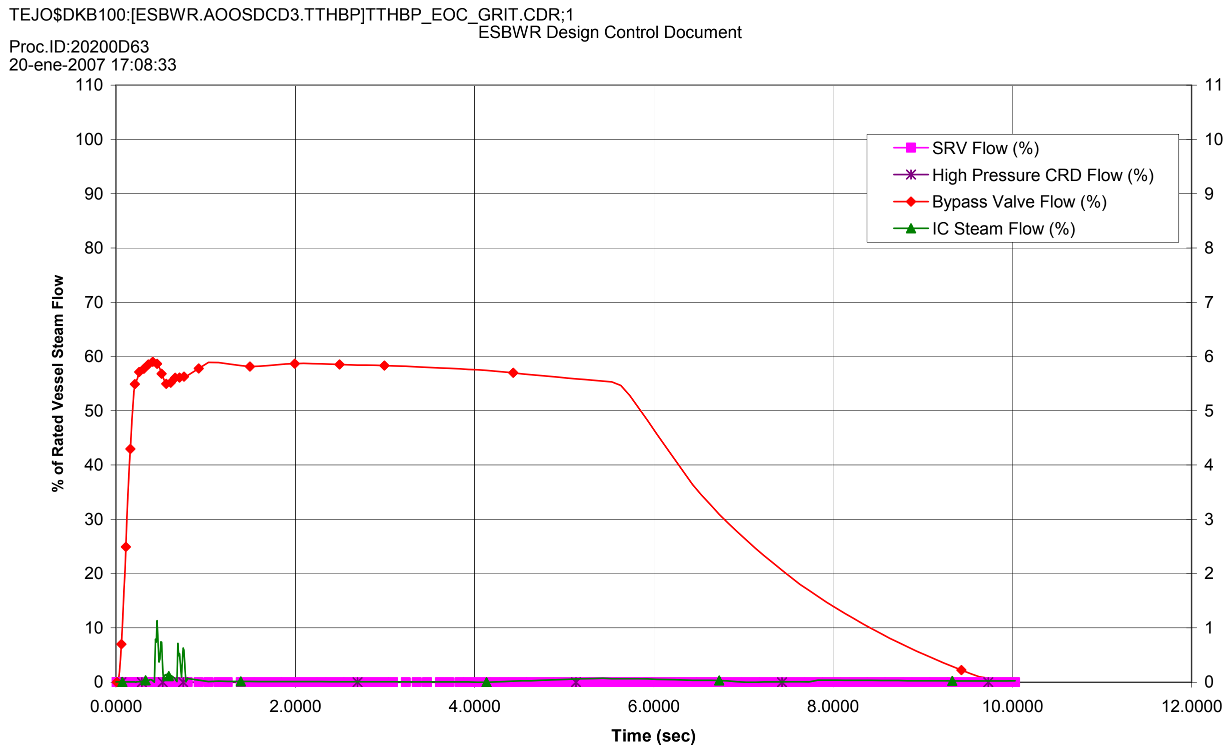


Figure 15.2-7b. Turbine Trip with a Single Failure in the Turbine Bypass System

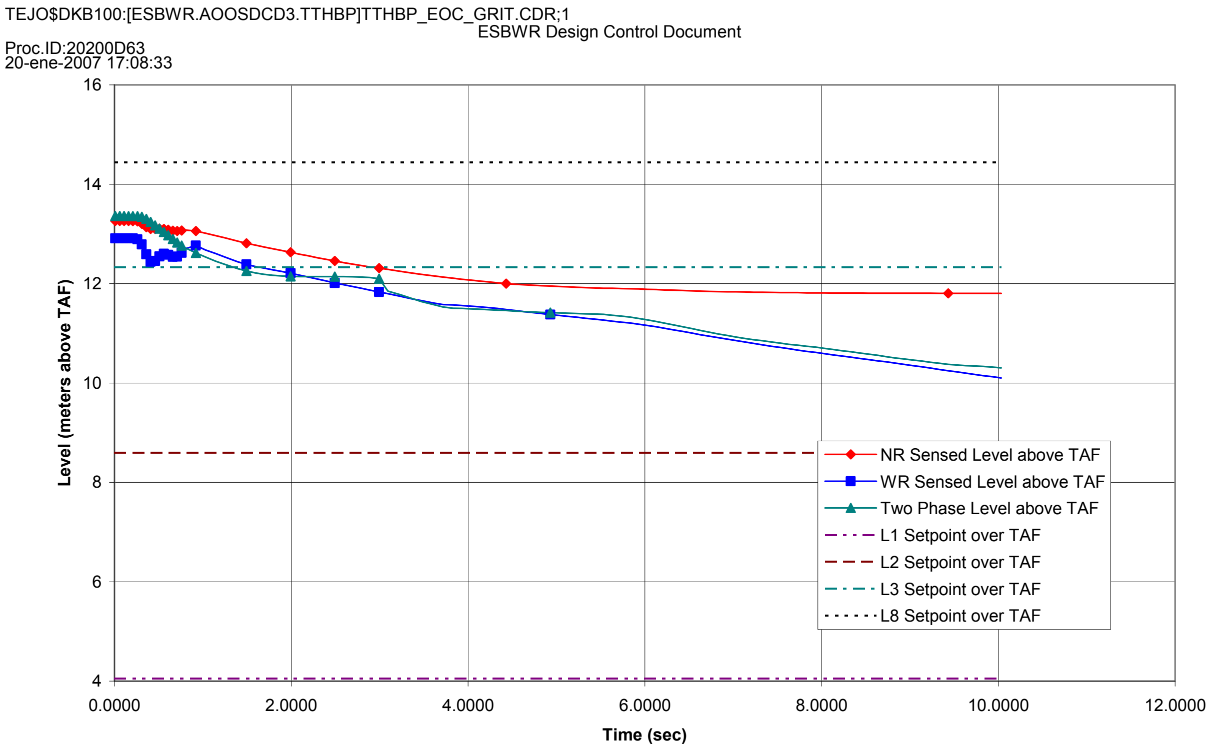


Figure 15.2-7c. Turbine Trip with a Single Failure in the Turbine Bypass System

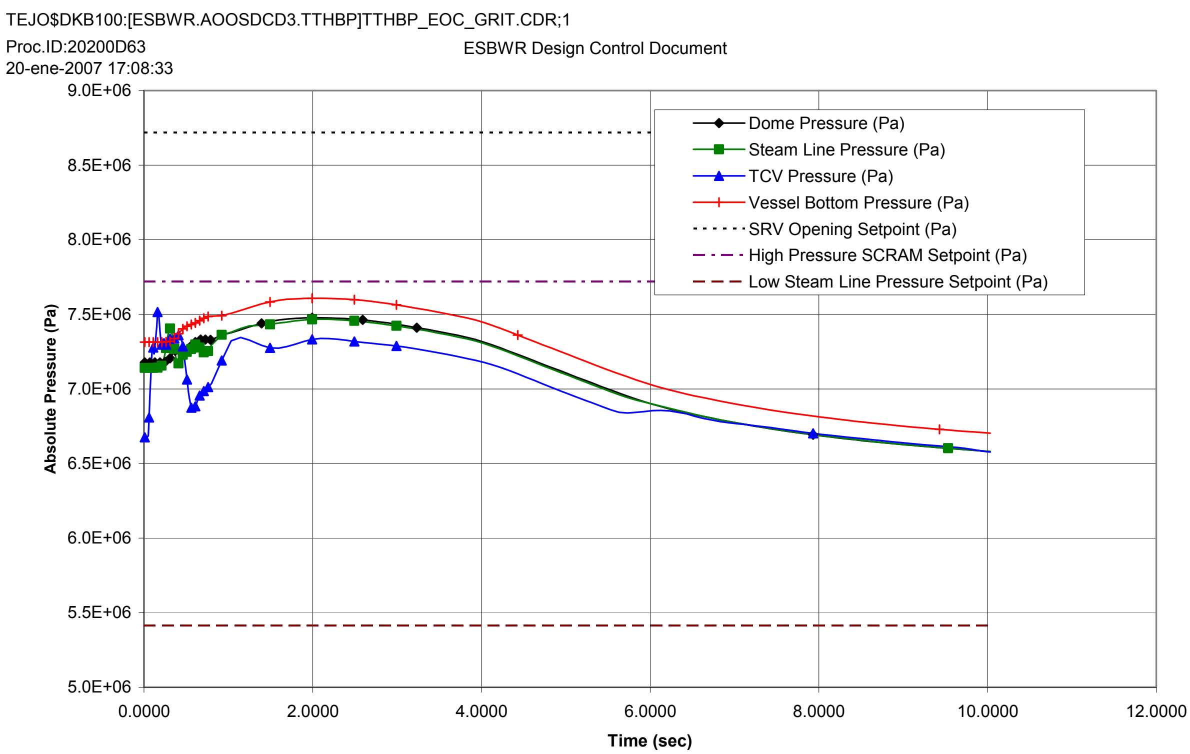


Figure 15.2-7d. Turbine Trip with a Single Failure in the Turbine Bypass System

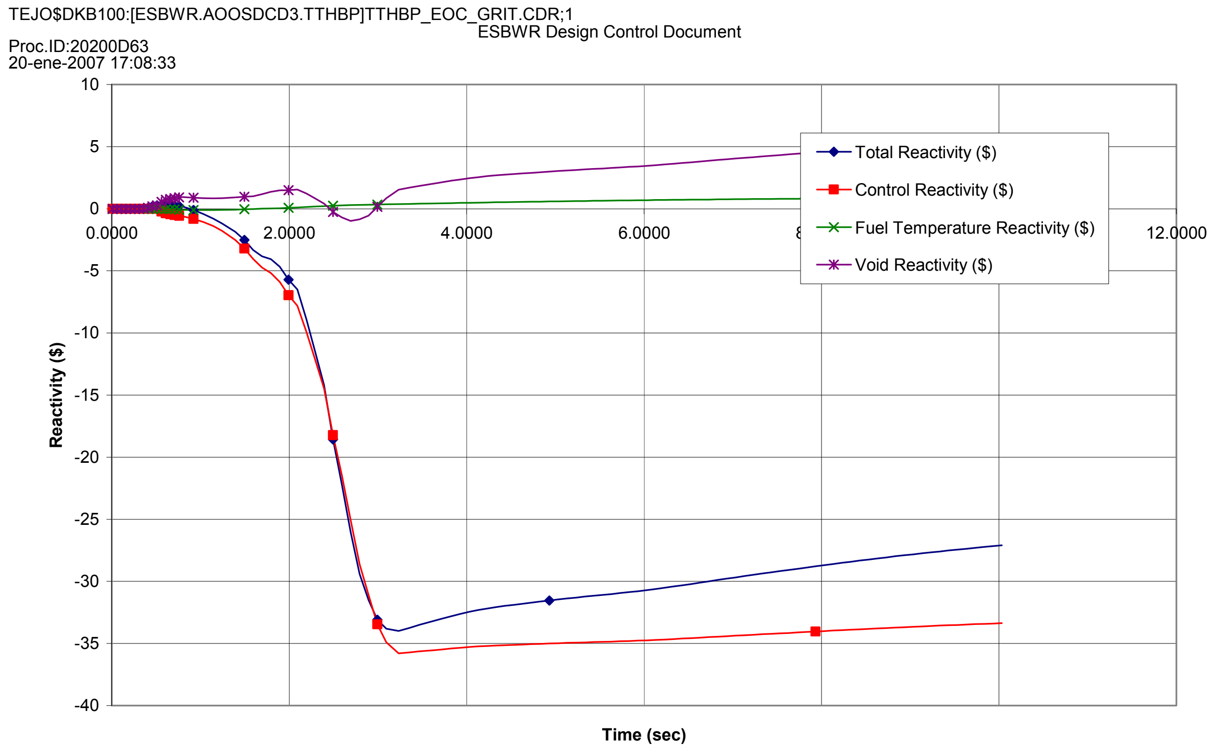


Figure 15.2-7e. Turbine Trip with a Single Failure in the Turbine Bypass System

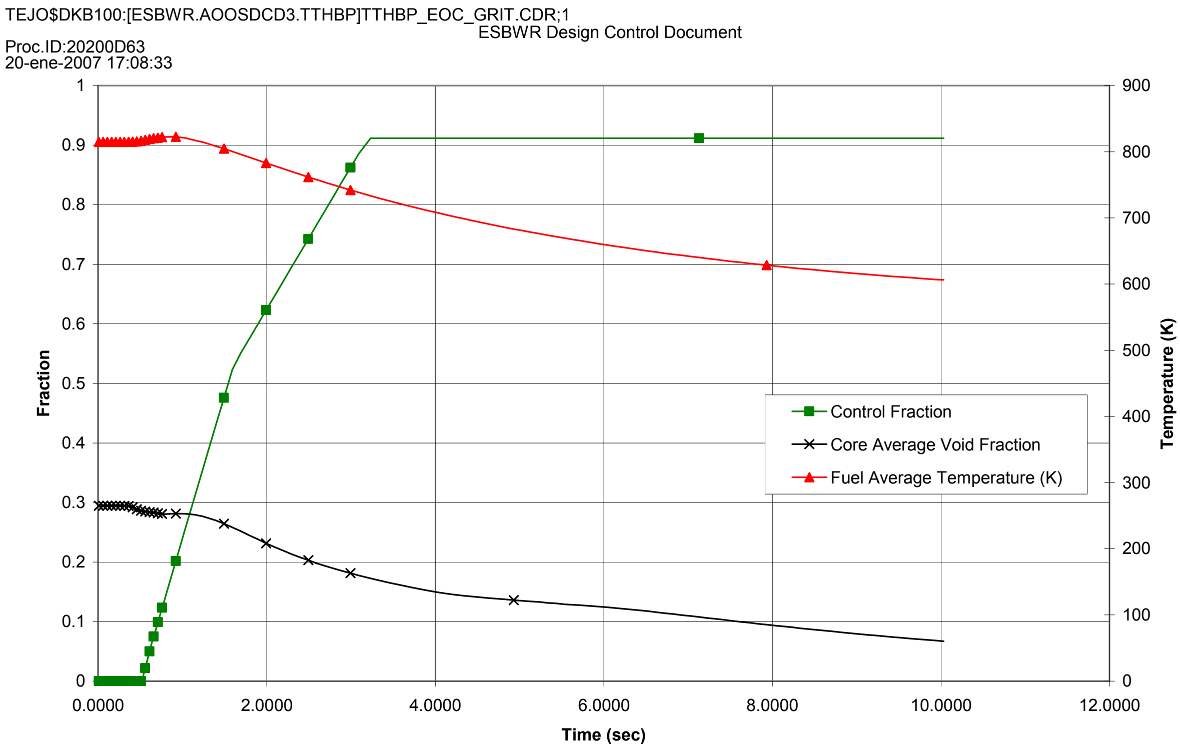


Figure 15.2-7f. Turbine Trip with a Single Failure in the Turbine Bypass System

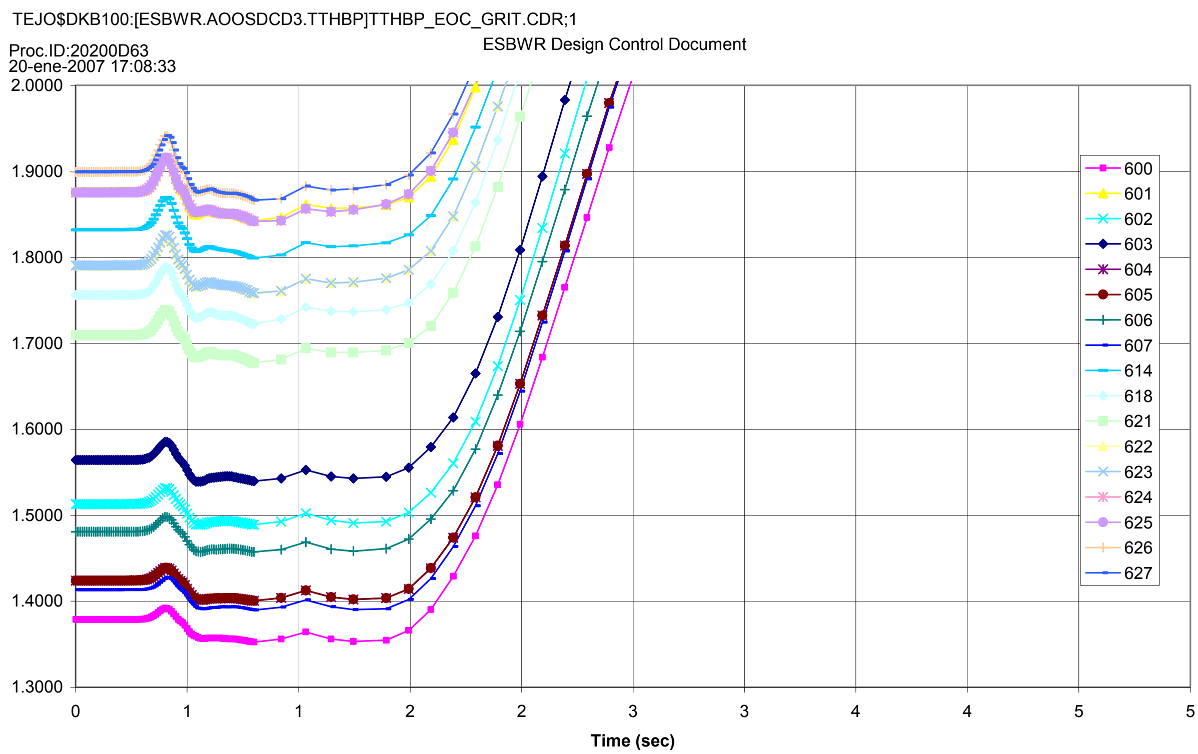


Figure 15.2-7g. Turbine Trip with a Single Failure in the Turbine Bypass System

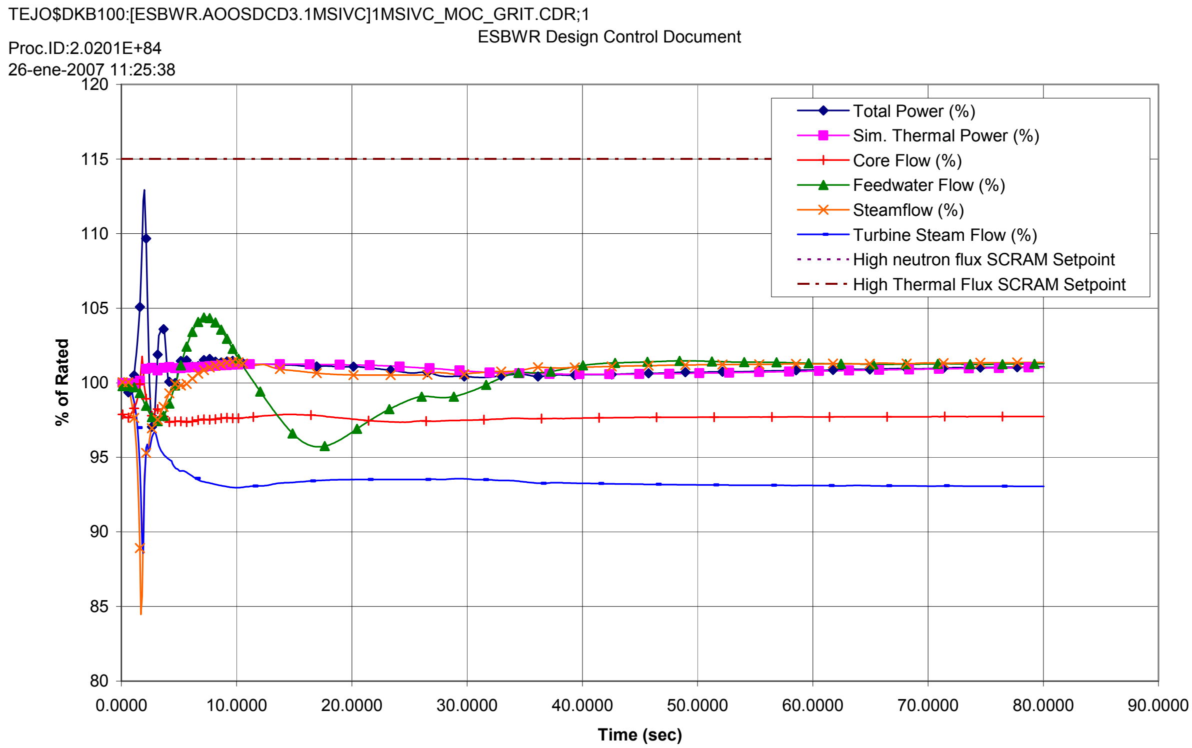


Figure 15.2-8a. One MSIV Closure

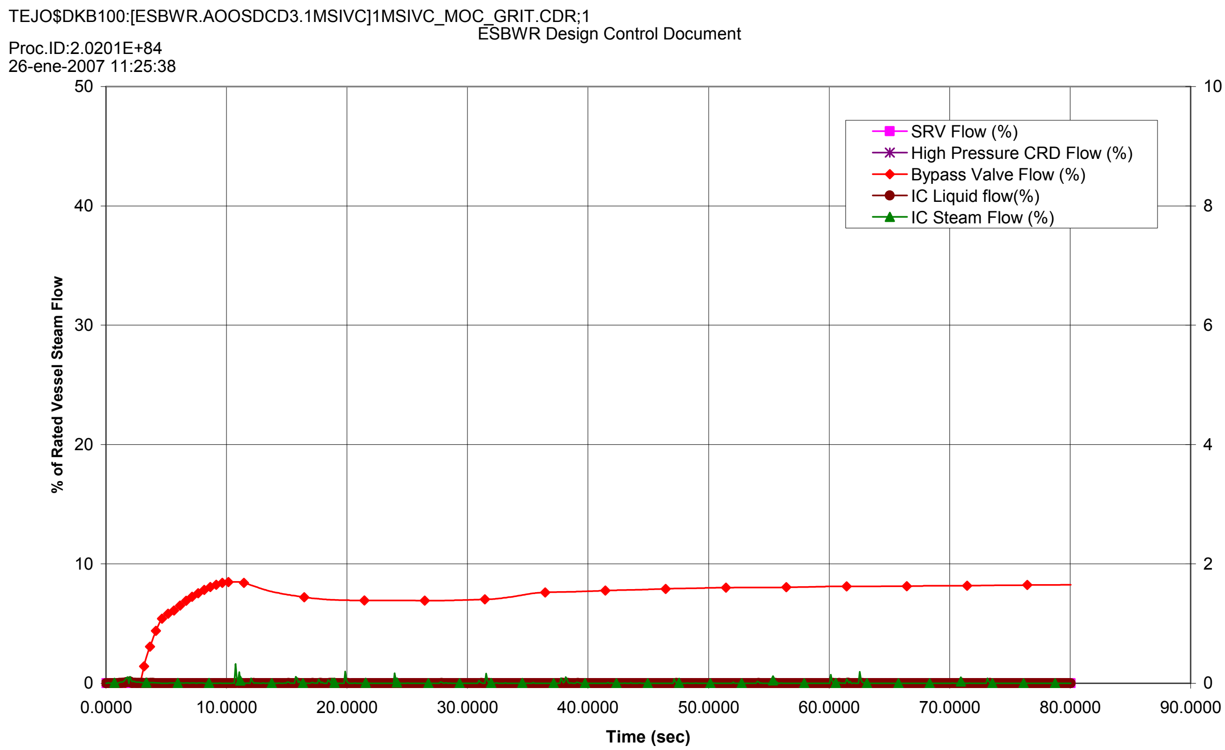


Figure 15.2-8b. One MSIV Closure

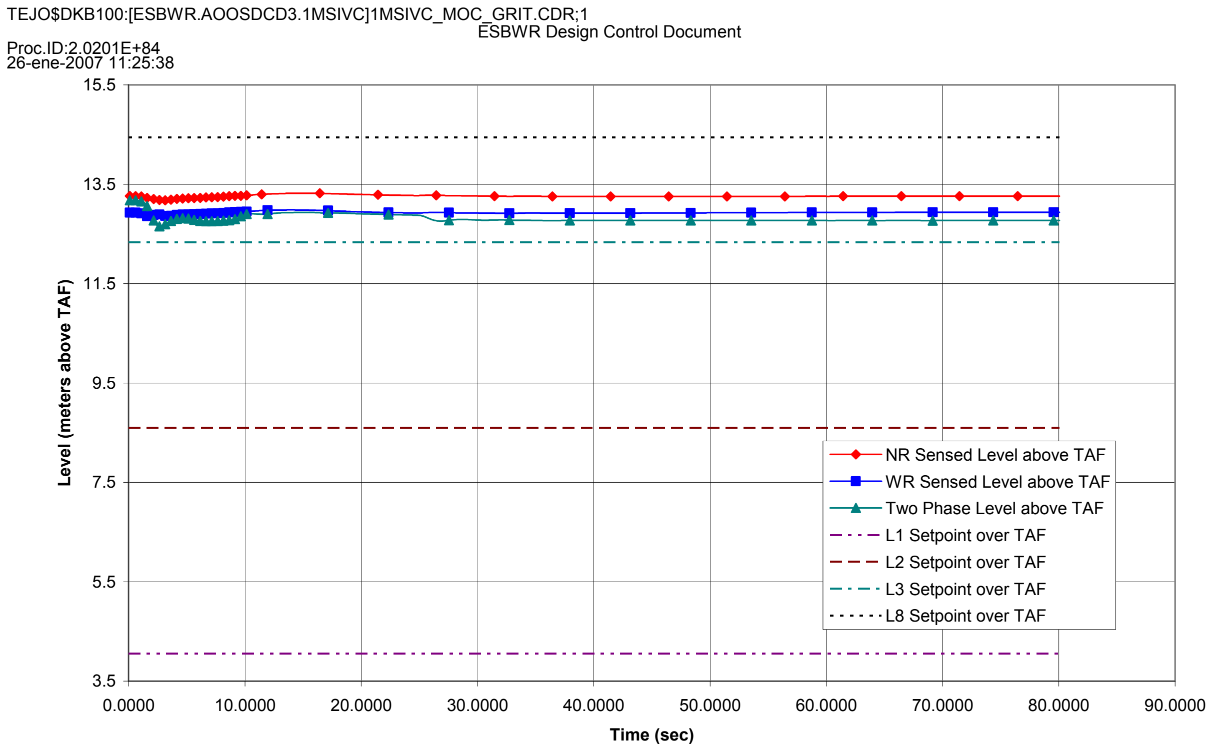


Figure 15.2-8c. One MSIV Closure

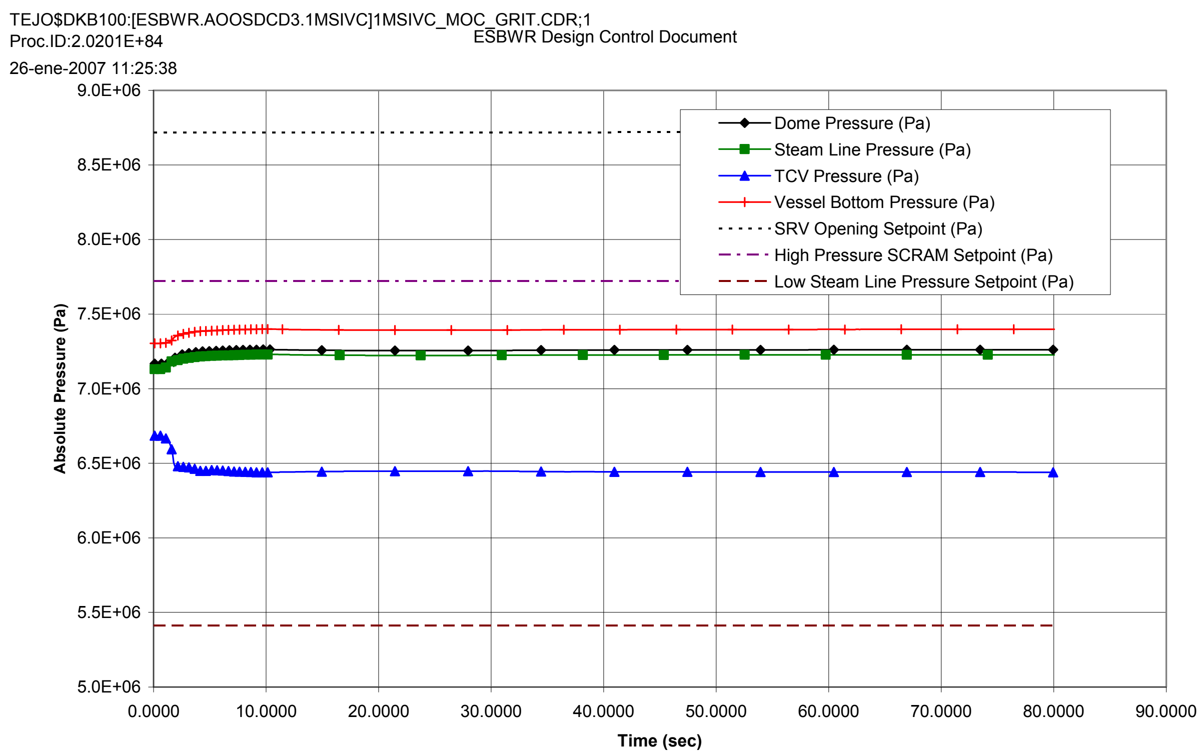


Figure 15.2-8d. One MSIV Closure

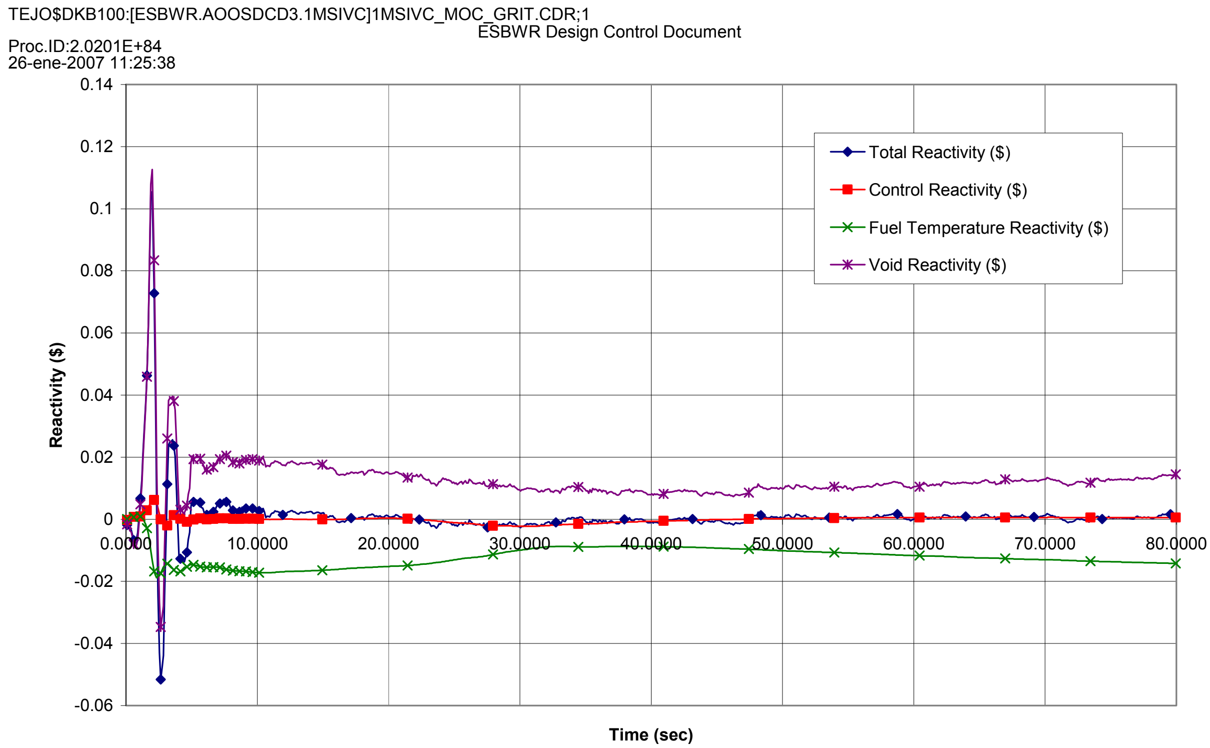


Figure 15.2-8e. One MSIV Closure

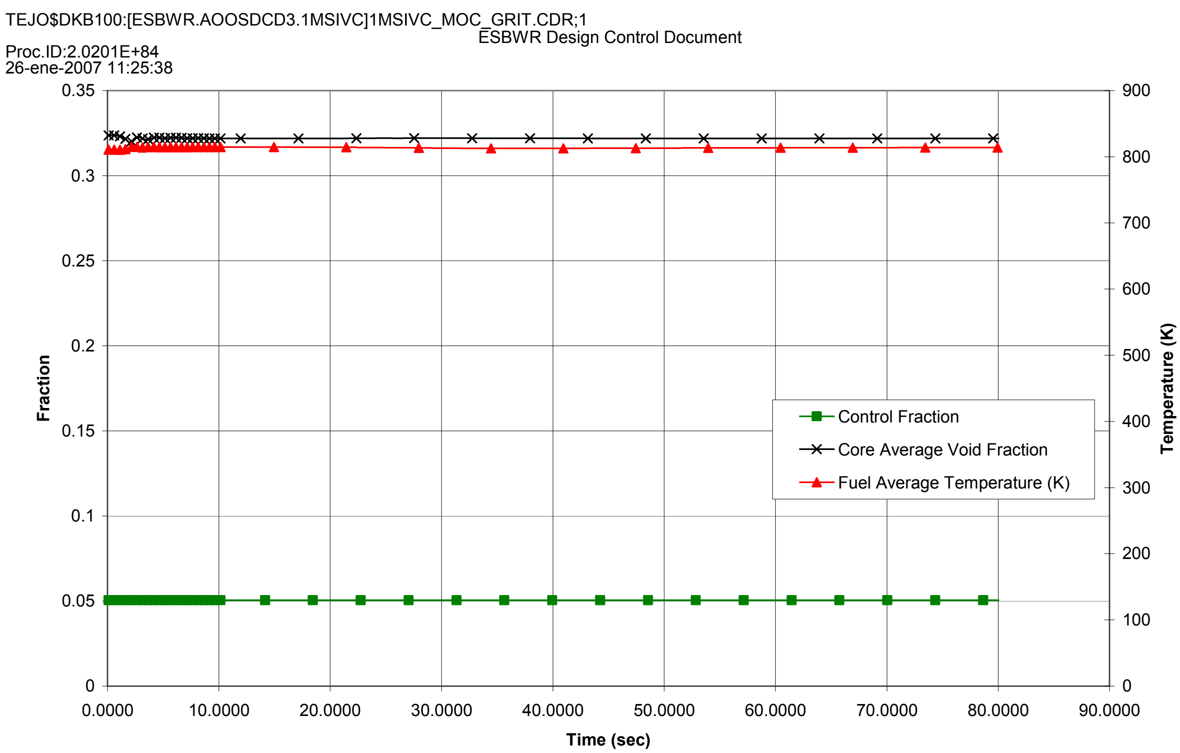


Figure 15.2-8f. One MSIV Closure

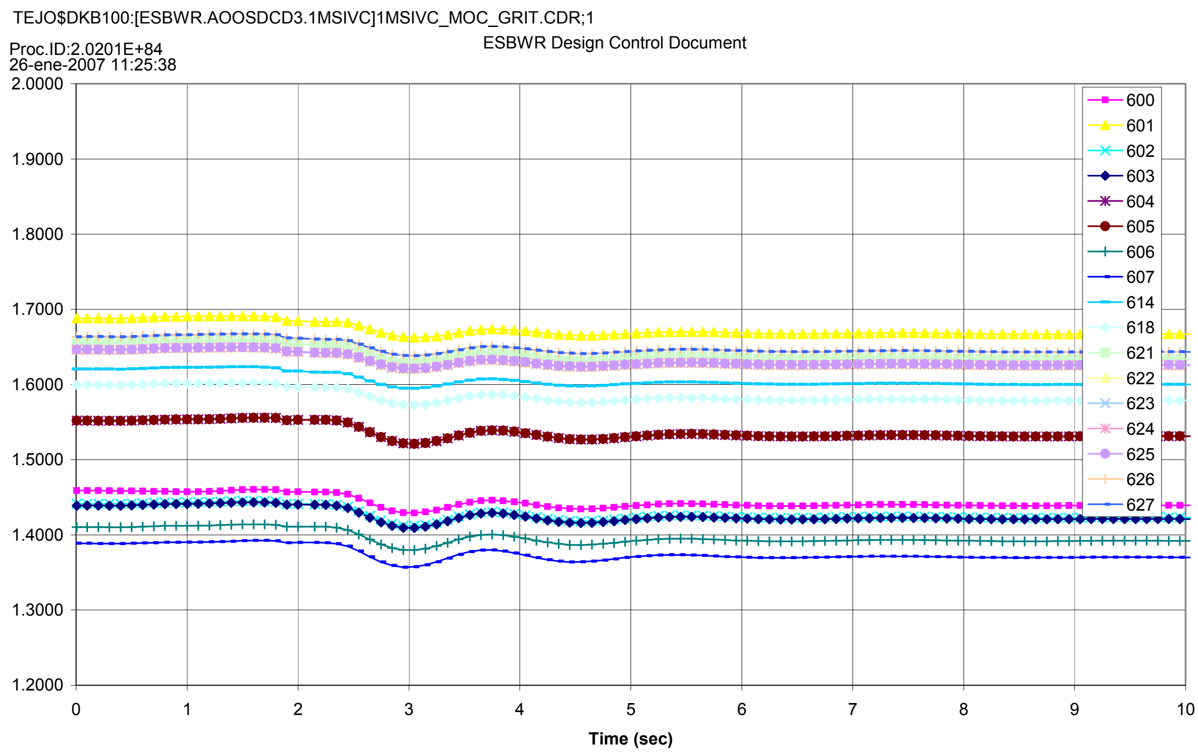


Figure 15.2-8g. One MSIV Closure

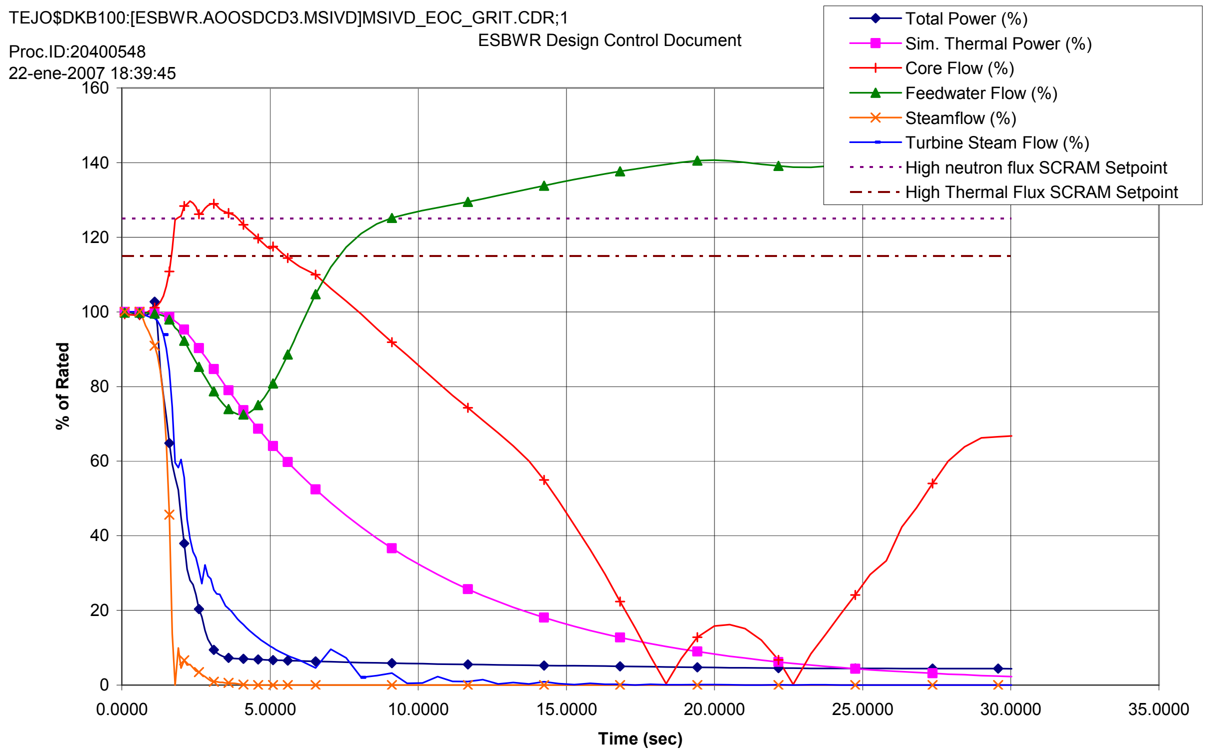


Figure 15.2-9a. MSIV Closure

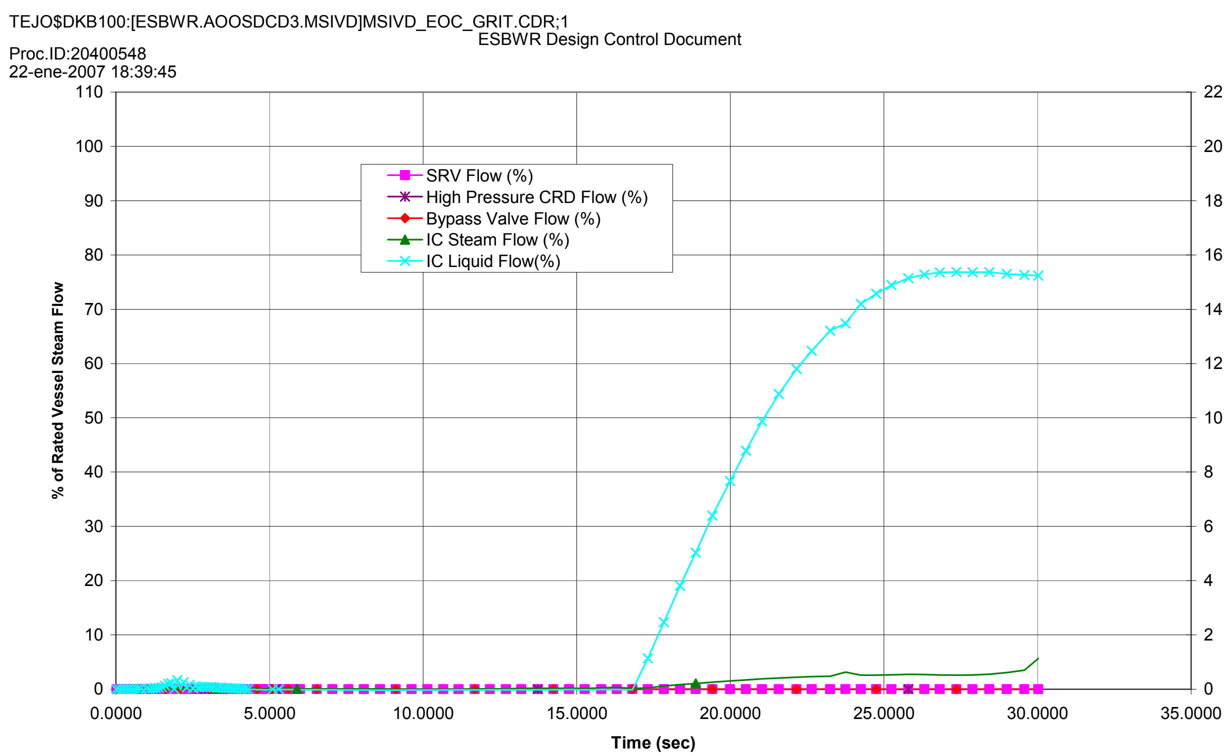


Figure 15.2-9b. MSIV Closure

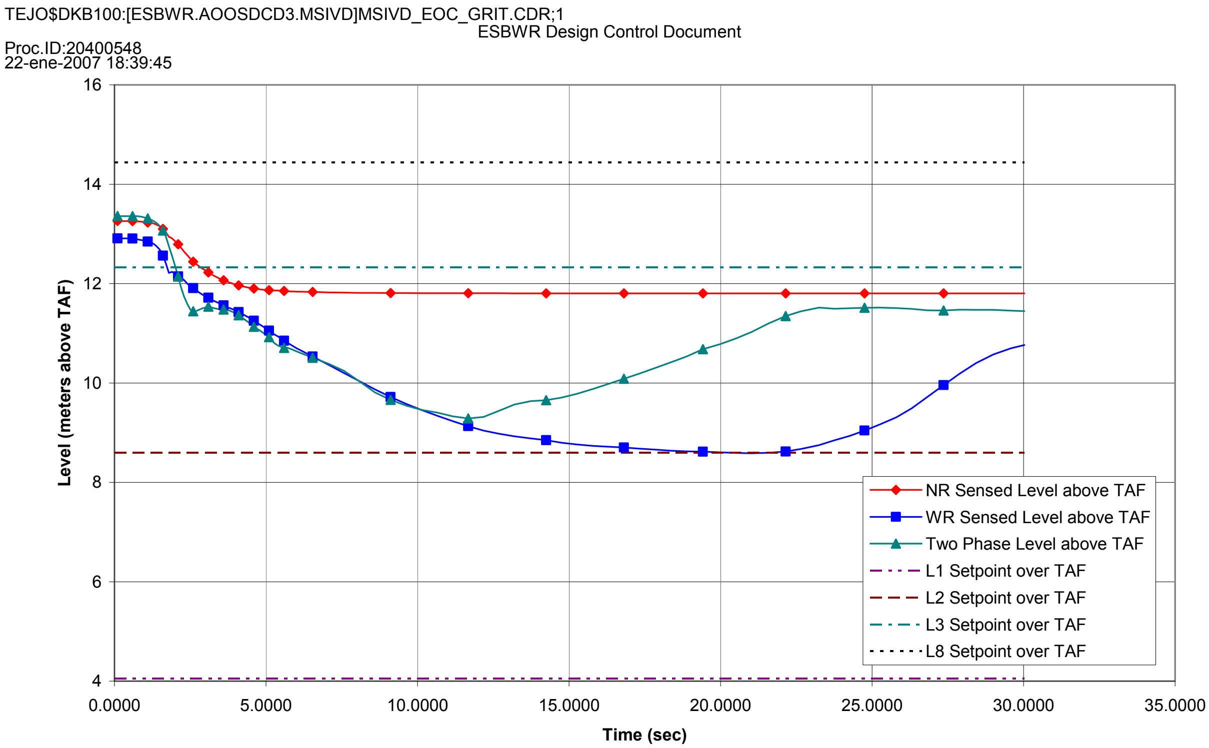


Figure 15.2-9c. MSIV Closure

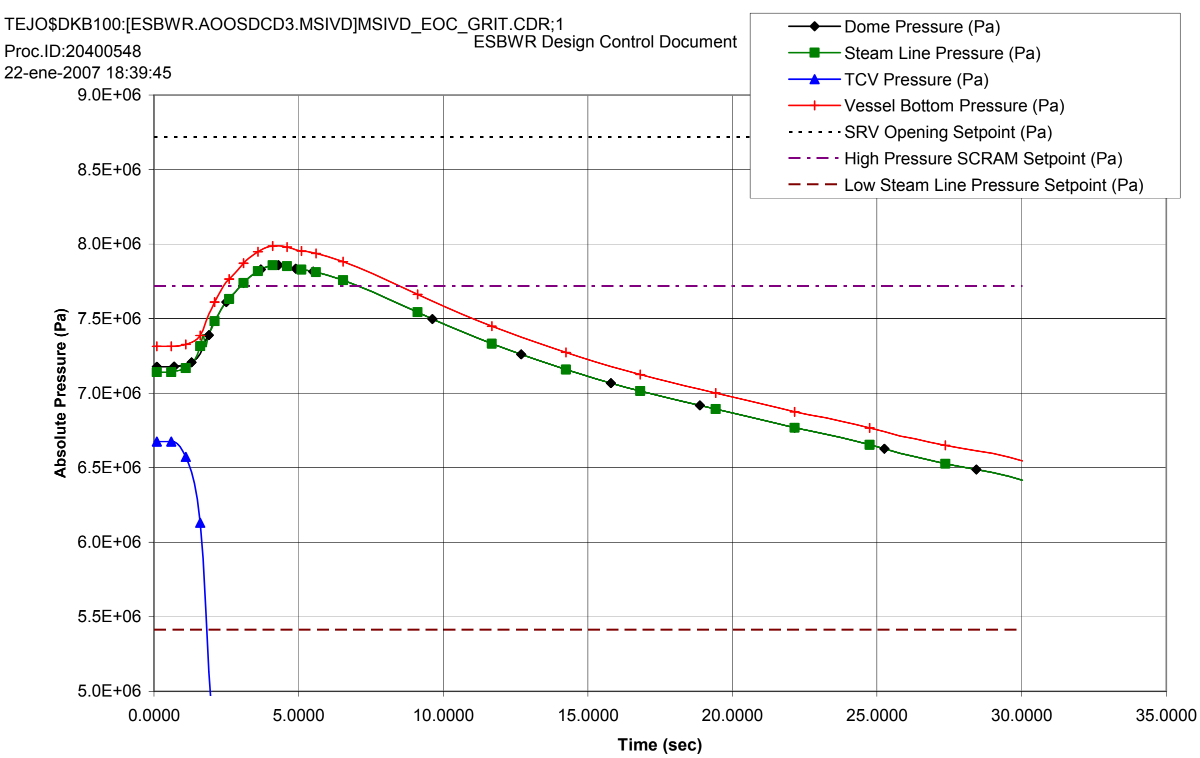


Figure 15.2-9d. MSIV Closure

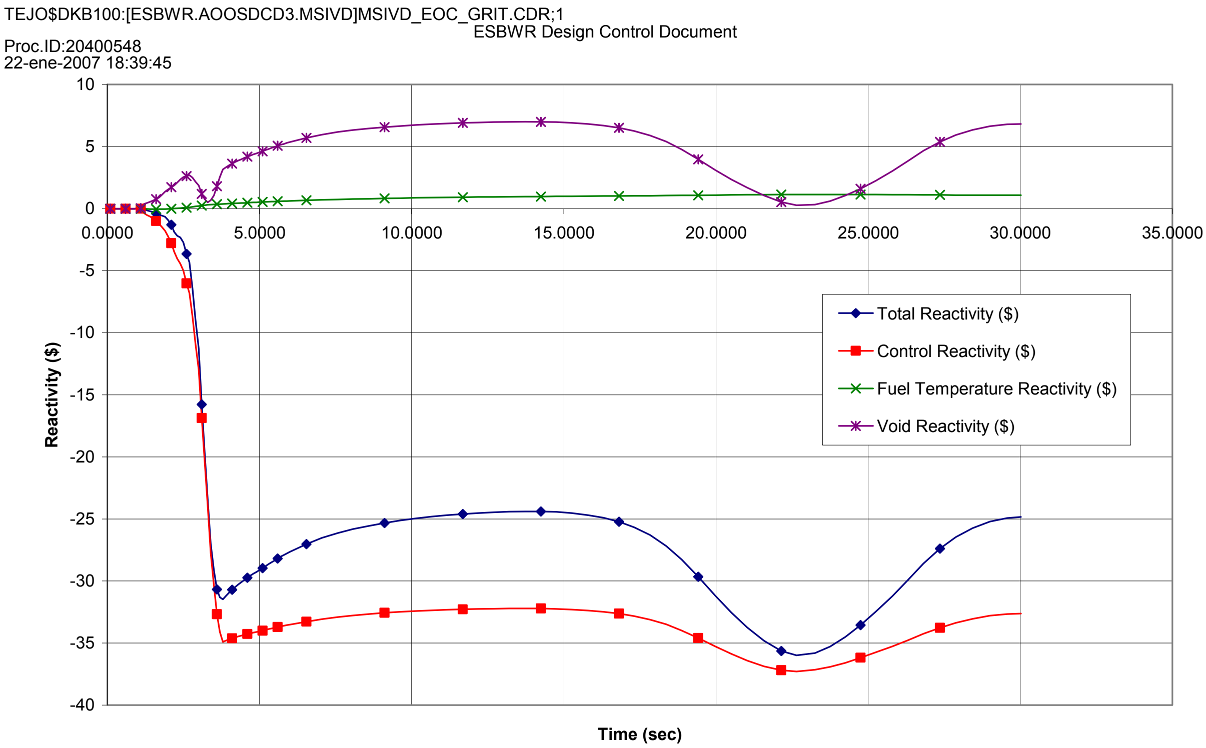


Figure 15.2-9e. MSIV Closure

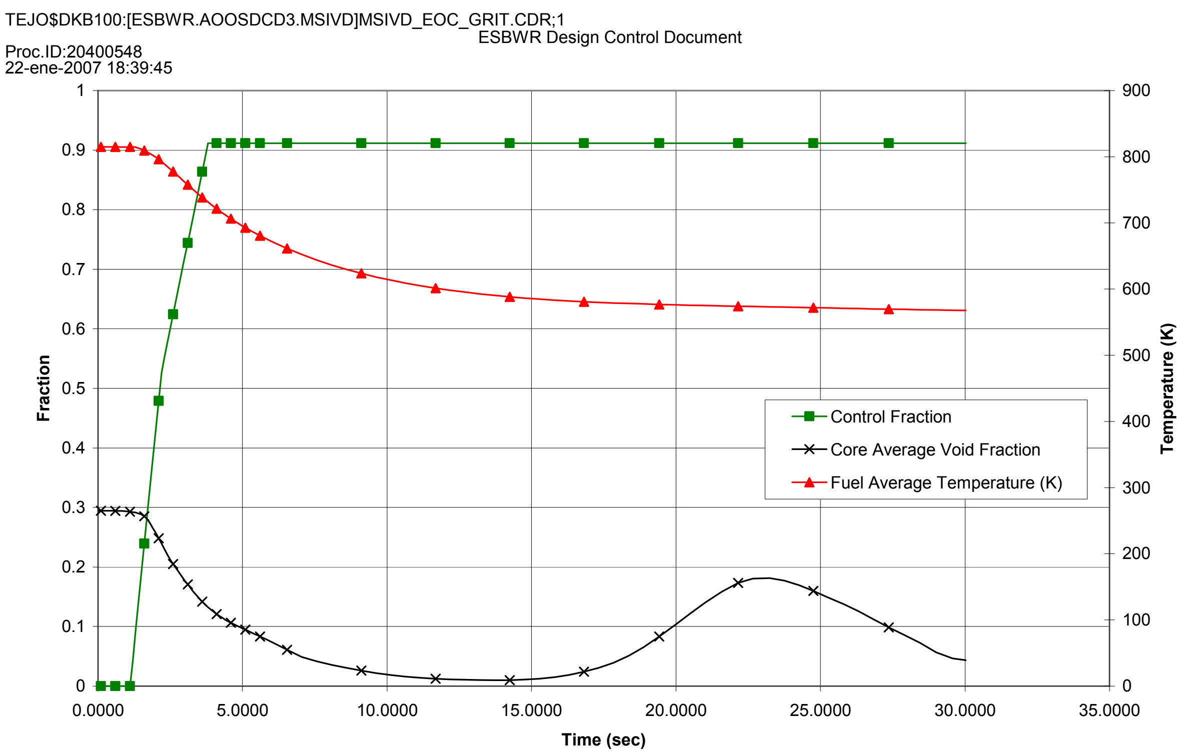


Figure 15.2-9f. MSIV Closure

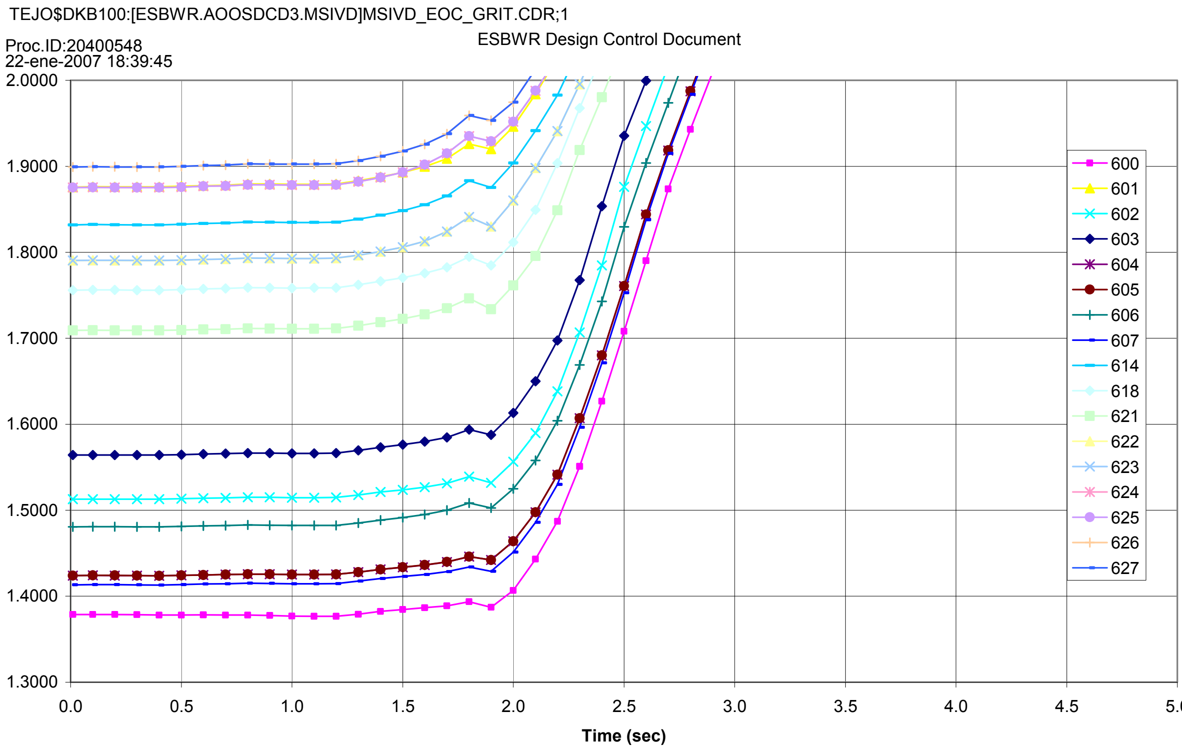


Figure 15.2-9g. MSIV Closure

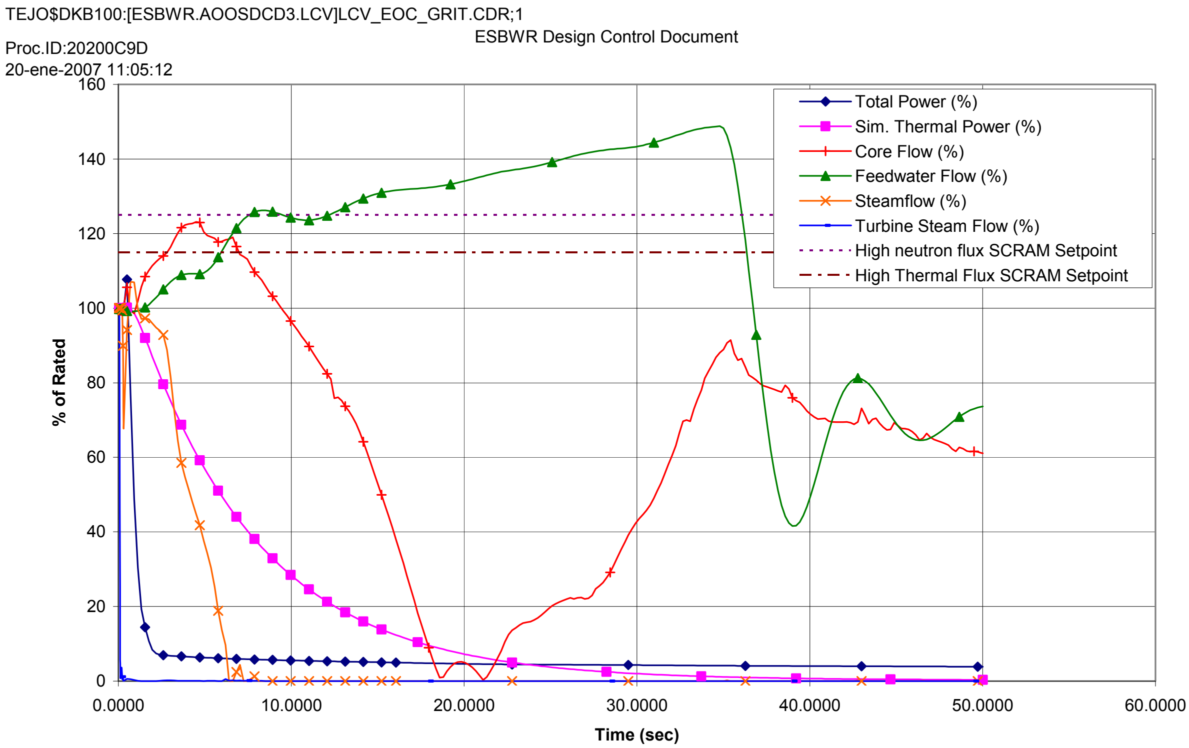


Figure 15.2-10a. Loss of Condenser Vacuum

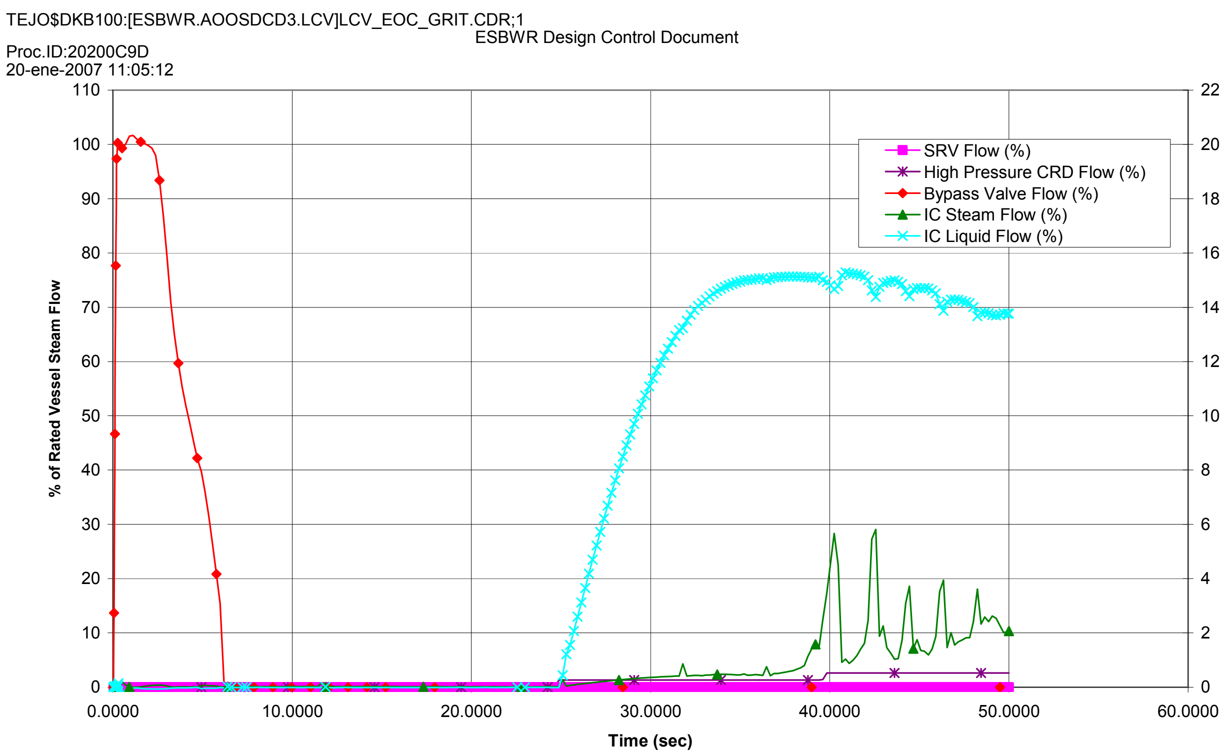


Figure 15.2-10b. Loss of Condenser Vacuum

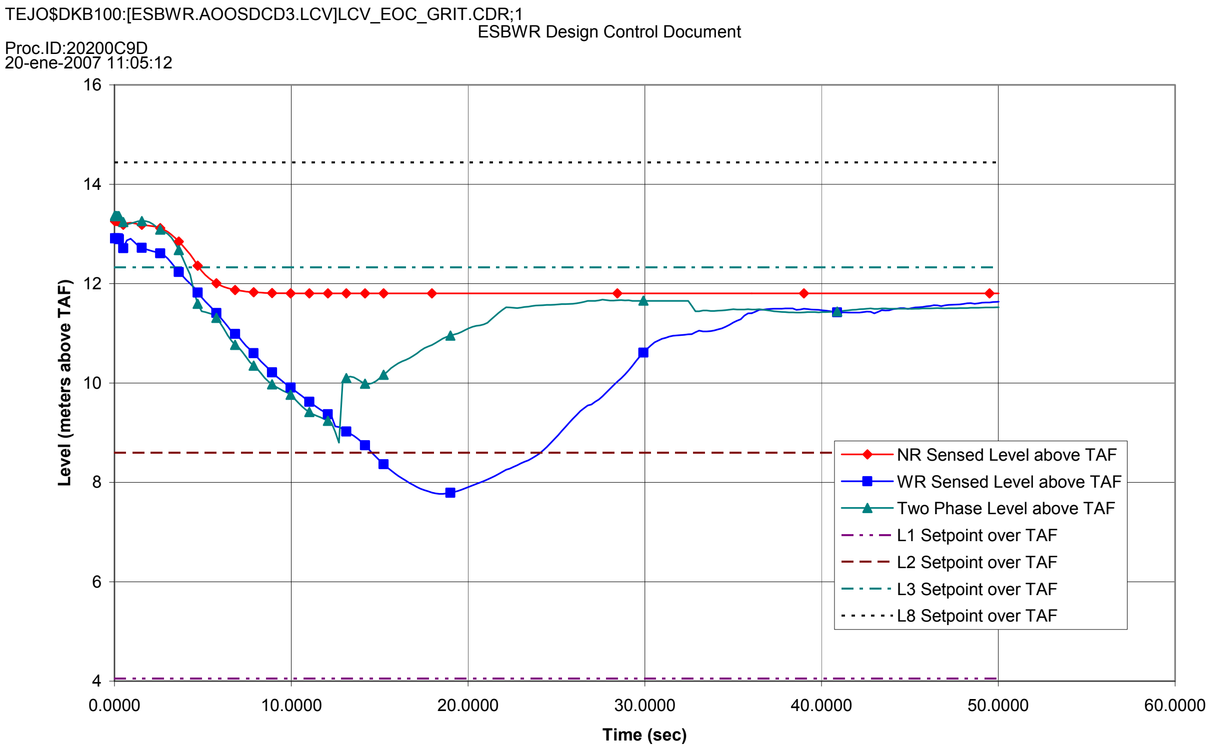


Figure 15.2-10c. Loss of Condenser Vacuum

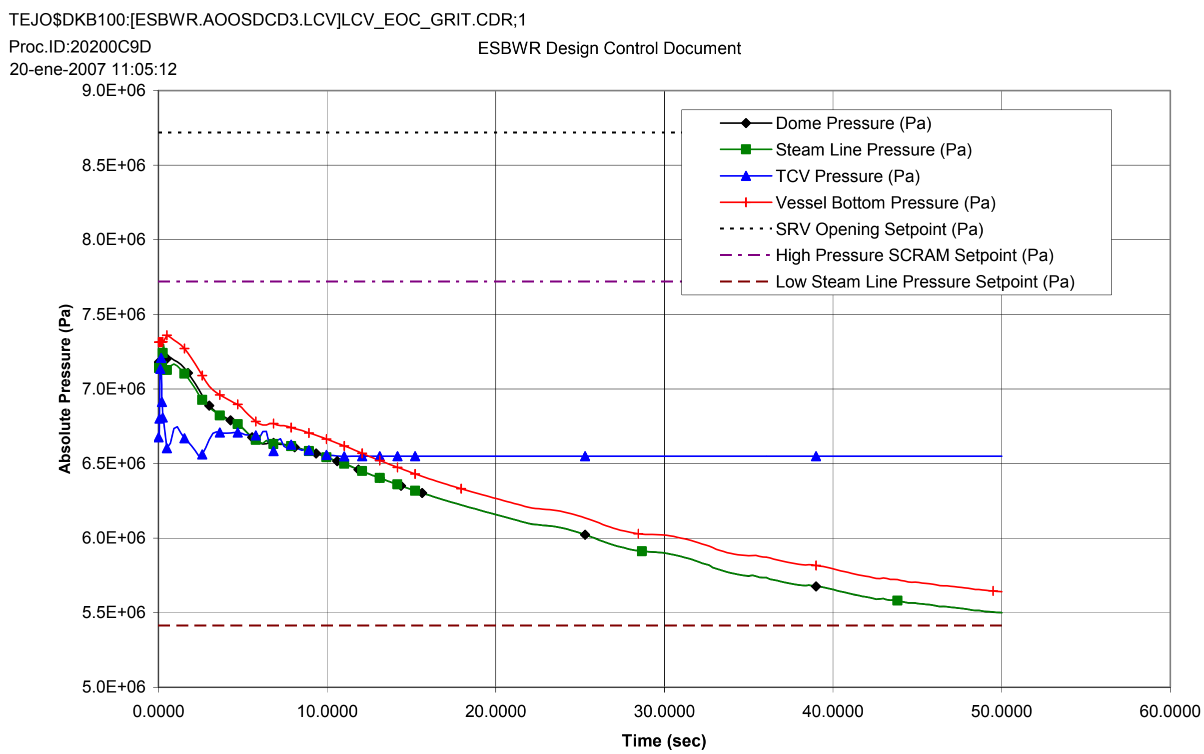


Figure 15.2-10d. Loss of Condenser Vacuum

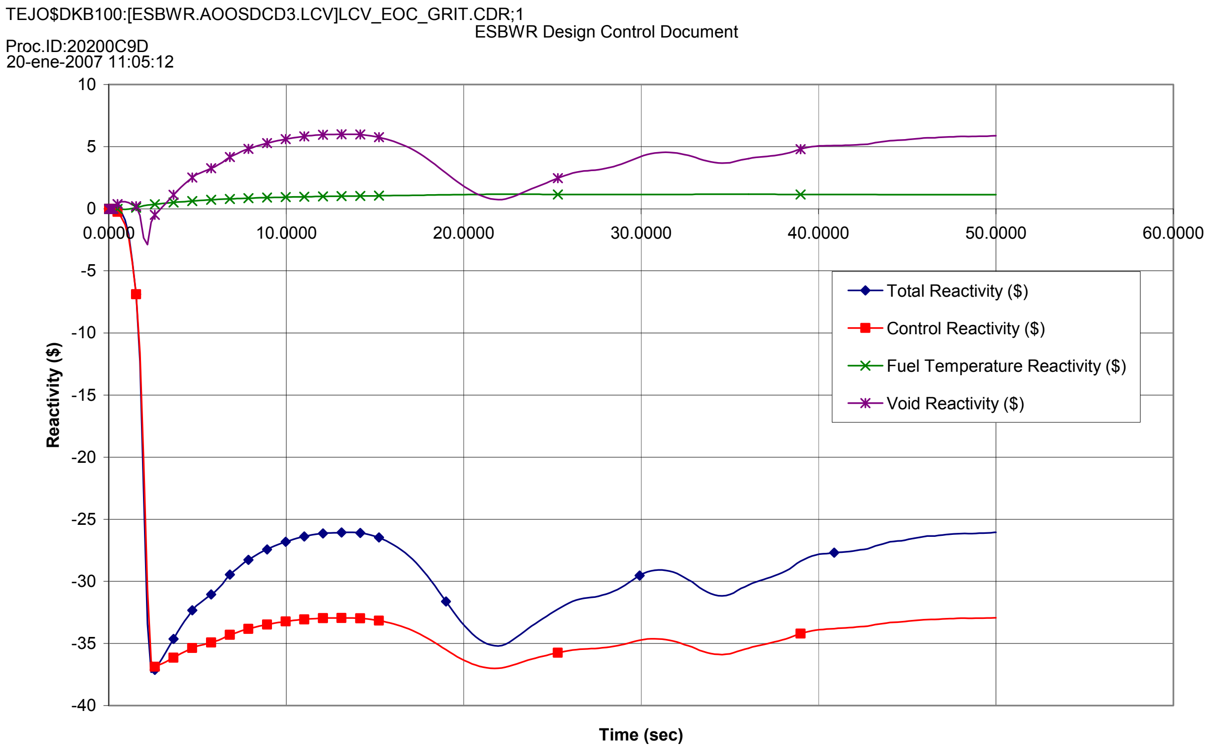


Figure 15.2-10e. Loss of Condenser Vacuum

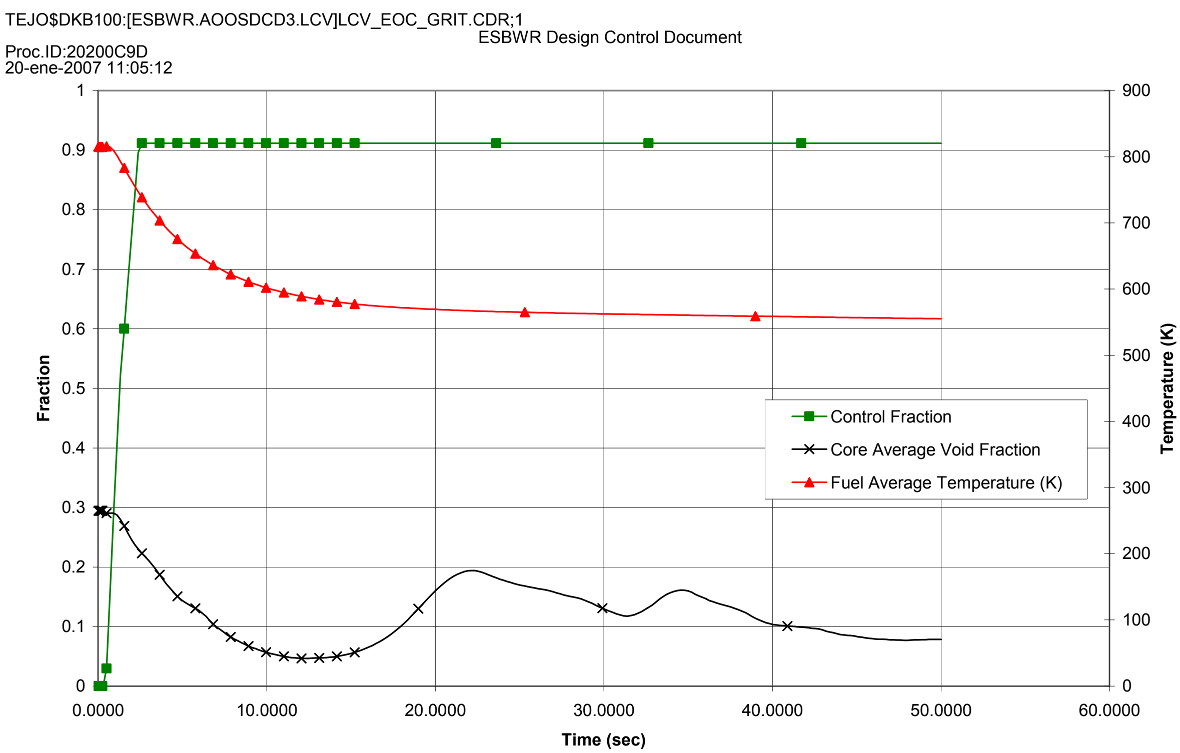


Figure 15.2-10f. Loss of Condenser Vacuum

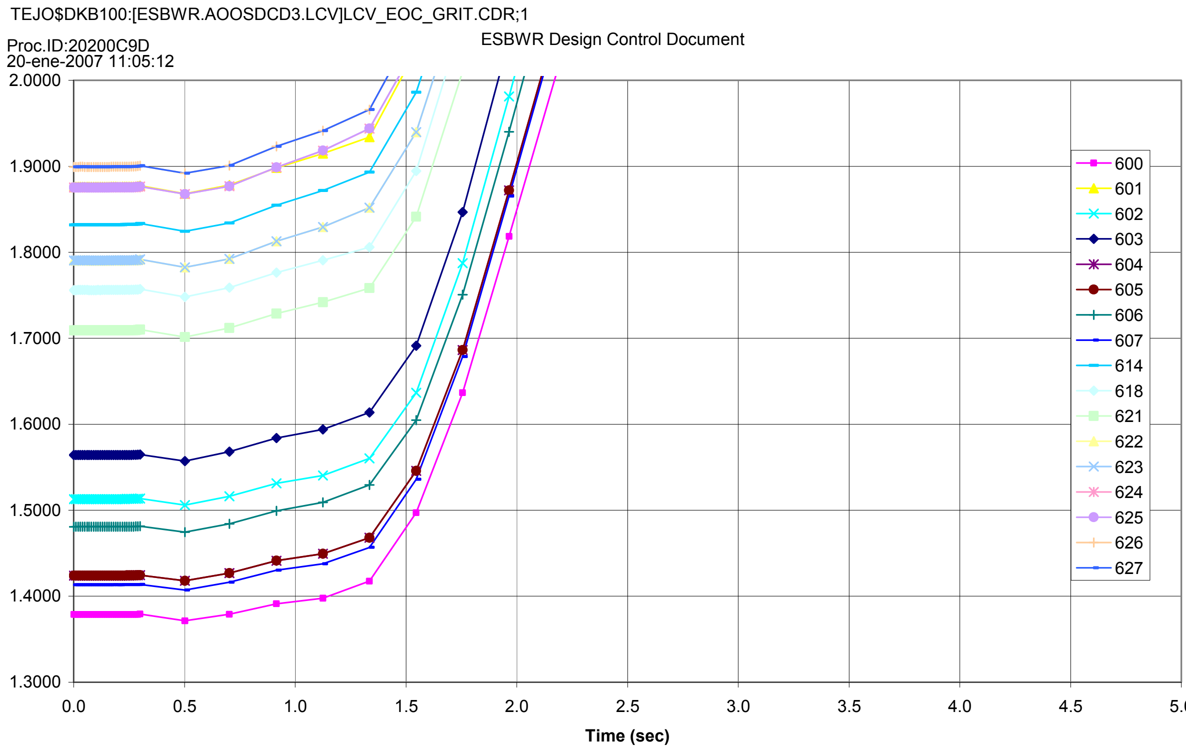


Figure 15.2-10g. Loss of Condenser Vacuum

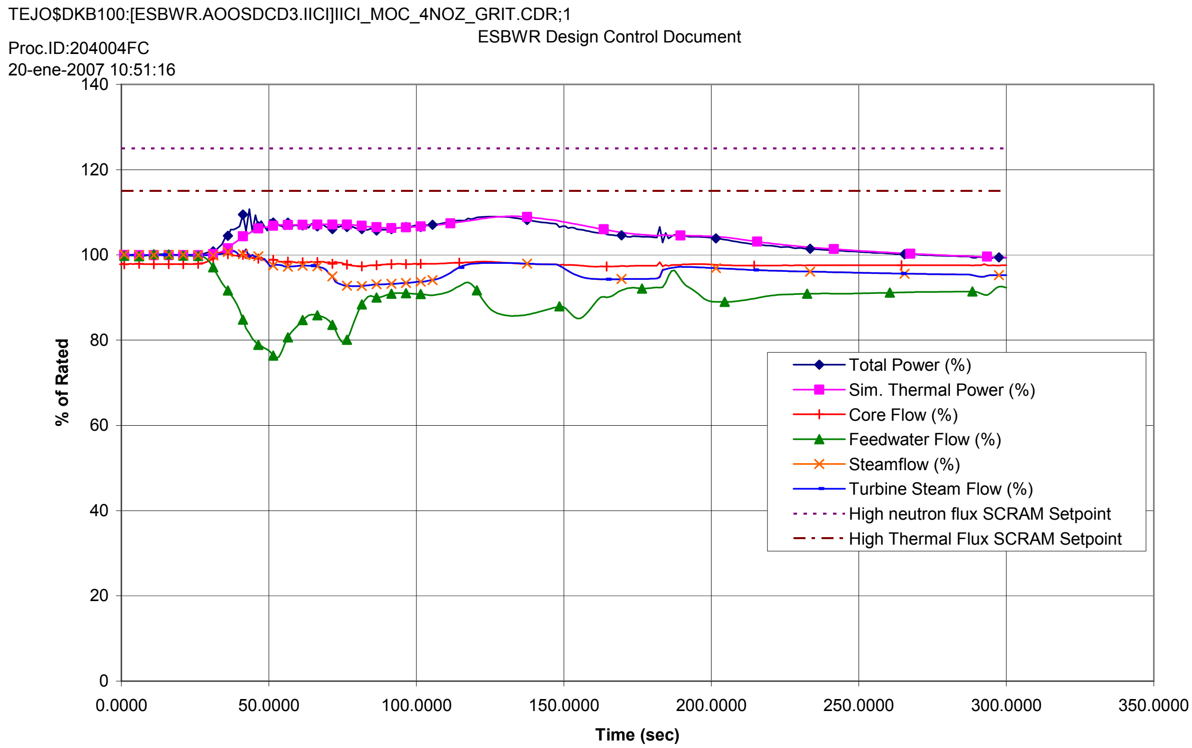


Figure 15.2-11a. Inadvertent Isolation Condenser Initiation

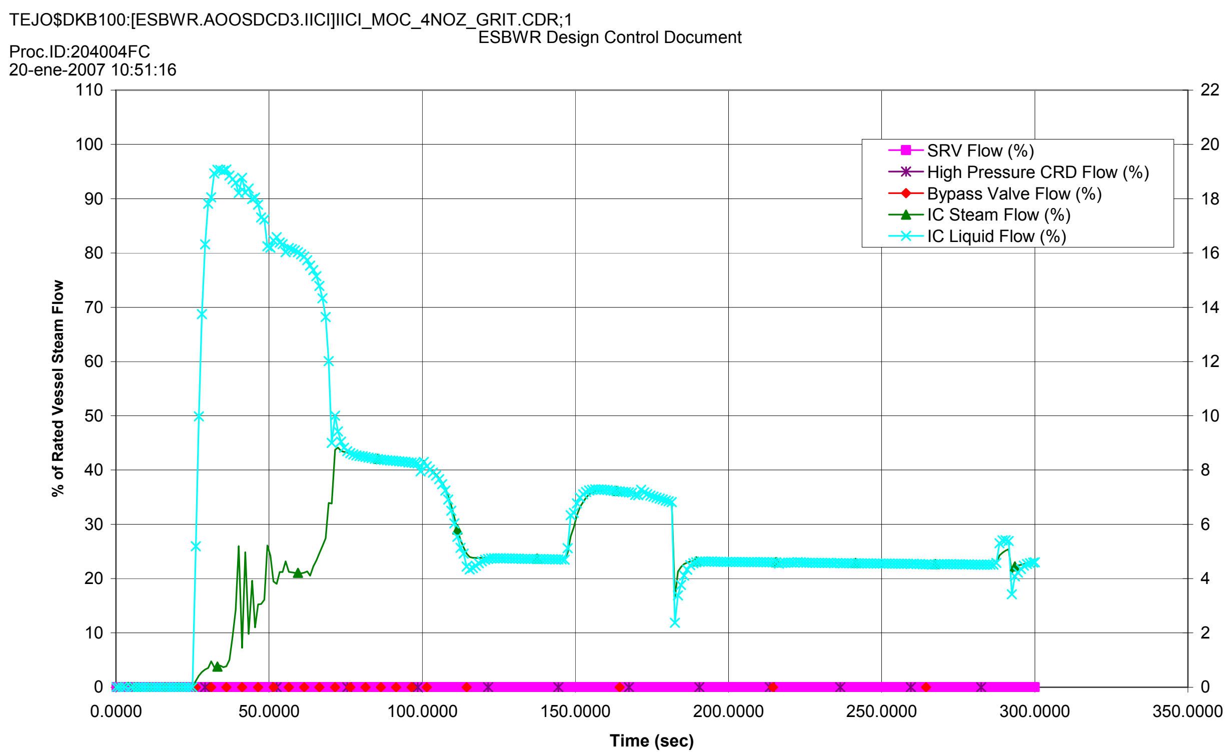


Figure 15.2-11b. Inadvertent Isolation Condenser Initiation

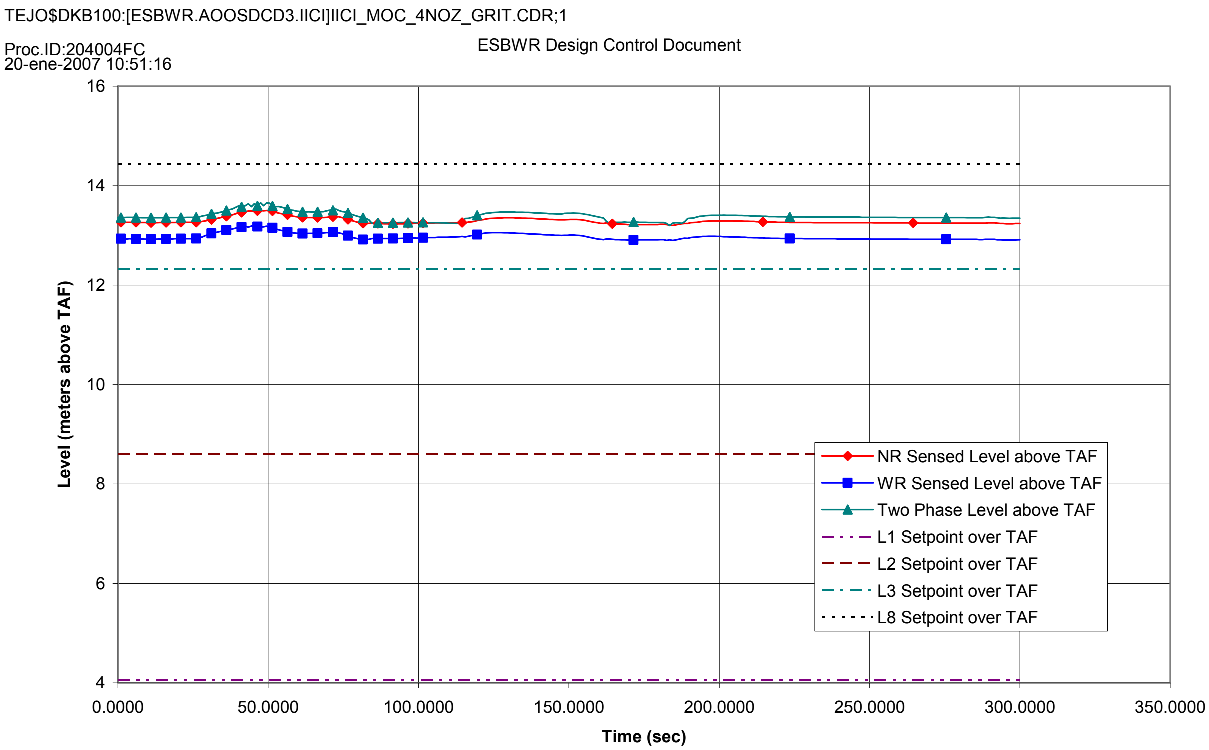


Figure 15.2-11c. Inadvertent Isolation Condenser Initiation

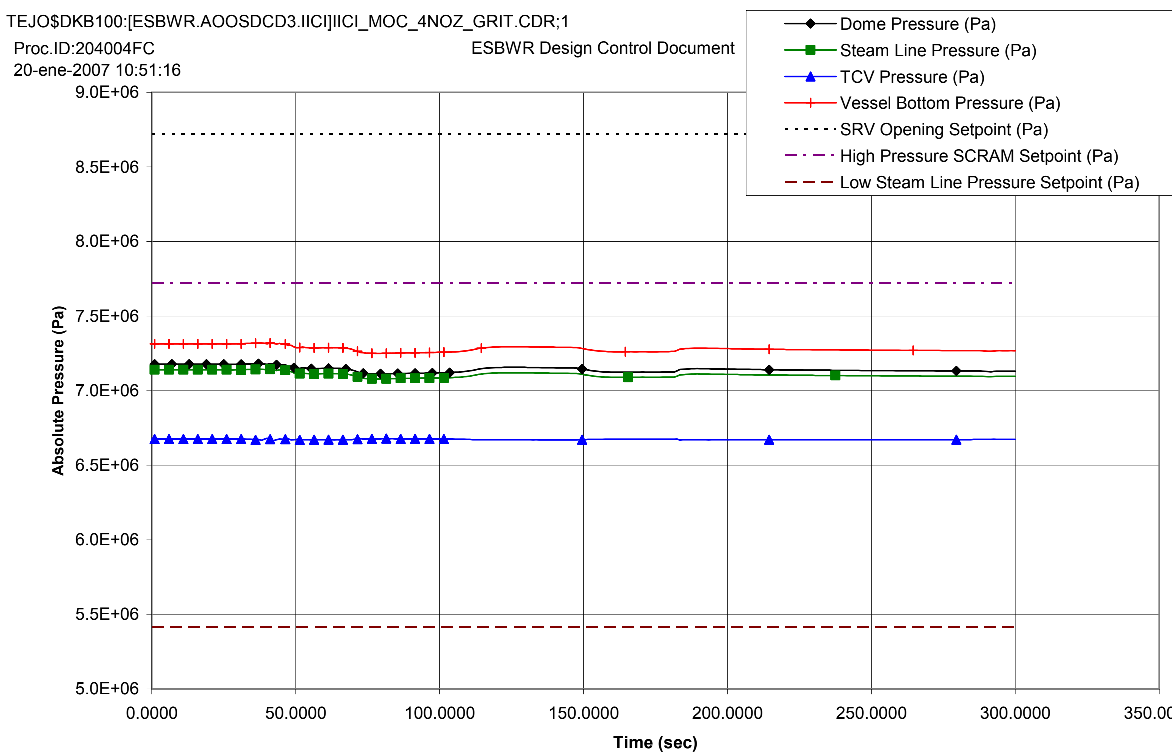


Figure 15.2-11d. Inadvertent Isolation Condenser Initiation

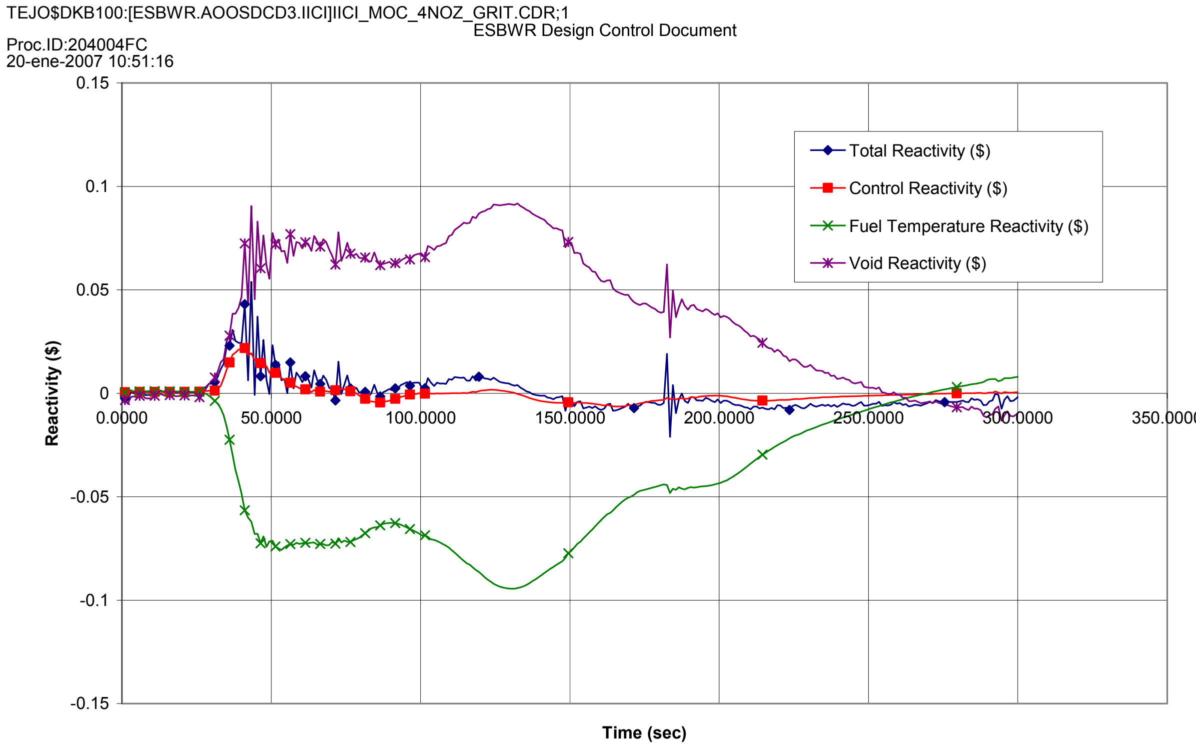


Figure 15.2-11e. Inadvertent Isolation Condenser Initiation

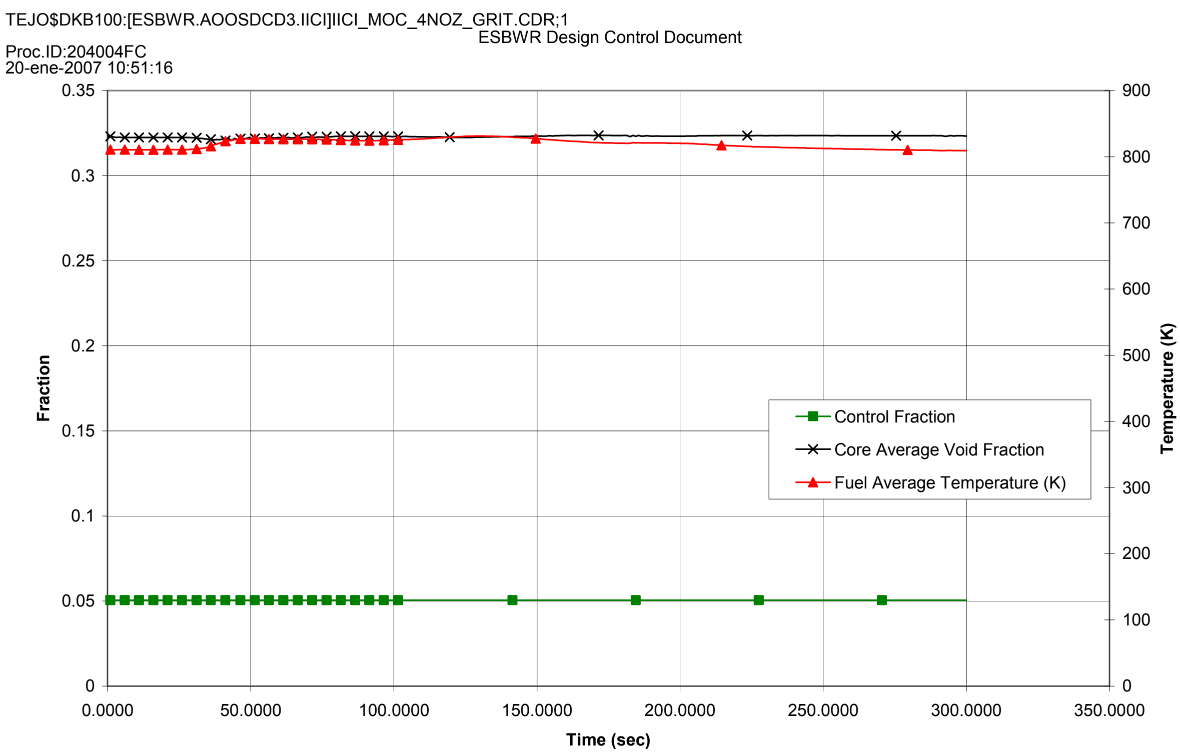


Figure 15.2-11f. Inadvertent Isolation Condenser Initiation

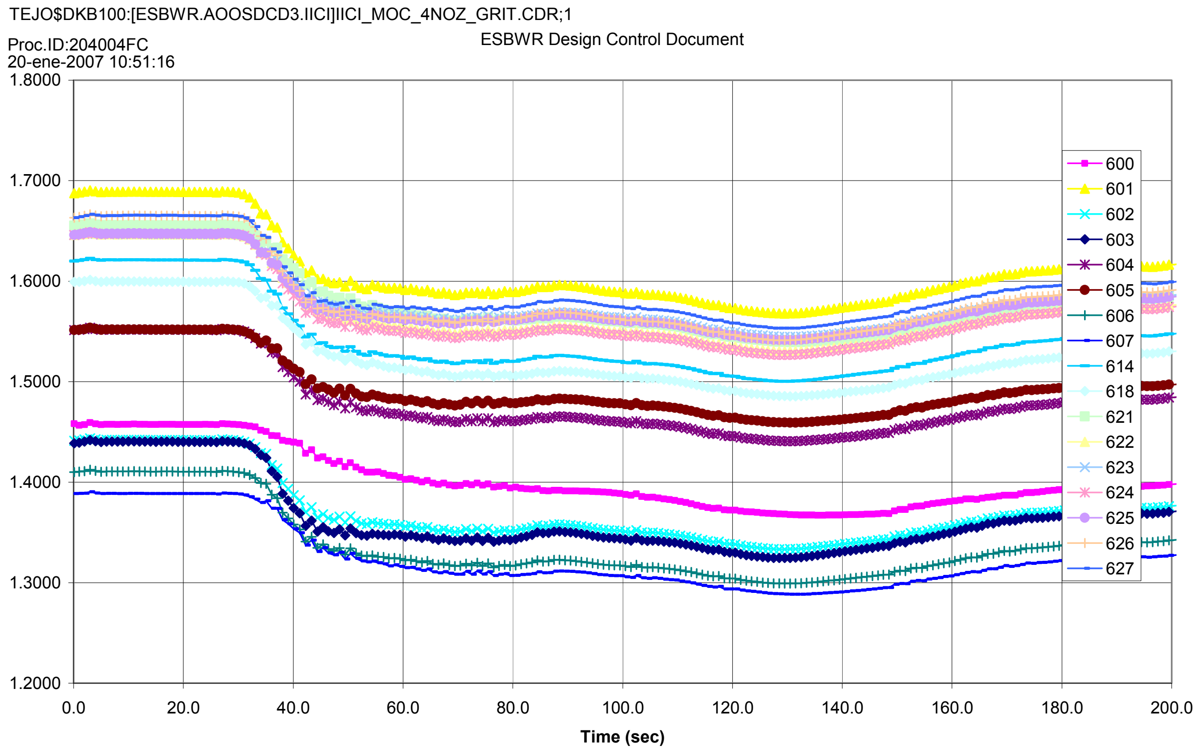


Figure 15.2-11g. Inadvertent Isolation Condenser Initiation

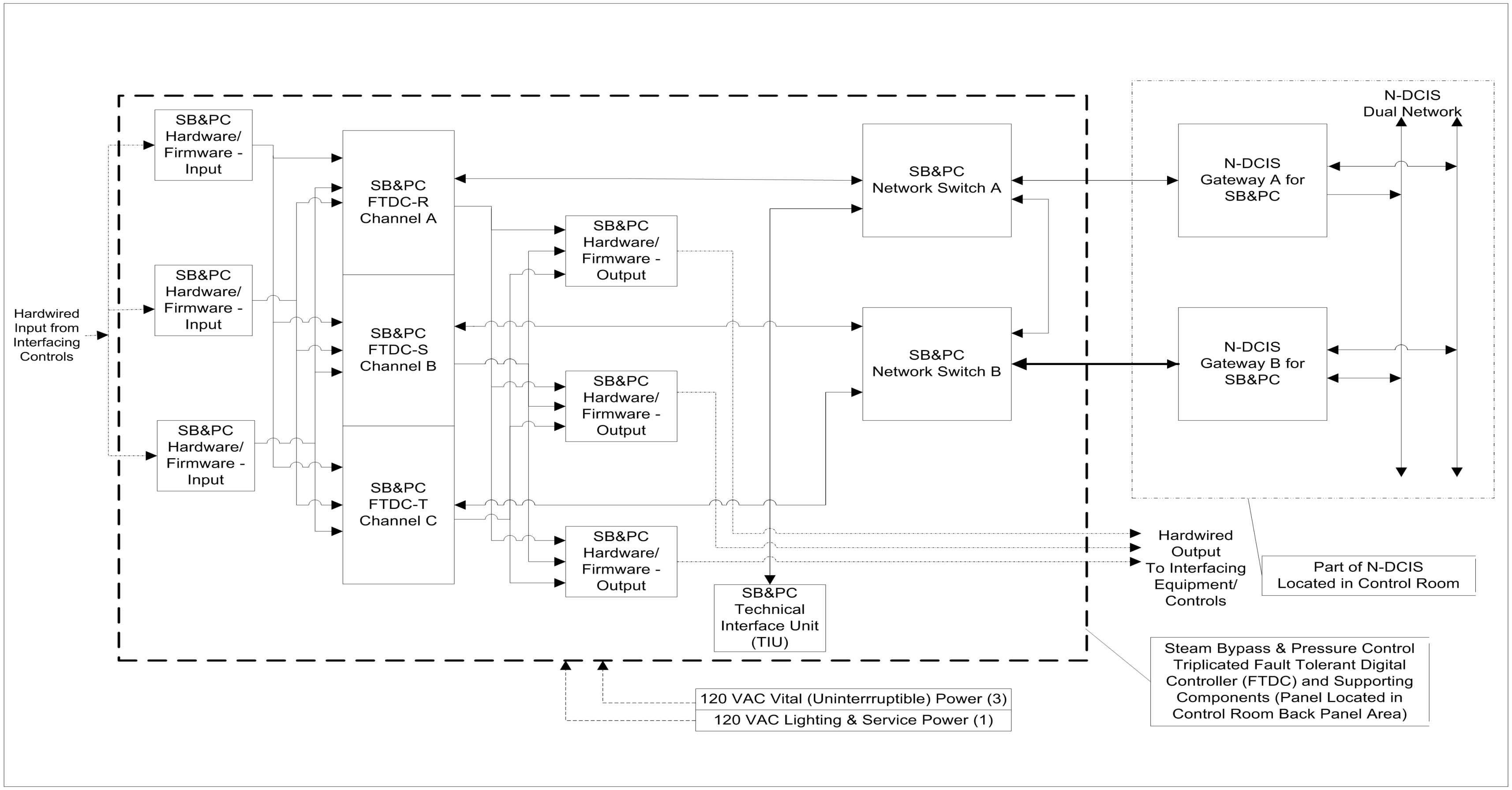


Figure 15.2-12. Simplified Block Diagram of Fault-Tolerant Digital Controller System

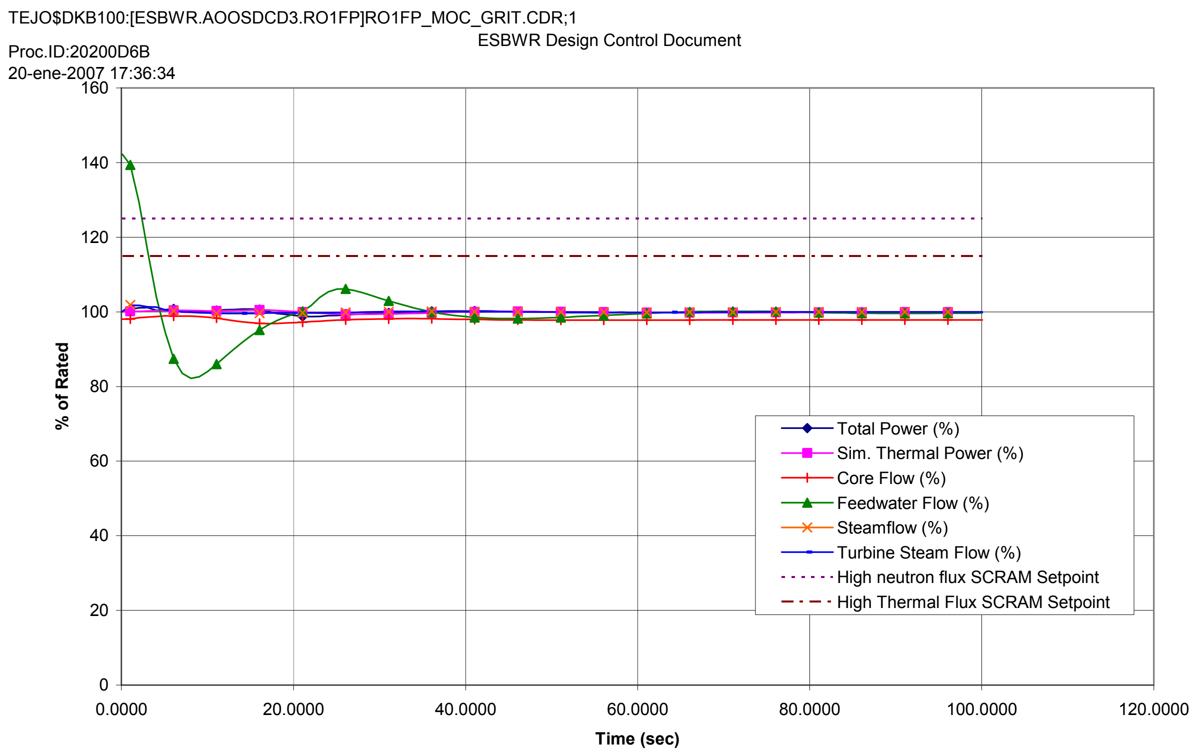


Figure 15.2-13a. Runout of One Feedwater Pump

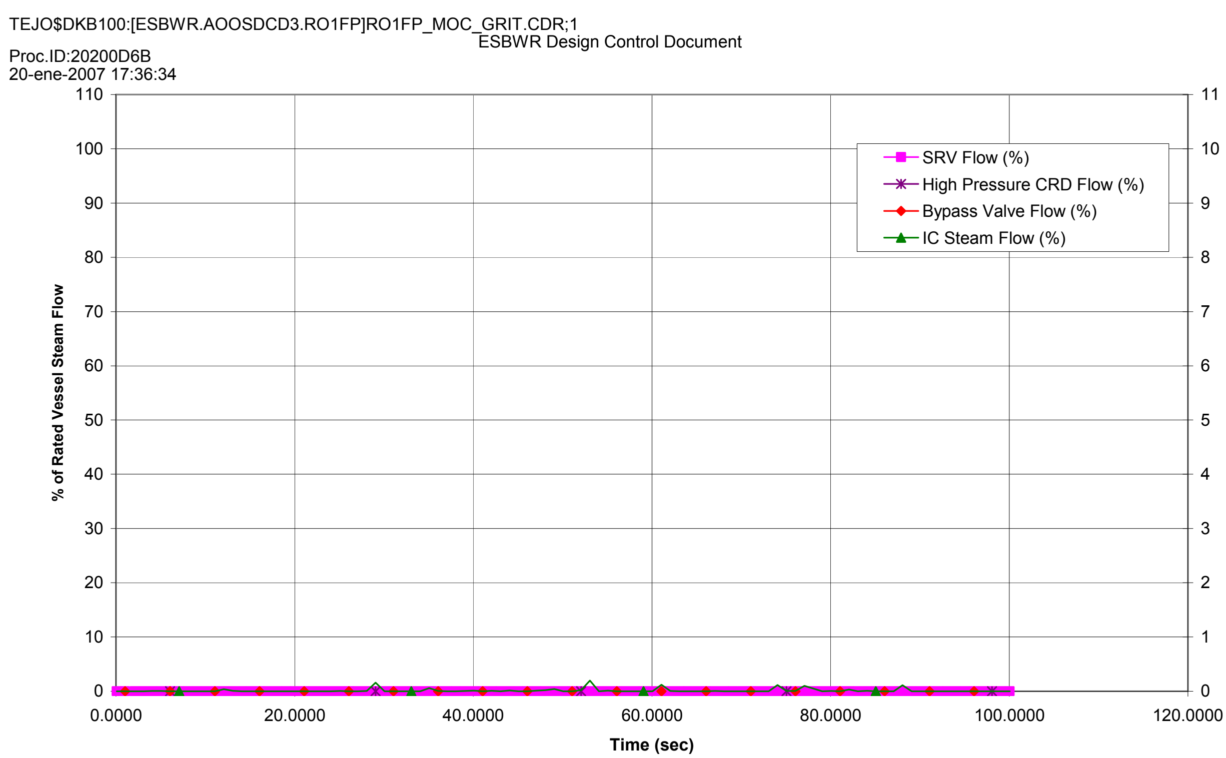


Figure 15.2-13b. Runout of One Feedwater Pump

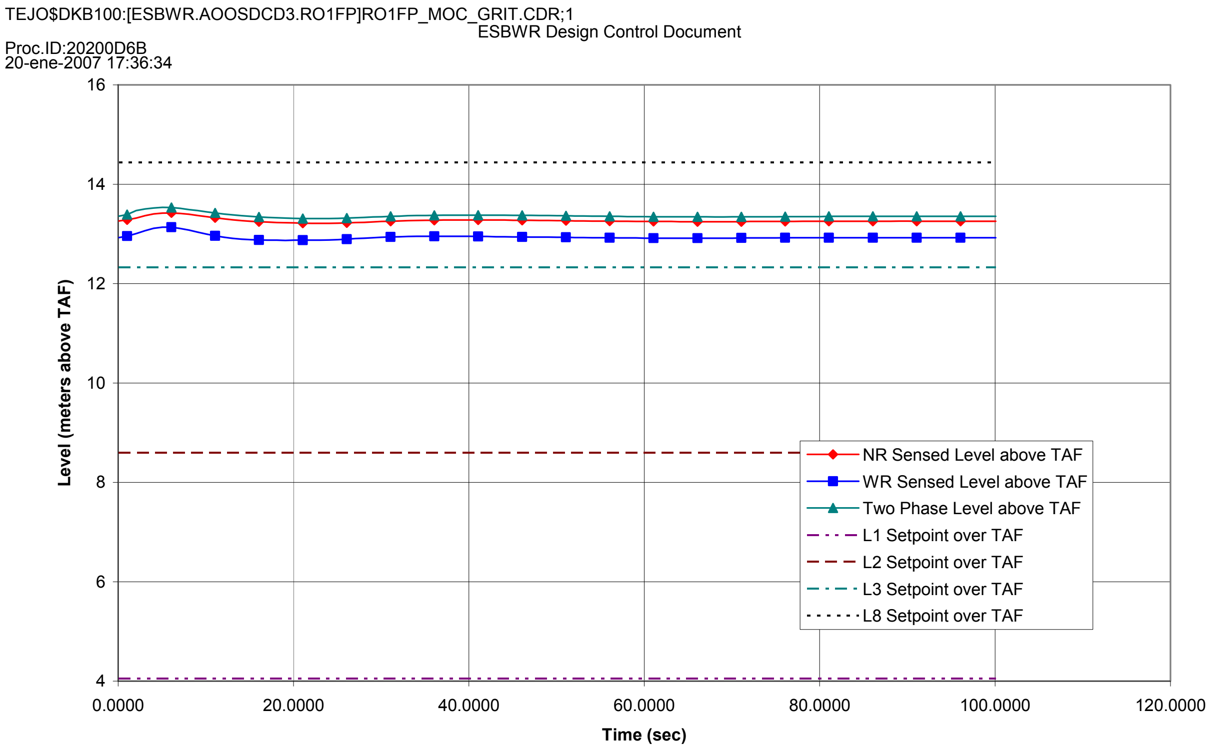


Figure 15.2-13c. Runout of One Feedwater Pump

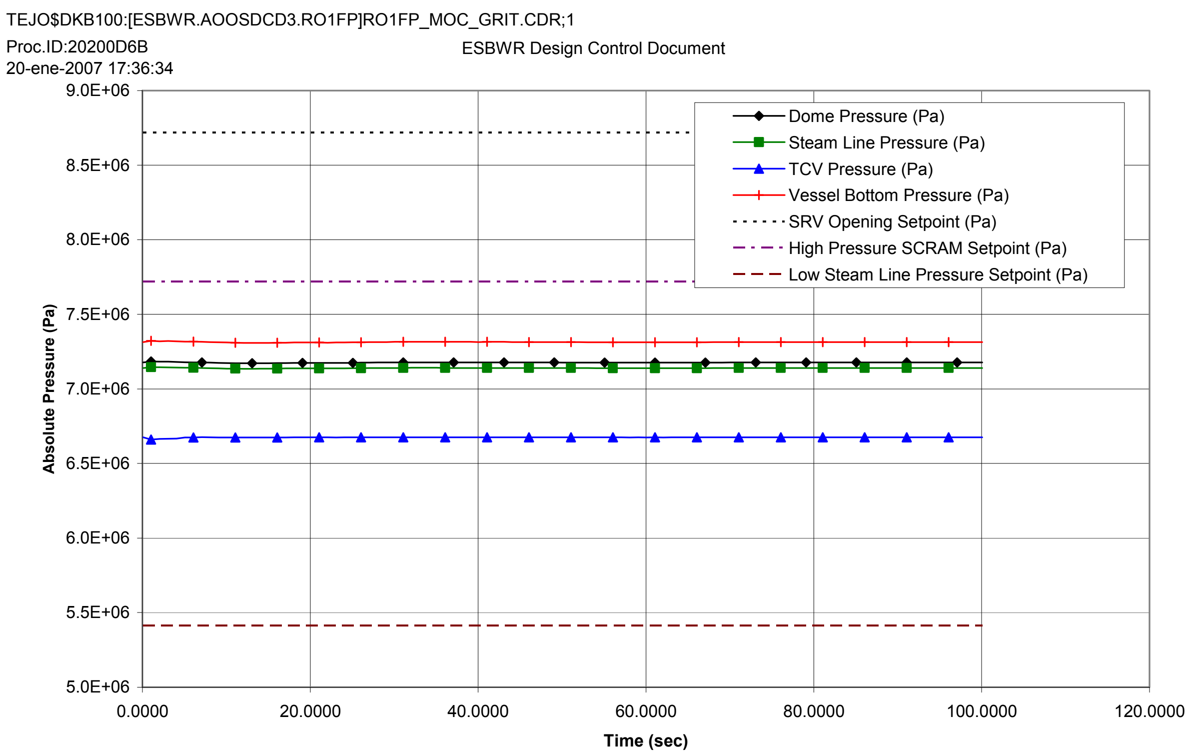


Figure 15.2-13d. Runout of One Feedwater Pump

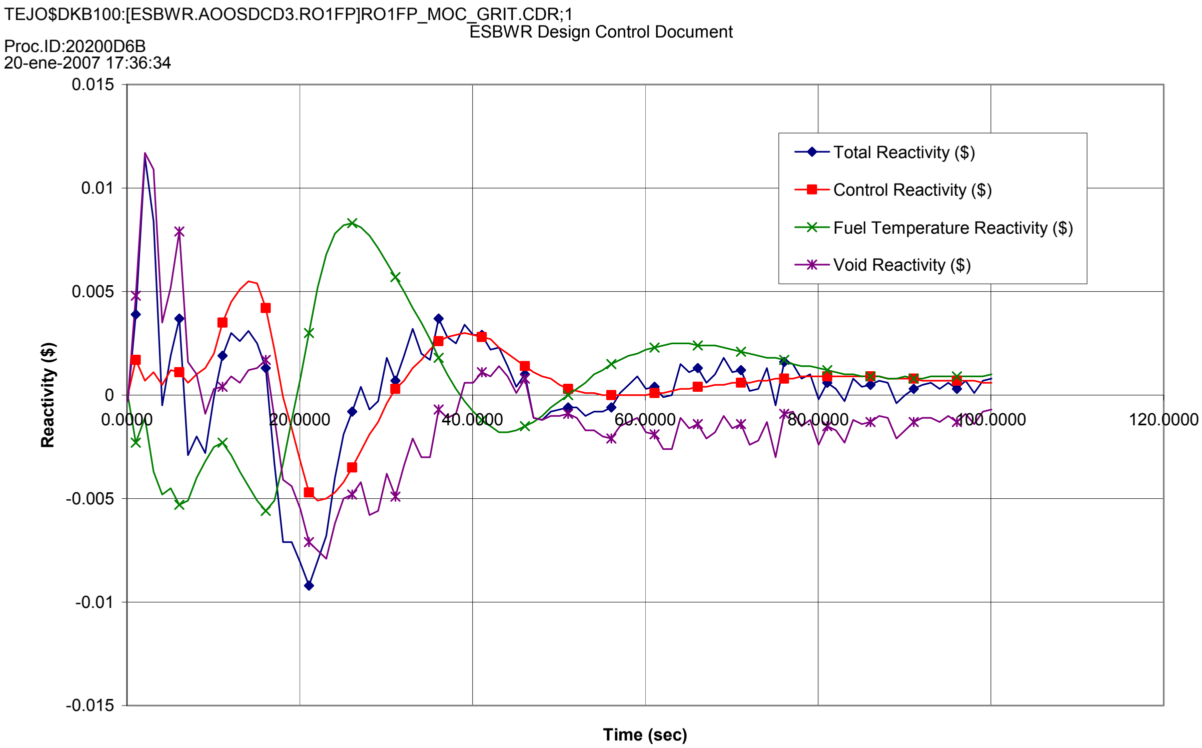


Figure 15.2-13e. Runout of One Feedwater Pump

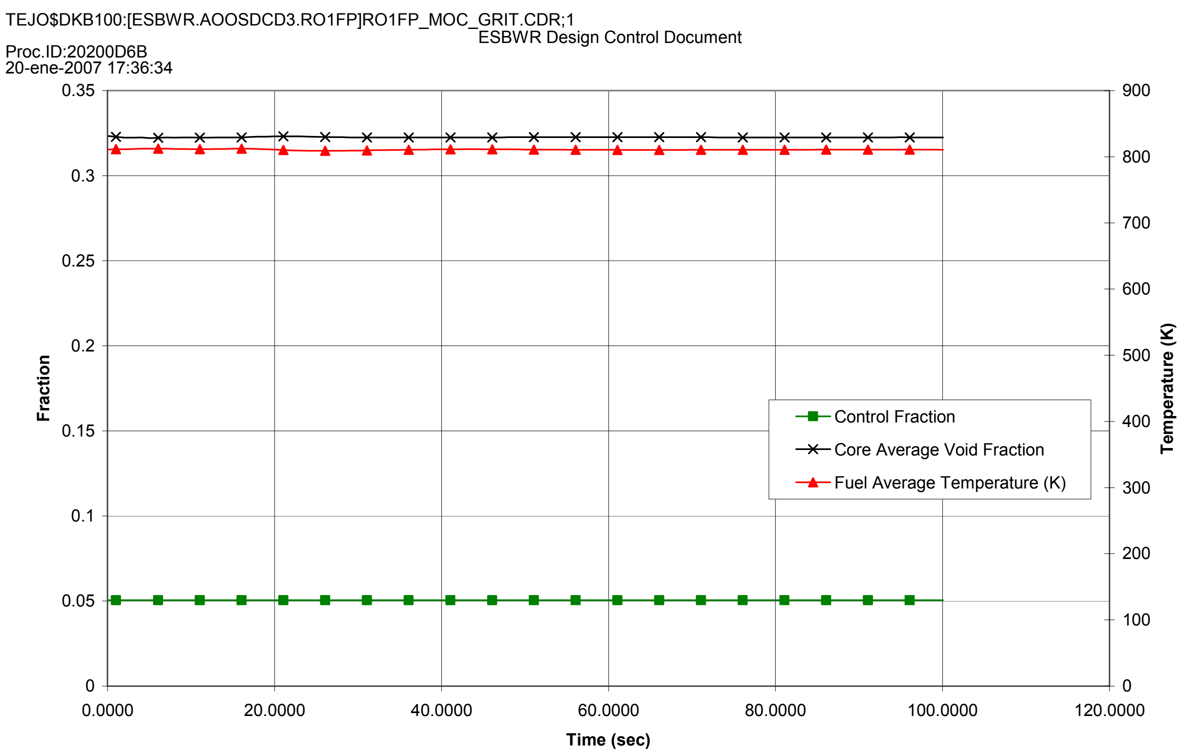


Figure 15.2-13f. Runout of One Feedwater Pump

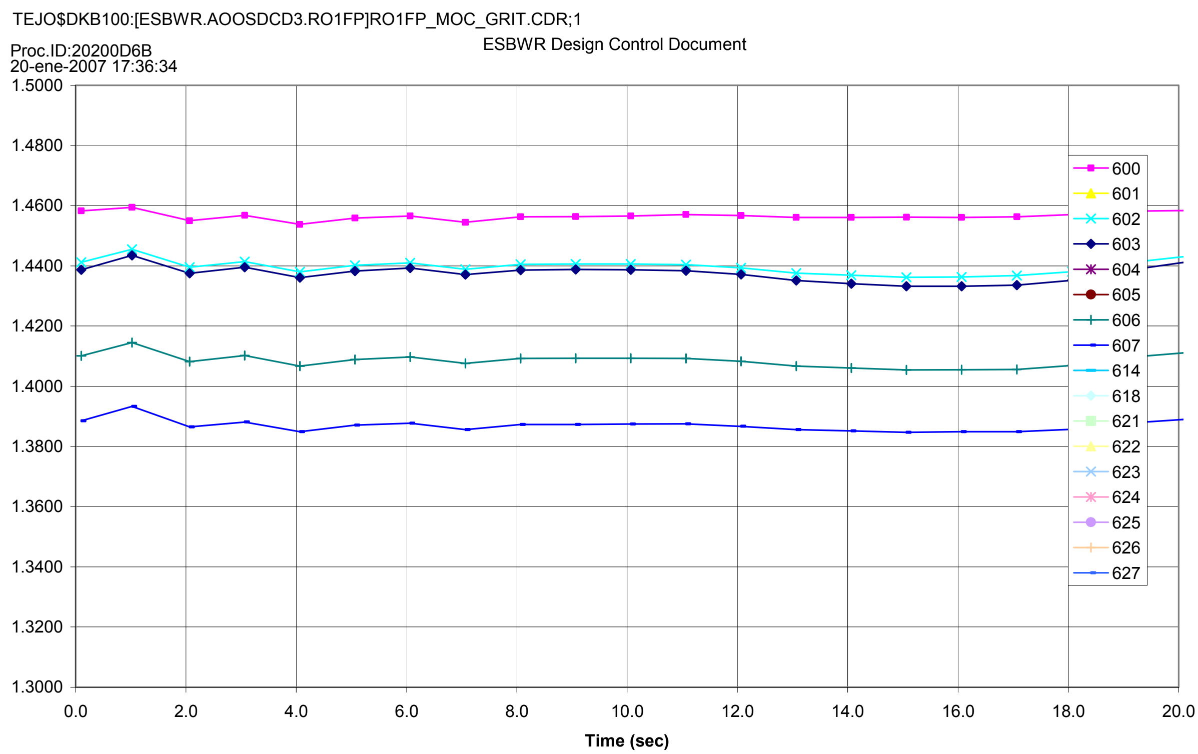


Figure 15.2-13g. Runout of One Feedwater Pump

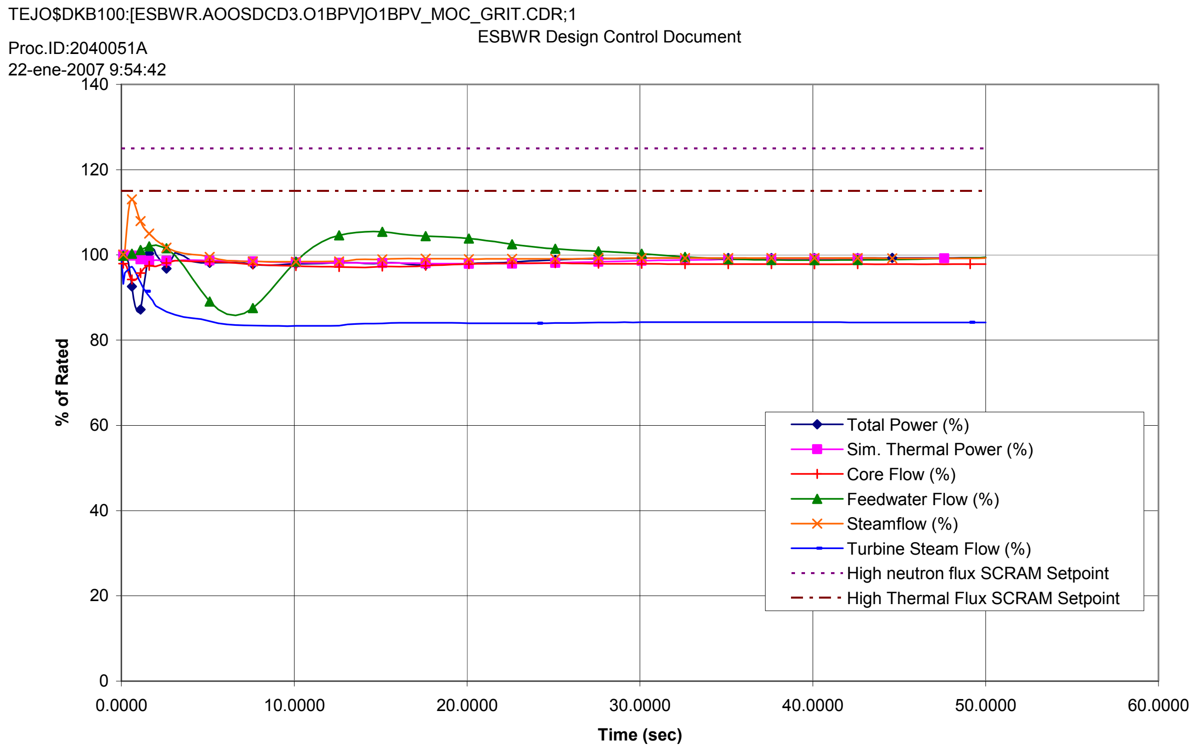


Figure 15.2-14a. Opening of One Turbine Control or Bypass Valve

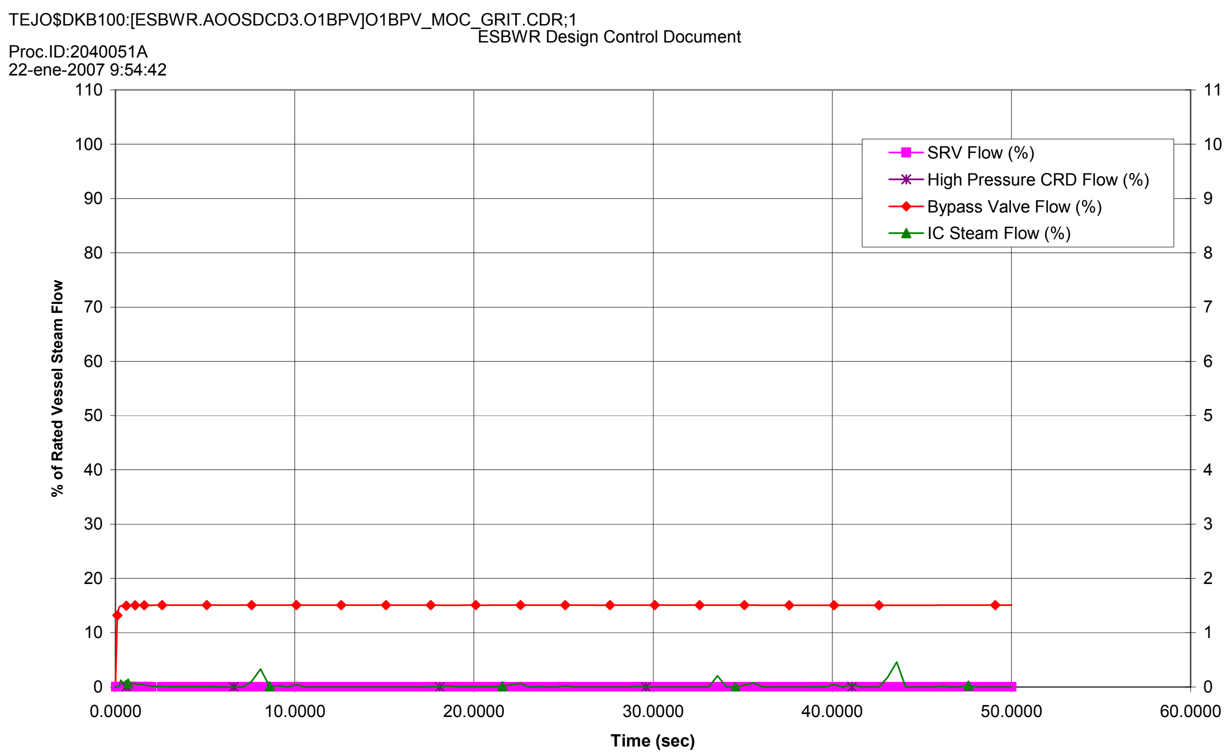


Figure 15.2-14b. Opening of One Turbine Control or Bypass Valve

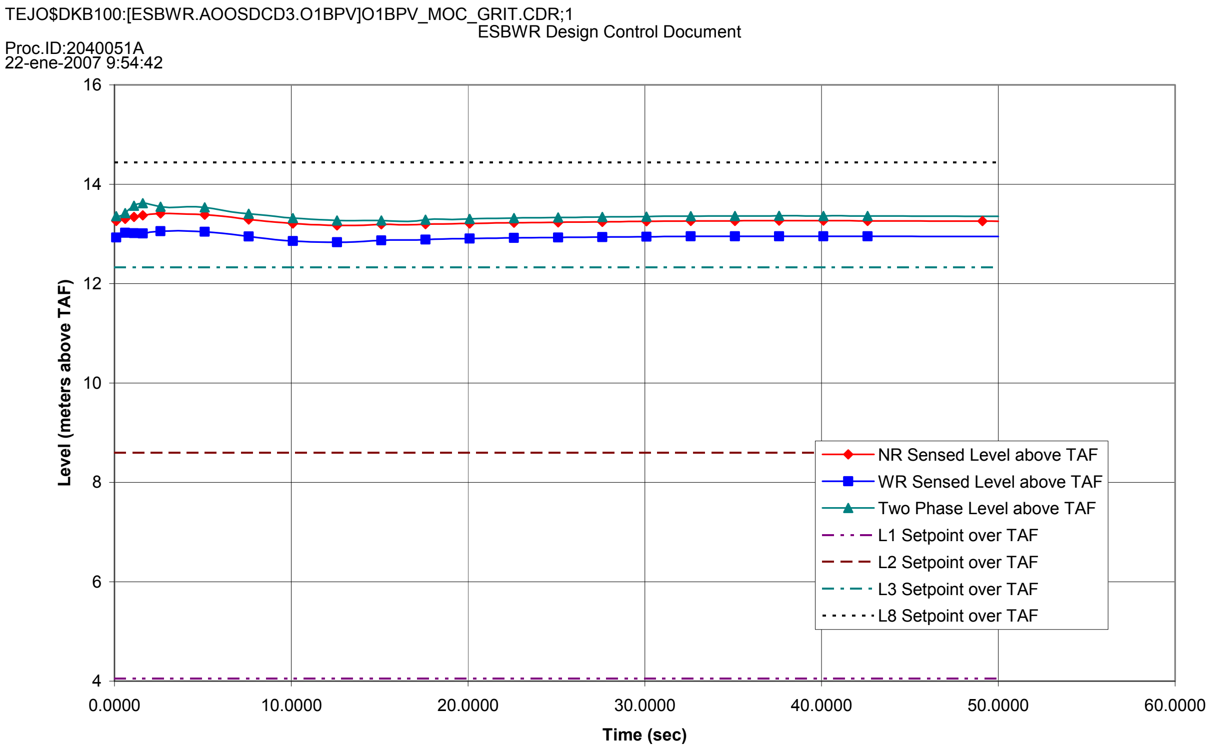


Figure 15.2-14c. Opening of One Turbine Control or Bypass Valve

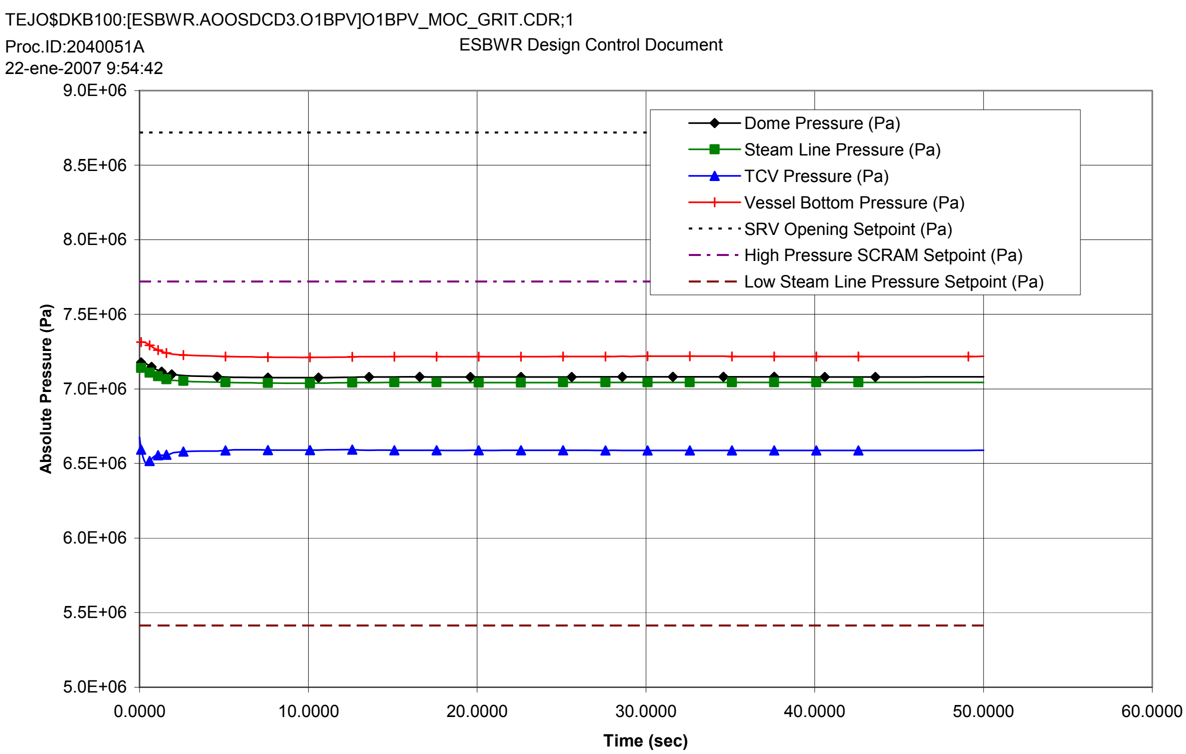


Figure 15.2-14d. Opening of One Turbine Control or Bypass Valve

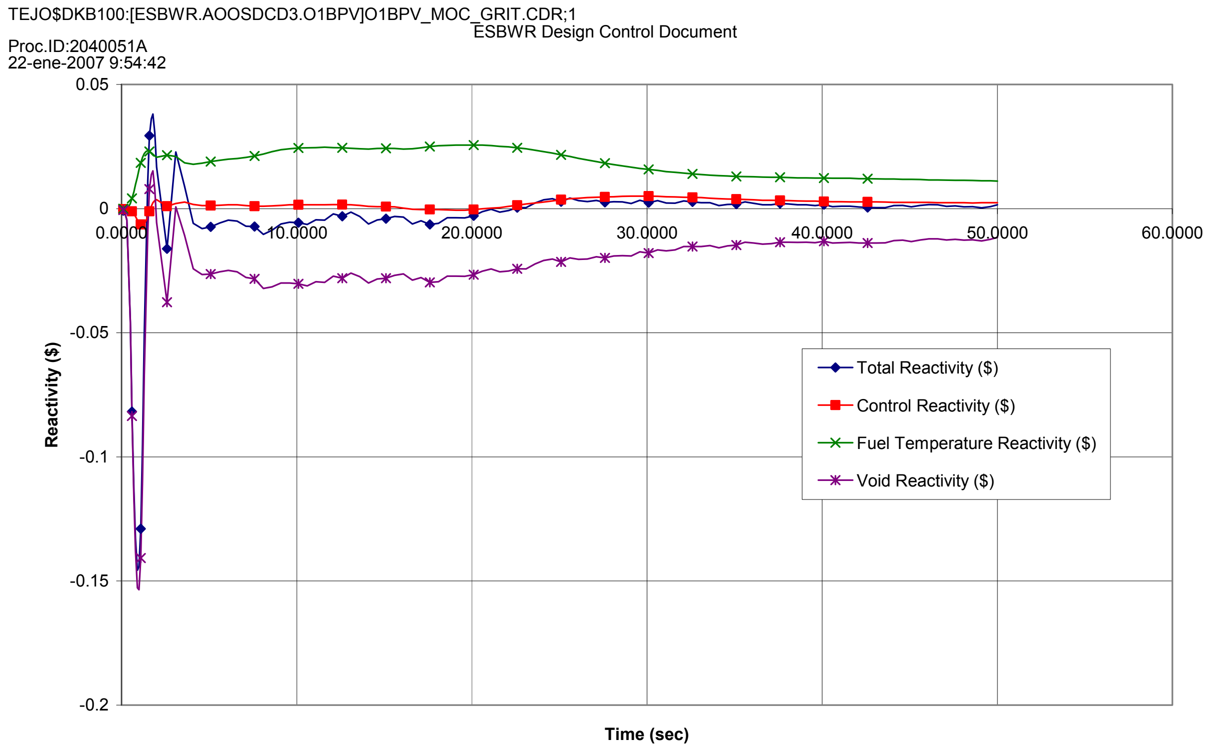


Figure 15.2-14e. Opening of One Turbine Control or Bypass Valve

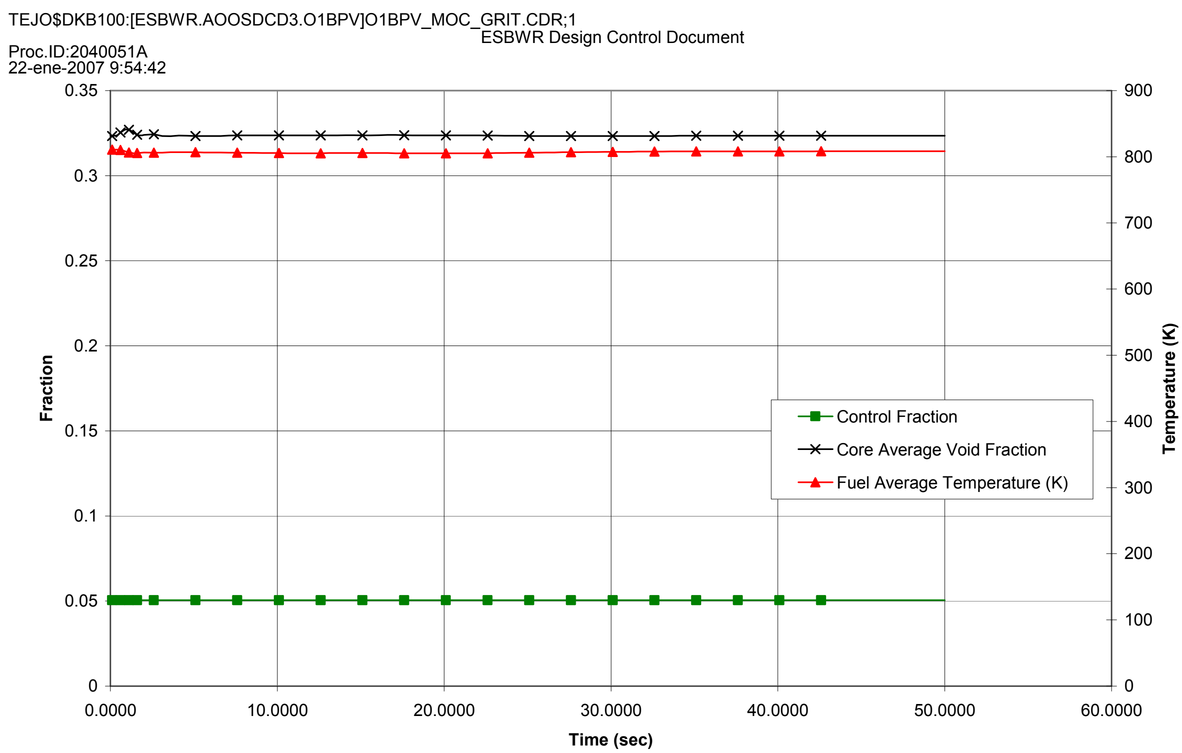


Figure 15.2-14f. Opening of One Turbine Control or Bypass Valve

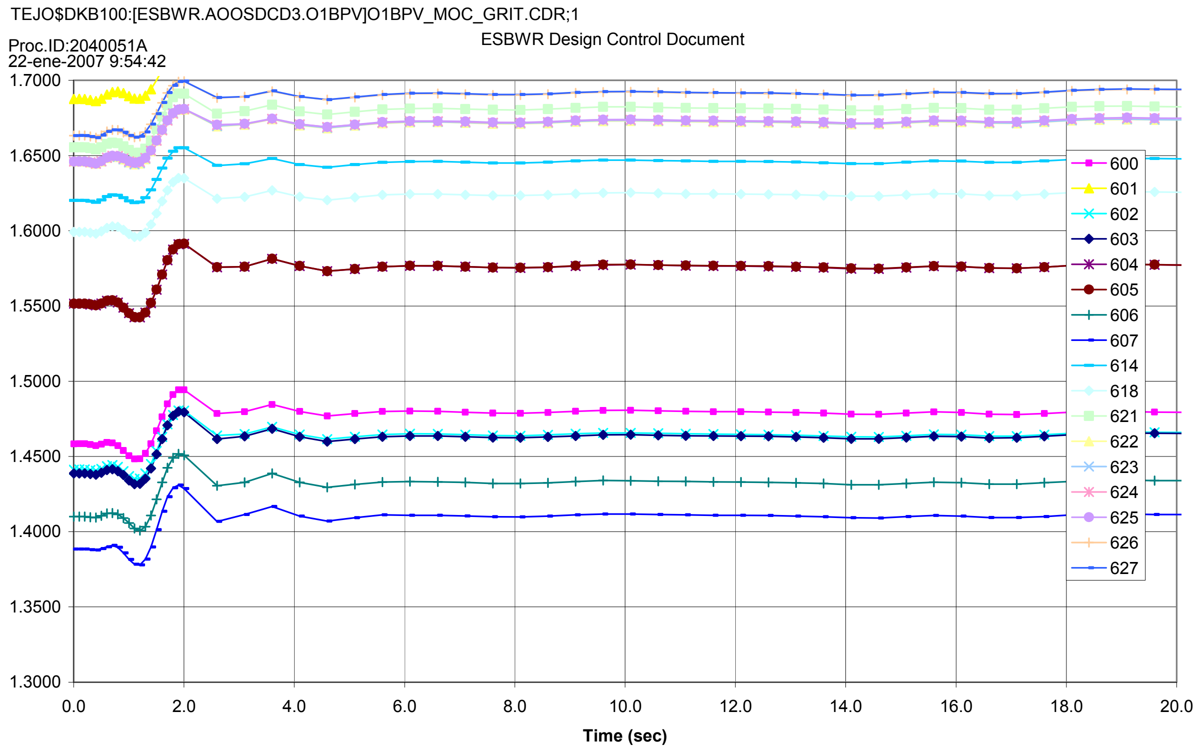


Figure 15.2-14g. Opening of One Turbine Control or Bypass Valve

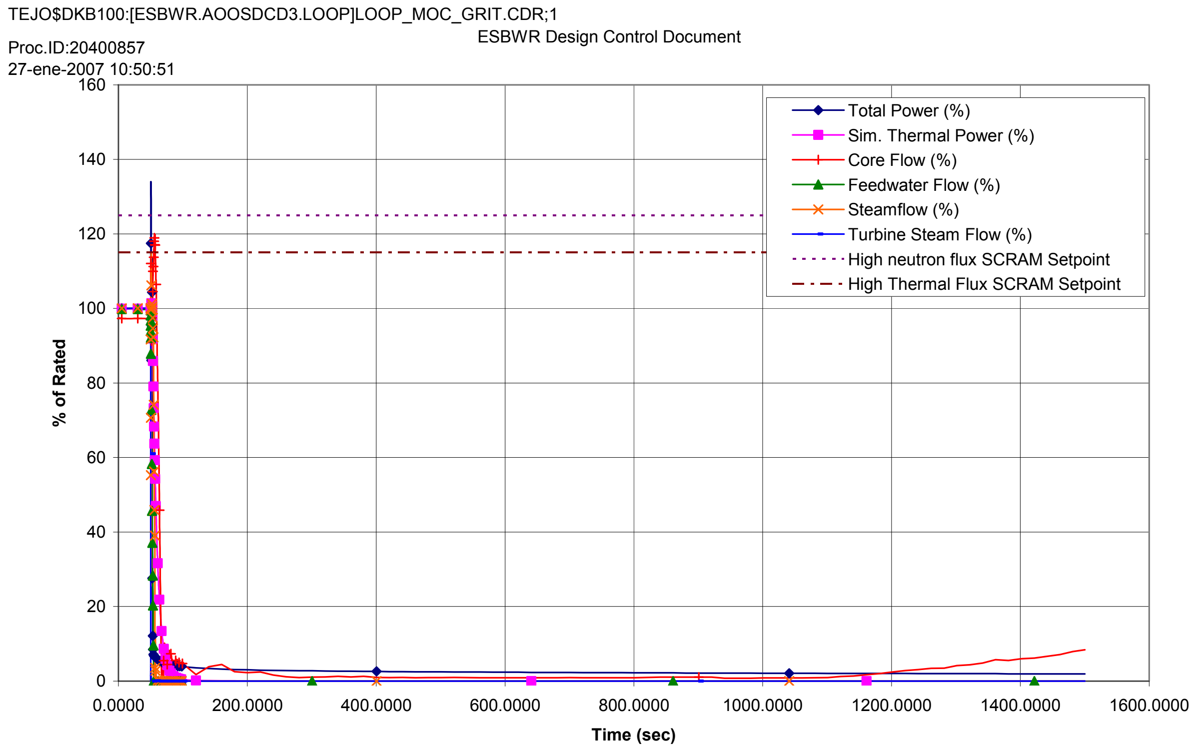


Figure 15.2-15a. Loss of Non-Emergency AC Power to Station Auxiliaries

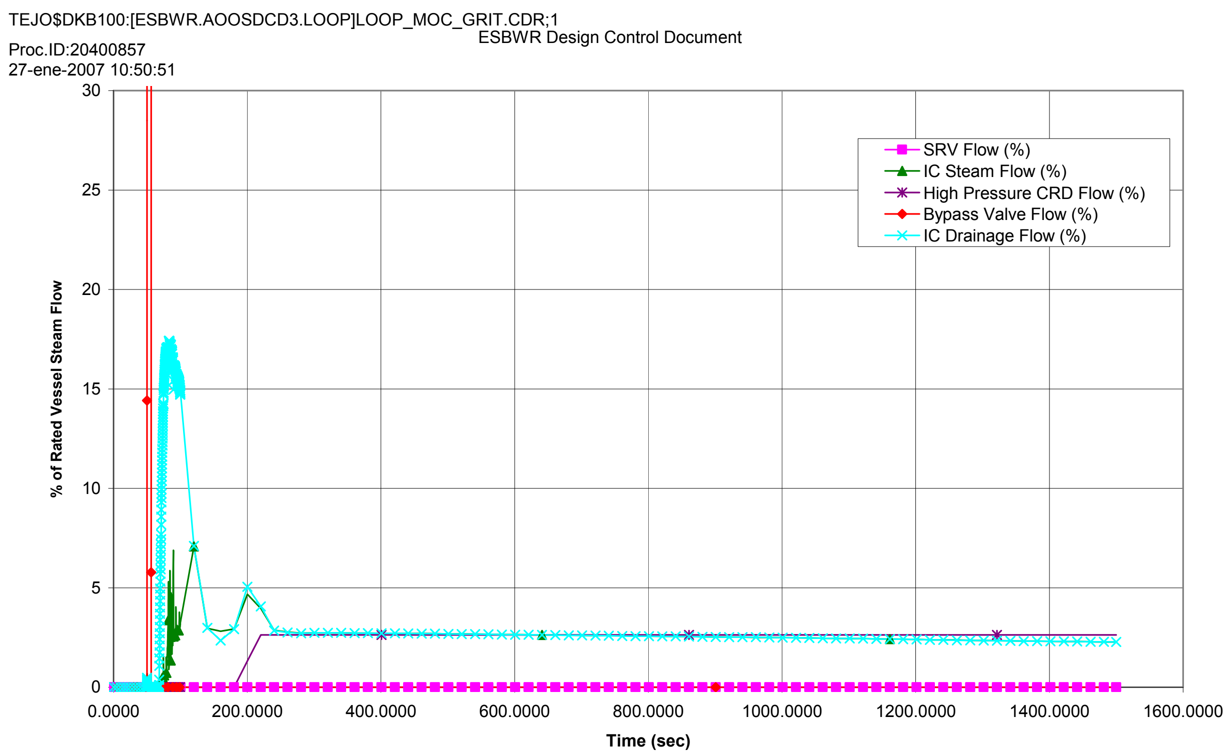


Figure 15.2-15b. Loss of Non-Emergency AC Power to Station Auxiliaries

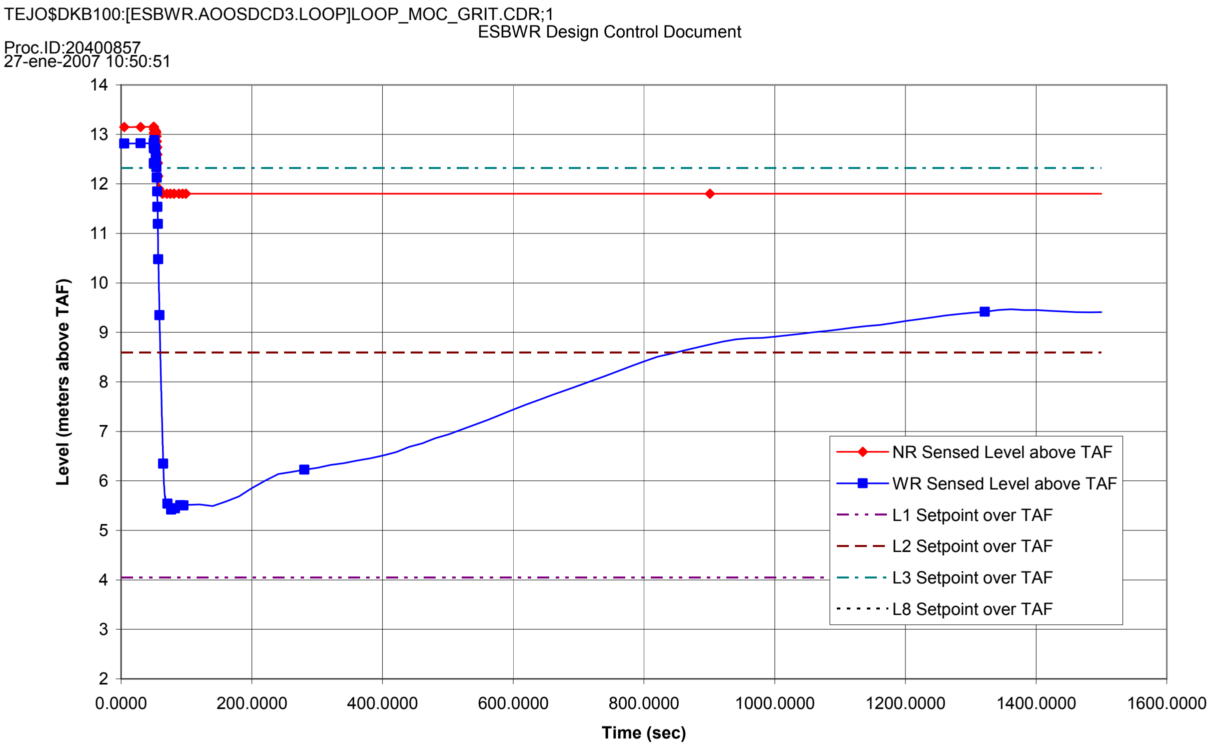


Figure 15.2-15c. Loss of Non-Emergency AC Power to Station Auxiliaries

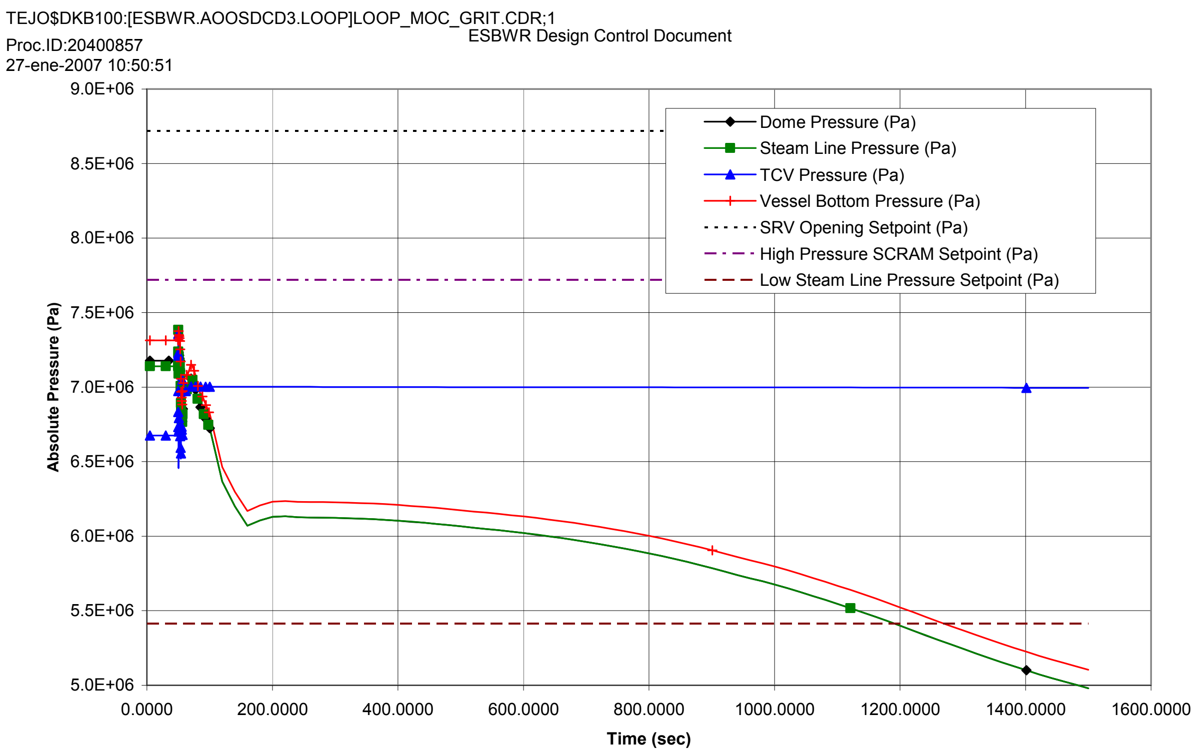


Figure 15.2-15d. Loss of Non-Emergency AC Power to Station Auxiliaries

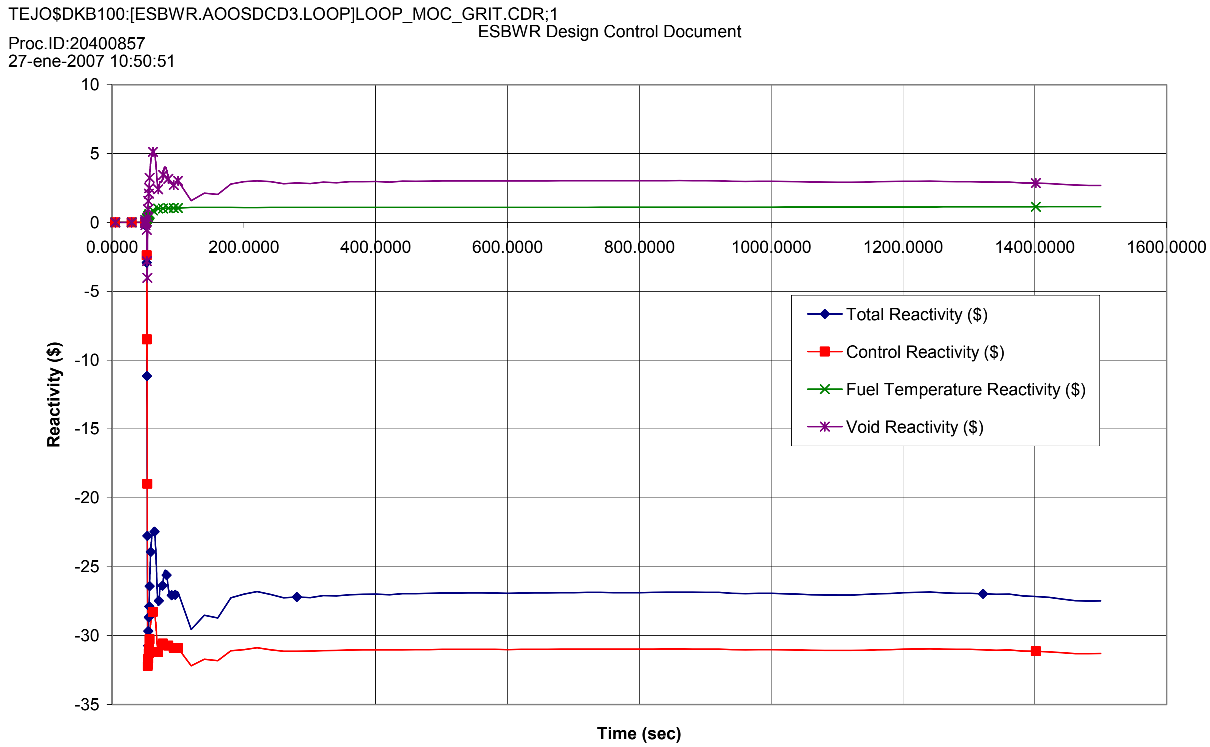


Figure 15.2-15e. Loss of Non-Emergency AC Power to Station Auxiliaries

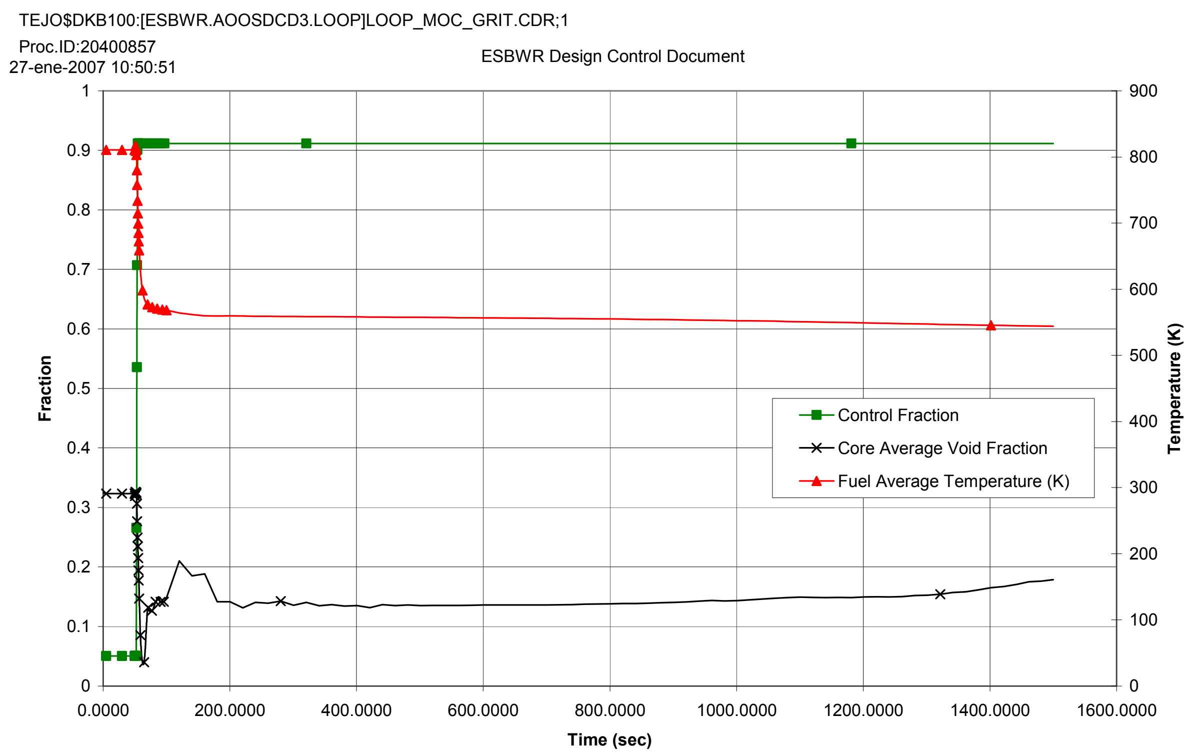


Figure 15.2-15f. Loss of Non-Emergency AC Power to Station Auxiliaries

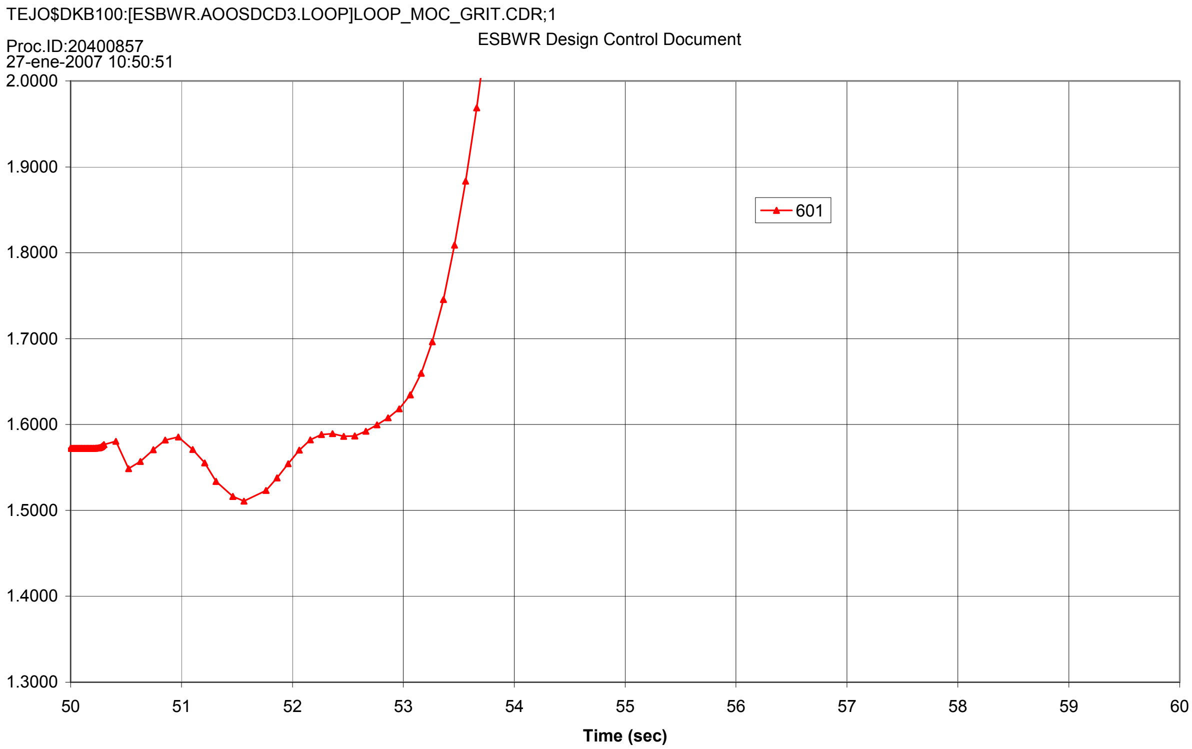


Figure 15.2-15g. Loss of Non-Emergency AC Power to Station Auxiliaries

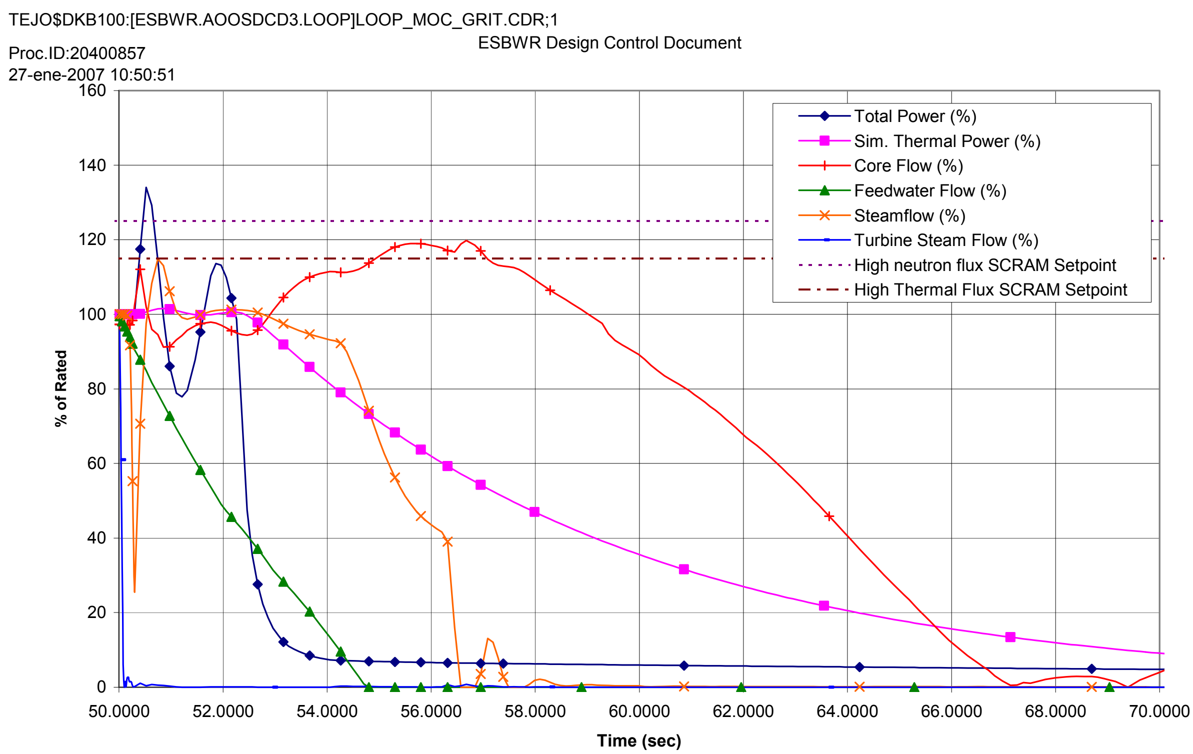


Figure 15.2-15h. Loss of Non-Emergency AC Power to Station Auxiliaries (Figure 15.2-15a from 50 to 70 s)

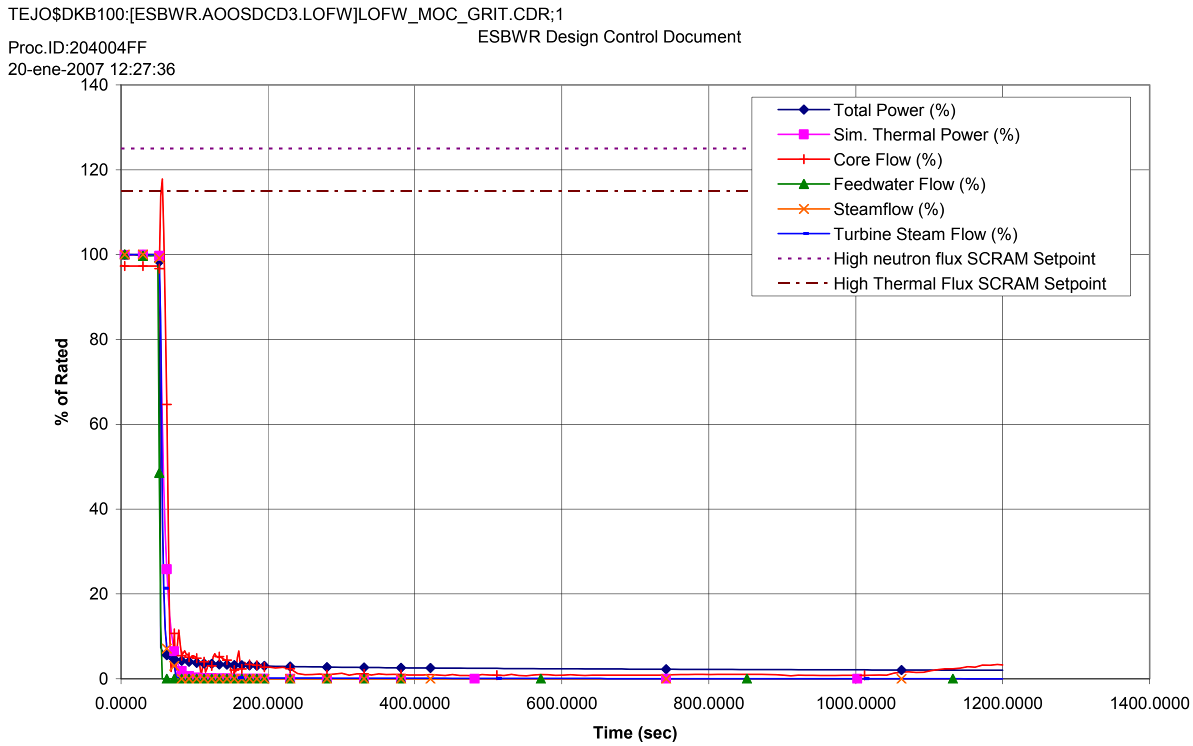


Figure 15.2-16a. Loss of All Feedwater Flow

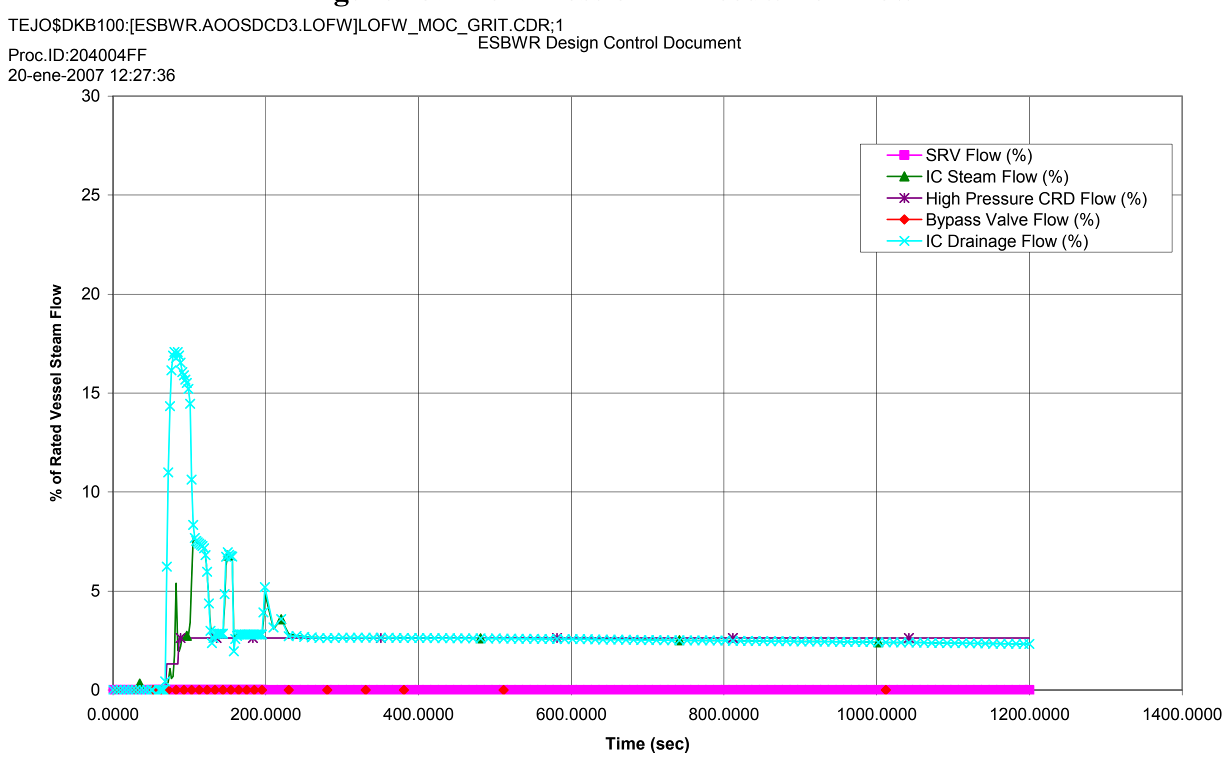


Figure 15.2-16b. Loss of All Feedwater Flow

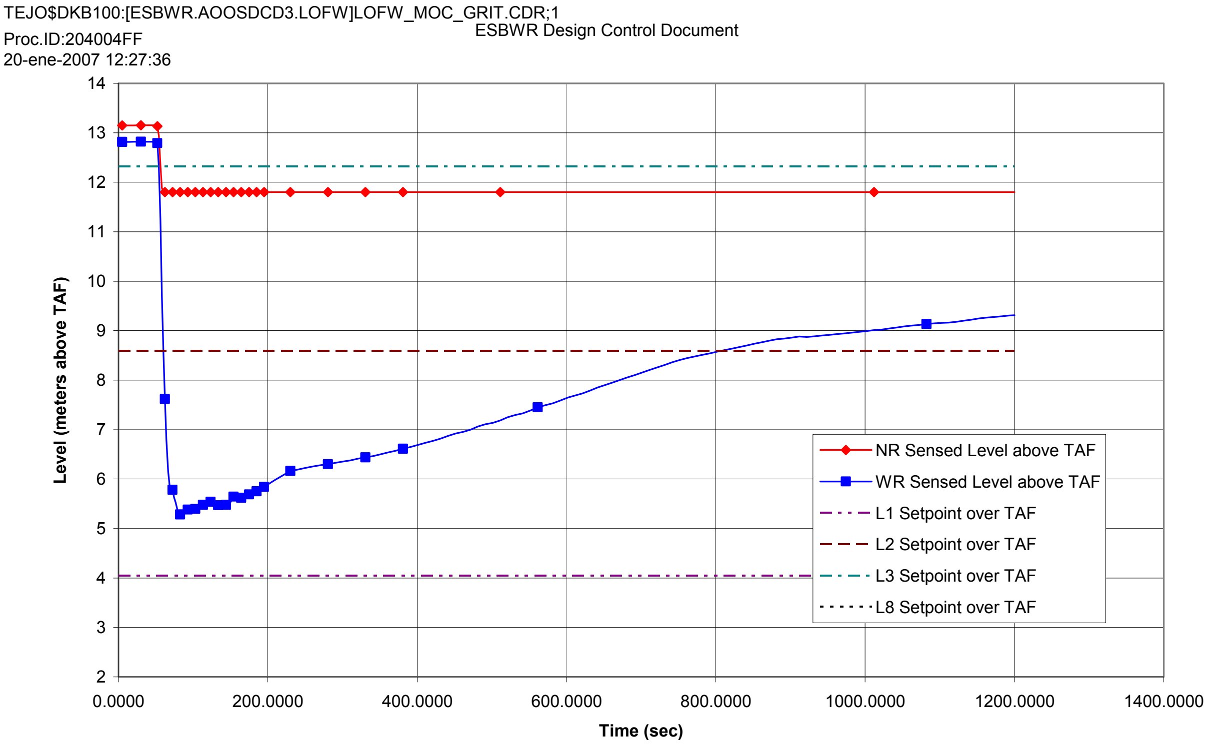


Figure 15.2-16c. Loss of All Feedwater Flow

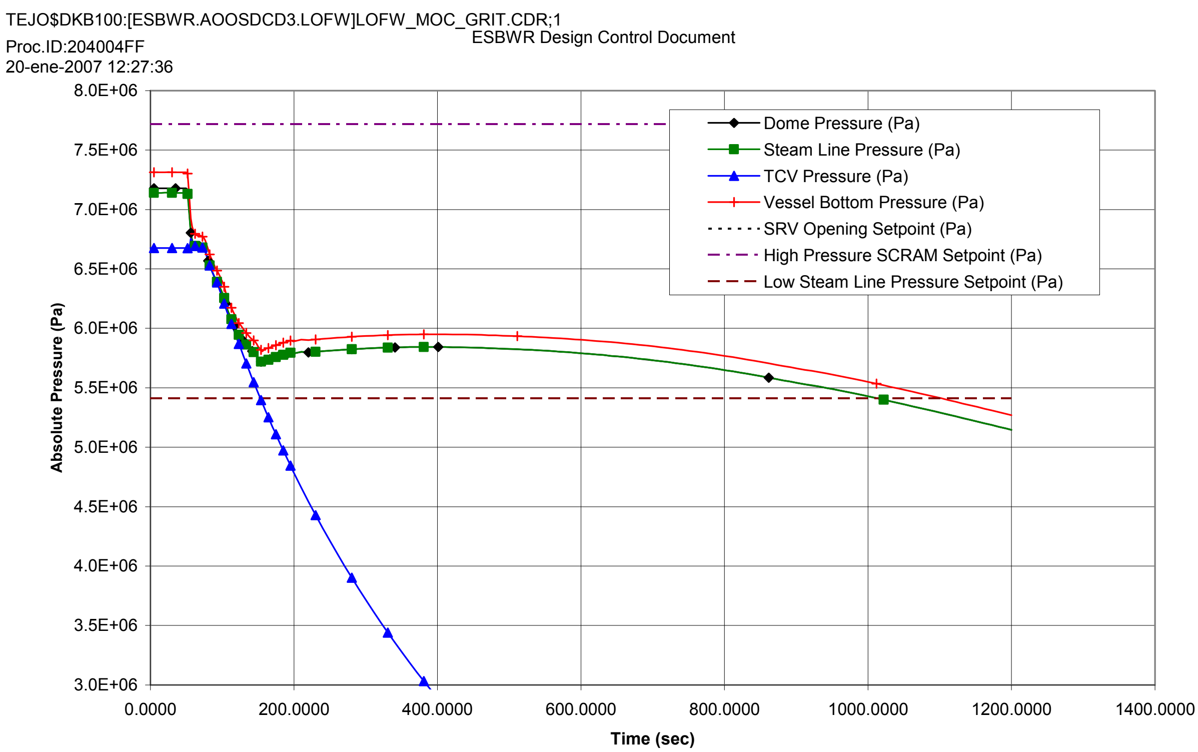


Figure 15.2-16d. Loss of All Feedwater Flow

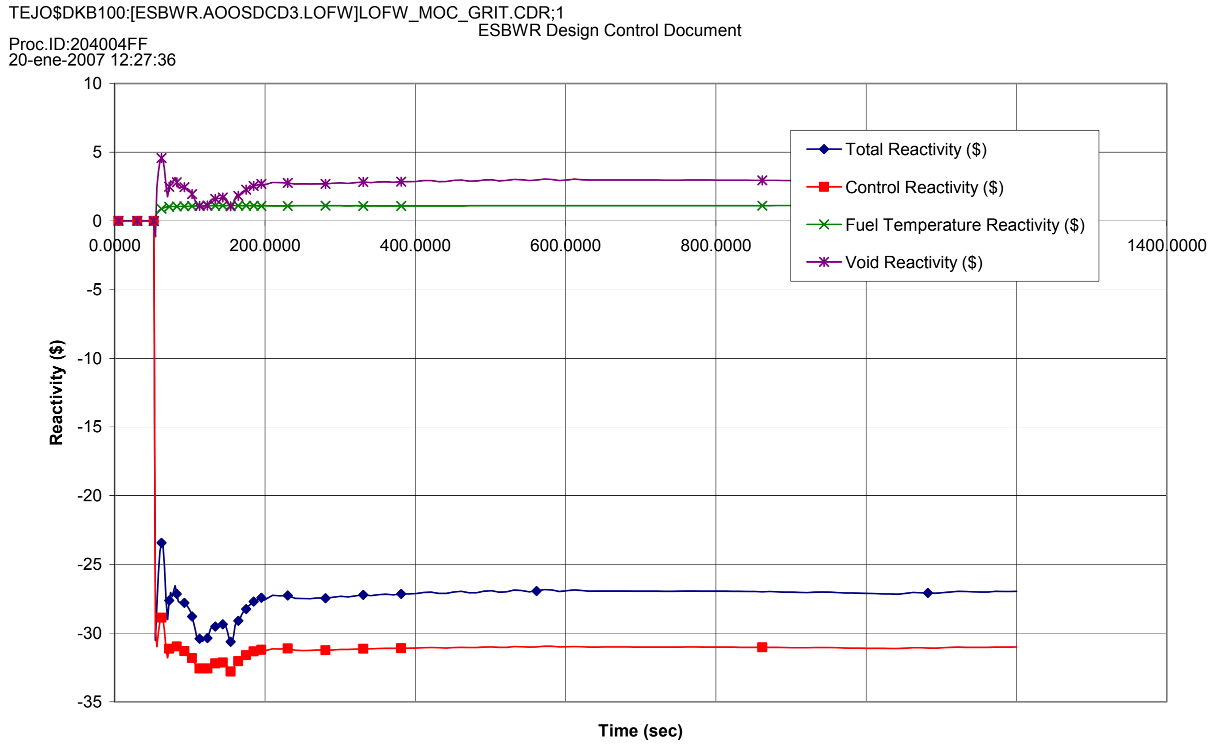


Figure 15.2-16e. Loss of All Feedwater Flow

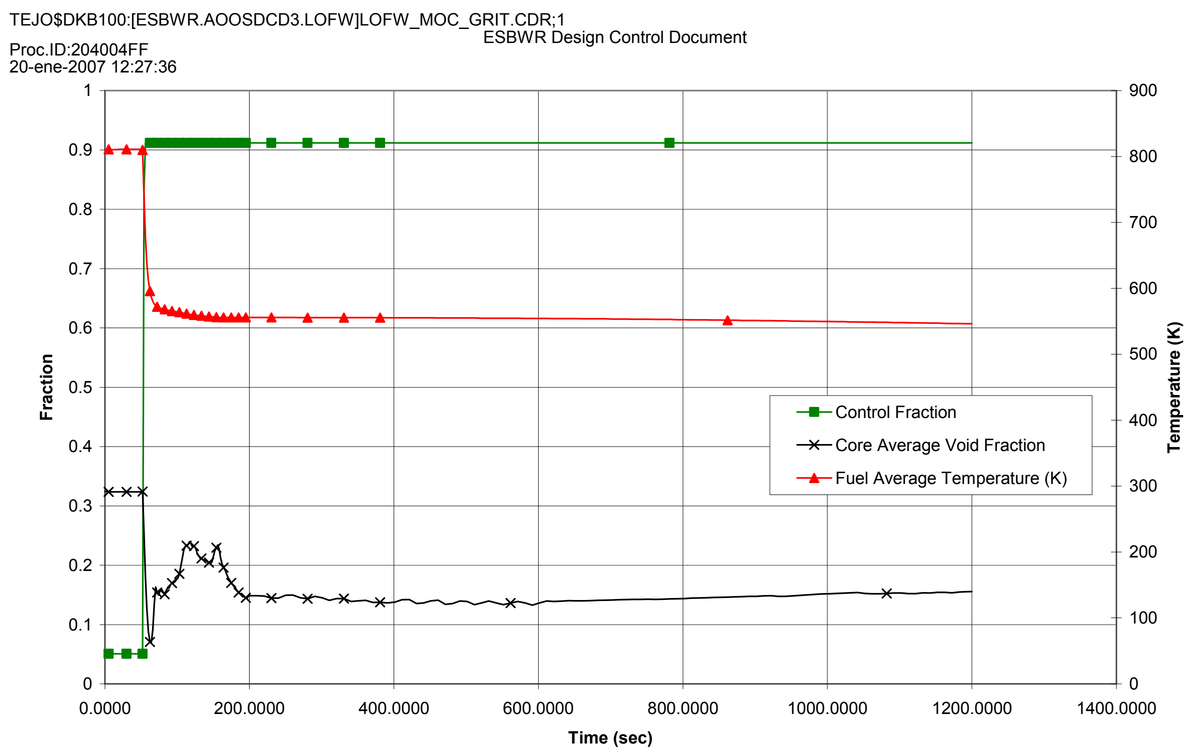


Figure 15.2-16f. Loss of All Feedwater Flow

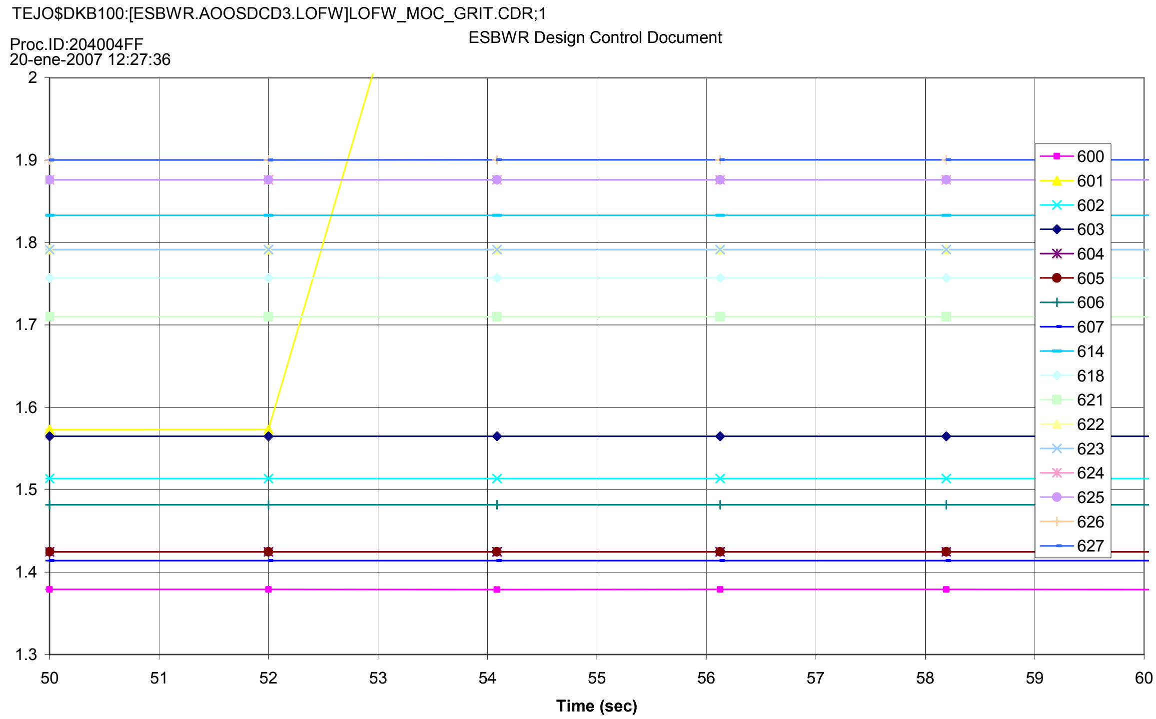


Figure 15.2-16g. Loss of All Feedwater Flow

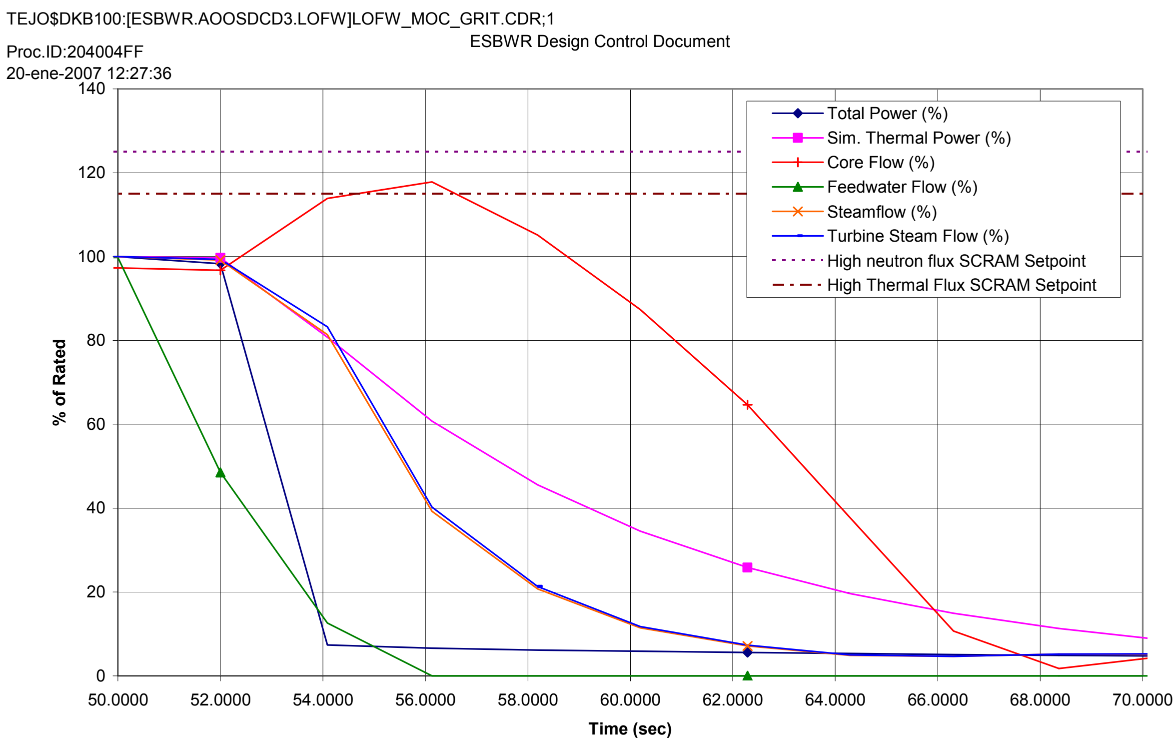


Figure 15.2-16h. Loss of All Feedwater Flow (Figure 15.2-16a from 50 to 70 s)

15.3 ANALYSIS OF INFREQUENT EVENTS

Appendix 15A provides a determination of event frequency to categorize AOOs as defined in 10 CFR 50 Appendix A, or Infrequent Events. Section 15.0 describes the licensing basis for this categorization.

The input parameters, initial conditions, and assumptions in Tables 15.2-1, 2 and 3 are applied in the TRACG calculations, based on the equilibrium core in Reference 15.3-4, for the Infrequent Events addressed in Subsections 15.3.1 through 15.3.6 and Subsections 15.3.13 and 15.3.15. The summary of the Infrequent Events analyses is given in Table 15.3-1.

The results of the system response analyses for the initial core design documented in Reference 15.3-5 are provided in Reference 15.3-6. System response analyses bounding operation in the feedwater temperature operating domain is documented in Reference 15.3-7.

15.3.1 Loss Of Feedwater Heating With Failure of Selected Control Rod Run-In

15.3.1.1 Identification of Causes

The loss of a feedwater (FW) heater can occur in at least two ways:

- Steam extraction line to heater is closed; or
- FW is bypassed around heater.

The first case produces a gradual FW cooling. In the second case, the FW bypasses the heater and no FW heating occurs. In either case, the reactor vessel receives colder FW. The maximum number of FW heaters that can be tripped or bypassed by a single event represents the most severe event for analysis considerations.

The ESBWR is designed such that no single operator error or equipment failure shall cause a loss of more than 55.6°C (100°F) FW heating.

This event conservatively assumes the loss of FW heating as shown on Table 15.2-1, causing an increase in core inlet subcooling and core power due to the negative void reactivity coefficient. However, the power increase is slow.

The Feedwater Control System (FWCS) logic is provided in Subsection 7.7.3, and includes logic provided to mitigate the effects of a loss of FW heating capability. The FWCS is constantly monitoring the actual FW temperature and comparing it with a reference temperature. When a loss of FW heating is detected [i.e., when the difference between the actual and reference temperatures exceeds a ΔT setpoint], the FWCS sends an alarm to the operator and sends a signal to the Rod Control and Information System (RC&IS) to initiate the Selected Control Rods Run-In and Select Rod Insertion (SCRRI/SRI) function to automatically reduce the reactor power and avoid a scram. However, for this event, SCRRI/SRI is assumed to fail and reactor scram on high simulated thermal power is not credited due to uncertainties. Therefore a new steady state is reached.

The frequency of this event is evaluated in Subsection 15A.3.6.

15.3.1.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.3-2 lists the sequence of events for Figure 15.3-1.

Systems Operation

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

The high simulated thermal power trip (STPT) scram is the primary protection system trip in mitigating the effects of this event. However, credit was not taken for this scram to consider the possibility that, for a similar case with a somewhat lower loss of heating, the scram setpoint might not be reached, while the consequences would only be slightly less severe for this case than the event analyzed here.

15.3.1.3 Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by programming a change in FW enthalpy corresponding to the assumed loss in FW heating, shown in Table 15.2-1.

Results

Reactor scram should be initiated during this event. However, as explained above, credit for STPT scram was not taken. The nuclear system pressure does not significantly change during the event, and consequently, the RCPB is not threatened.

This event is potentially limiting with respect to the number of rods in boiling transition. The OLMCPR is established to the limiting event and documented in the COLR in accordance with Technical Specifications.

15.3.1.4 Barrier Performance

As noted previously, the effects of this event do not result in any temperature or pressure transient in excess of the criteria for which the pressure vessel or containment are designed. Therefore, these barriers maintain their integrity and function as designed. In this event, the number of fuel rods that enter transition boiling is bounded by 1000 rods. It was assumed that all rods entering transition boiling fail.

15.3.1.5 Radiological Consequences

A radiological analysis was performed for an event where 1000 fuel rods fail as a result of entering transition boiling.

The scenario considered is for the fission product release paths to the environment consists of the fission products traveling down the main steam lines, eventually reaching the condenser, where they leak from the condenser to the environment. This scenario is modeled after the BWR rod drop accident described in Regulatory Guide 1.183, Appendix C.

The source term for the event is defined in Tables 15.3-13, 15.3-14 and 15.3-15. As can be seen in Table 15.3-16, the off site dose for this event is less than 2.5 REM Total Effective Dose Equivalent (TEDE) assuming the bounding number (1000 rods) of fuel failures.

Assumptions Requiring Confirmation

Site parameter assumptions in the radiological analysis are confirmed in Subsection 2.0.1.

15.3.2 Feedwater Controller Failure – Maximum Demand

15.3.2.1 Identification of Causes

See Subsection 15.2.4.2. This event assumes multiple control system failures, to simultaneously increase the flow in multiple FW pumps to their maximum limit. The frequency of this event is evaluated in Subsection 15A.3.5.

15.3.2.2 Sequence of Events and Systems Operation

Sequence of Events

With excess FW flow, the water level rises to the high water level reference point (Level 8), at which time the FW pumps are run back, the main turbine is tripped and a scram is initiated. Table 15.3-3 lists the sequence of events for Figure 15.3-2. The figure shows the changes in important variables during this event.

Because Level 8 is located near the top of the separators, some moisture entrainment and carry-over to the turbine and bypass valve may occur. While this is potentially harmful to the turbine's integrity, it has no safety implications for the plant.

Identification of Operator Actions

The operator should:

- Follow the scram procedure.
- Observe that FW flow runback due to high water level has terminated the failure event.
- Switch the FW controller from auto to manual control to try to regain a correct output signal.
- Identify causes of the failure and report all key plant parameters during the event.

15.3.2.2.1 System Operation

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. Important system operational actions for this event are tripping of the main turbine, FW flow runback, and scram due to high water level (Level 8).

15.3.2.3 Core and System Performance

15.3.2.3.1 Input Parameters and Initial Conditions

The total FW flow for all pumps runout is provided in Table 15.2-1.

15.3.2.3.2 Results

The simulated runout of all FW pumps is shown in Figure 15.3-2. The high water level turbine trip and FW pump runback are initiated early in the event as shown in Table 15.3-3. Scram occurs and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. The Turbine Bypass System (TBS) opens to limit peak pressure in the steamline near the SRVs and the peak pressure at the bottom of the vessel. The peak pressure in the bottom of the vessel remains below the ASME code upset limit. Peak steam line pressure near the SRVs remains below the setpoint of the SRVs.

The water level gradually drops, and can reach the Low Level reference point (Level 2), activating the IC system for long-term level control and the HP_CRD system to permit a slow recovery to the desired level.

This event is reanalyzed for each specific initial core configuration.

15.3.2.4 Barrier Performance

As previously noted, the effect of this event does not result in any temperature or pressure transient in excess of the criteria for which the pressure vessel or containment are designed. Therefore, these barriers maintain their integrity and function as designed. In this event, there are no fuel rods that enter transition boiling.

15.3.2.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.3.3 Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

15.3.3.1 Identification of Causes

The ESBWR Steam Bypass and Pressure Control (SB&PC) system uses a triplicated digital control system. The SB&PC system controls the turbine control valves and turbine bypass valves to maintain reactor pressure. As presented in Subsection 15.2.4.2, no credible single failure in the control system results in a maximum demand to all actuators for all turbine control valves and bypass valves. A voter or actuator failure may result in an inadvertent opening of one turbine control valve or one turbine bypass valve. In this case, the SB&PC system senses the pressure change and commands the remaining control valves to close, and thereby automatically mitigates the transient and maintains reactor power and pressure.

As presented in Subsection 15.2.4.2, multiple failures might cause the SB&PC system to erroneously issue a maximum demand to all turbine control valves and bypass valves. Should this occur, all turbine control valves and bypass valves could be fully opened. However, the probability of this event is extremely low, and thus, the event is considered as a limiting fault. The frequency of this event is evaluated in Subsection 15A.3.1.

15.3.3.2 Sequence of Events and Systems Operation

15.3.3.2.1 Sequence of Events

Table 15.3-4 lists the sequence of events for Figure 15.3-3.

15.3.3.2.2 Identification of Operator Actions

If the reactor scrams as a result of the isolation caused by the low pressure at the turbine inlet in the run mode, the following sequence of operator actions is expected during the course of the event.

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Monitor reactor water level and pressure;
- Monitor turbine coastdown and break vacuum before the loss of steam seals (check turbine auxiliaries for proper operation);
- Observe that ICS is initiated on low-water level or MSIV closure;
- Use ICS to control reactor pressure and level;
- Cooldown the reactor per standard procedure if a restart is not intended; and
- Complete the scram report and initiate a maintenance survey of the SB&PC system before reactor restart.

15.3.3.2.3 Systems Operations

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, except as otherwise noted.

15.3.3.3 Core and System Performance

15.3.3.3.1 Input Parameters and Initial Condition

A five-second isolation valve closure (maximum isolation valve closing time plus instrument delay) instead of a three second closure is assumed when the turbine pressure decreases below the turbine inlet low-pressure setpoint for main steamline isolation initiation. This is within the specification limits of the valve and represents a conservative assumption.

15.3.3.3.2 Results

Figure 15.3-3 presents graphically how the low steam line pressure trips the isolation valve closure, stops vessel depressurization and produces a normal shutdown of the isolated reactor.

Depressurization results in formation of voids in the reactor coolant and causes a decrease in reactor power almost immediately. Position switches on the isolation valves initiate reactor scram.

The isolation limits the duration and severity of the depressurization so that no significant thermal stresses are imposed on the reactor coolant pressure boundary.

15.3.3.4 Barrier Performance

Barrier performance analyses were not required because 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. The peak pressure in the bottom of the vessel remains below its ASME code faulted limit for the RCPB. Peak steam line pressure near the SRVs remains below the setpoint of the SRVs.

15.3.3.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.3.4 Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

15.3.4.1 Identification of Causes

The ESBWR Steam Bypass and Pressure Control (SB&PC) system uses a triplicated digital control system, instead of an analog system as was originally supplied in BWR/2 through BWR/6 plants. This system is similar to the one used in the ABWR design. The SB&PC system controls turbine control valves and turbine bypass valves to maintain reactor pressure. As presented in Subsection 15.2.4.2, no credible single failure in the control system results in a minimum demand to all turbine control valves and bypass valves. A voter or actuator failure may result in an inadvertent closure of one turbine control valve or one turbine bypass valve if it is open at the time of failure. In this case, the SB&PC system senses the pressure change and commands the remaining control valves or bypass valves, if needed, to open, and thereby automatically mitigates the transient and tries to maintain reactor power and pressure.

No single failure causes the SB&PC system to erroneously issue a minimum demand to all turbine control valves and bypass valves. However, as discussed in Subsection 15.2.4.2, multiple failures might cause the SB&PC system to fail and erroneously issue a minimum demand. Should this occur, it would cause full closure of turbine control valves as well as inhibition of steam bypass flow and thereby increase reactor power and pressure. When this occurs, reactor scram is initiated when the high reactor flux scram setpoint is reached. The SB&PC system design includes provision to mitigate the effects of this postulated multiple failure event. In the event of a detected failure of two channels of the triplicated control system, a turbine trip is automatically initiated. This event is analyzed here as the simultaneous undetected failure of two control processors, called "pressure regulator downscale failure." However, the probability of this event to occur is extremely low and hence the event is considered as an Infrequent Event rather than an AOO. The frequency of this event is evaluated in Subsection 15A.3.2.

15.3.4.2 Sequence of Events and Systems Operation

15.3.4.2.1 Sequence of Events

Table 15.3-5 lists the sequence of events for Figure 15.3-4.

15.3.4.2.2 Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Monitor reactor water level and pressure;
- Monitor turbine coastdown and break vacuum before the loss of steam seals (check turbine auxiliaries for proper operation);
- Cool down the reactor per standard procedure if a restart is not intended; and
- Complete the scram report and initiate a maintenance survey of pressure regulator before reactor restart.

15.3.4.2.3 Systems Operation

Except for the failures in the SB&PC system, normal plant instrumentation and controls and plant protection and reactor protection systems are assumed to function normally. Specifically, this event takes credit for high neutron flux scram to shut down the reactor.

The turbine control valves, in their servo mode, have a full stroke closure time, from fully open to fully closed, from 2.5 seconds to 3.5 seconds. The worst case of 2.5 seconds is assumed in the analysis.

15.3.4.3 Core and System Performance

A pressure regulator downscale failure is simulated as shown in Figure 15.3-4.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. When the sensed neutron flux reaches the high neutron flux scram setpoint, a reactor scram is initiated. The neutron flux and pressure increase is limited by the reactor scram.

15.3.4.4 Barrier Performance

The peak pressure in the bottom of the vessel remains below the ASME code limit for the RCPB. The peak vessel bottom pressure is below its ASME Code faulted pressure limit. The peak pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. In this event, there are no fuel rods that enter transition boiling.

15.3.4.5 Radiological Consequences

A radiological analysis was performed for an event where 1000 fuel rods fail as a result of entering transition boiling. It is described in detail in Subsection 15.3.1.5.

The off site dose for this event is less than 2.5 REM Total Effective Dose Equivalent (TEDE) assuming the bounding number (1000 rods) of fuel failures.

15.3.5 Generator Load Rejection With Total Turbine Bypass Failure

15.3.5.1 Identification of Causes

Fast closure of the turbine control 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 (TG) rotor. Closure of the TCVs causes a sudden reduction in steam flow that results in an increase in system pressure and reactor shutdown.

After sensing a significant loss of electrical load on the generator, the TCVs are commanded to close rapidly. At the same time, the turbine bypass valves are signaled to open in the "fast" opening mode by the SB&PC system, which uses a triplicated digital controller. As presented in Subsection 15.2.4.2, no single failure can cause all turbine bypass valves to fail to open on demand. The worst single failure can only cause one turbine bypass valve to fail to open on demand. The frequency of this event is evaluated in Subsection 15A.3.4.

15.3.5.2 Sequence of Events and System Operation

15.3.5.2.1 Sequence of Events

A loss of generator electrical load at high power with failure of all bypass valves produces the sequence of events listed in Table 15.3-6a.

15.3.5.2.2 Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Verify proper bypass valve performance;
- Observe that the feedwater/level controls have maintained the reactor water level at a satisfactory value;
- Observe that the pressure regulator is controlling reactor pressure at the desired value; and
- Observe reactor peak power and pressure.

15.3.5.2.3 Systems Operation

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 unless stated otherwise.

All plant control systems maintain normal operation unless specifically designated to the contrary, except that failure of all turbine bypass valves is assumed for the entire event.

15.3.5.3 Core and System Performance

15.3.5.3.1 Input Parameters and Initial Conditions

The turbine electrohydraulic control system (EHC) detects load rejection before a measurable turbine speed change takes place.

The closure characteristics of the TCVs are assumed conservatively such that the valves operate in the full arc (FA) mode. The TCVs have a full stroke closure time, from fully open to fully closed, from 0.15 seconds to 0.20 seconds. The worst case value (see Table 15.3-6a) is assumed in the analysis.

The pressurization and/or the reactor scram may compress the water level to the low level trip setpoint (Level 2) and initiate the CRD high pressure makeup function, MSIV closure, and isolation condensers. Should this occur, it would follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred.

15.3.5.3.2 Results

The results are shown in Figure 15.3-5.

This event is potentially limiting with respect to the number of rods in boiling transition. The OLMCPR is established to the limiting event and documented in the COLR in accordance with Technical Specifications.

15.3.5.4 Barrier Performance

Peak pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak vessel bottom pressure remains below its ASME code faulted pressure limit. In this event, the number of fuel rods that enter transition boiling is bounded by 1000 rods. It was assumed that all rods entering transition boiling fail.

15.3.5.5 Radiological Consequences

A radiological analysis was performed for an event where 1000 fuel rods fail as a result of entering transition boiling. It is described in detail in Subsection 15.3.1.5.

The off site dose for this event is less than 2.5 REM TEDE assuming the bounding number (1000 rods) of fuel failures.

15.3.6 Turbine Trip With Total Turbine Bypass Failure

15.3.6.1 Identification of Causes

A variety of turbine or nuclear system malfunctions initiate turbine trips. Some examples are high velocity separator drain tank high levels, large vibrations, operator lockout, loss of control fluid pressure, low condenser vacuum, and reactor high water level.

After the main turbine is tripped, turbine bypass valves are opened in their fast opening mode by the SB&PC system. As presented in Subsection 15.2.4.2, any single failures can only cause one bypass valve to fail to open on demand. Only multiple failures can cause all bypass valves fail to open on demand. The frequency of this event is evaluated in Subsection 15A.3.3.

15.3.6.2 Sequence of Events and System Operation

15.3.6.2.1 Sequence of Events

Turbine trip at high power with failure of all bypass valves produces the sequence of events listed in Table 15.3-7.

15.3.6.2.2 Identification of Operator Actions

The operator should:

- Verify that all rods are inserted;
- Follow the scram procedure;
- Verify that the generator breaker is automatically opened to allow electrical buses originally supplied by the generator to be supplied by the incoming power;
- Monitor reactor water level and pressure;
- Check turbine for proper operation of all auxiliaries during coastdown;
- Manually initiate ICS, if necessary, to control reactor pressure;
- Depending on conditions, maintain pressure for restart purposes, or initiate normal operating procedures for cooldown;
- Put the mode switch in the startup position before the reactor pressure decays to below 6 MPa;
- Cool down the reactor per standard procedure if a restart is not intended; and
- Investigate the cause of the trip, make repairs as necessary, and complete the scram report

15.3.6.2.3 Systems Operation

All plant control systems maintain normal operation unless specifically designated to the contrary, except that failure of all main turbine bypass valves is assumed for the entire transient time period analyzed. Credit is taken for successful operation of the RPS.

15.3.6.3 Core and System Performance

15.3.6.3.1 Input Parameters and Initial Conditions

Turbine stop valves full stroke closure time is in the range of 0.10 second to 0.15 second. The worst case (see Table 15.3-7) is assumed in the analysis. A reactor scram is initiated by the turbine stop valve position switch, and after the confirmation of no availability of the turbine bypass.

15.3.6.3.2 Results

A turbine trip with failure of the bypass system is simulated at 100% Nuclear Boiler Rated (NBR) power conditions in Figure 15.3-6.

The severity of this transient is similar to the generator load rejection with failure of bypass event presented in Subsection 15.3.5. This event does not have to be reanalyzed for a specific core configuration.

15.3.6.4 Barrier Performance

Peak pressure at the SRVs is below the SRV setpoints. Therefore, there is no steam discharged to the suppression pool. The peak pressure at the vessel bottom remains below its ASME Code faulted pressure limit. In this event, the number of fuel rods which enter transition boiling is bounded by 1000 rods. It was assumed that all rods entering transition boiling fail.

15.3.6.5 Radiological Consequences

A radiological analysis was performed for an event where 1000 fuel rods fail as a result of entering transition boiling (described in detail in Subsection 15.3.1.5).

The off site dose for this event is less than 2.5 REM Total Effective Dose Equivalent (TEDE) assuming the bounding number (1000 rods) of fuel failures.

15.3.7 Control Rod Withdrawal Error During Refueling

15.3.7.1 Identification of Causes

The event considered is inadvertent criticality due to the complete withdrawal or removal of the most reactive rod (or the most reactive pair of rods associated with the same HCU) during refueling. The probability of the initial causes, alone, is considered low enough to warrant being categorized as an infrequent incident, because there is no postulated set of circumstances that results in an inadvertent control rod withdrawal error while in the REFUEL mode. The frequency of this event is evaluated in Subsection 15A.3.11.

15.3.7.2 Sequence of Events and Systems Operation

Initial Control Rod Removal or Withdrawal

During refueling operation, system interlocks provide assurance that inadvertent criticality does not occur because a control rod (or a pair of control rods associated with the same HCU) has been removed or is withdrawn.

Fuel Insertion with Control Rod Withdrawn

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

Second Control Rod Removal or Withdrawal

When the platform is not over the core (or fuel is not on the hoist) and the mode switch is in the REFUEL position, only one control rod can be withdrawn when the RC&IS SINGLE/GANG rod selection status is in the SINGLE rod selection mode. When the RC&IS SINGLE/GANG rod selection status is in the GANG rod selection mode, only one control rod pair with the same HCU may be withdrawn. The RC&IS Single/Dual Rod Sequence Restriction Override by-pass feature controls the movement of the control rods. Any attempt to withdraw an additional rod results in a rod block initiated by the RC&IS RAPI/RWM rod block logic. Because the core is designed to meet shutdown requirements with one control rod pair (with the same HCU) or one rod of maximum worth withdrawn, the core remains subcritical even with one rod or a rod pair associated with the same HCU withdrawn.

Control Rod Removal Without Fuel Removal

The design of the control rod, incorporating the bayonet coupling system does not physically permit the upward removal of the control rod without decoupling by rotation and the simultaneous or prior removal of the four adjacent fuel bundles.

Identification of Operator Actions

No operator actions are required to preclude this event, because the protection system design, as previously presented, prevents its occurrence.

15.3.7.3 Core and System Performance

Because the possibility of inadvertent criticality during refueling is precluded, the core and system performances are not analyzed. The withdrawal of the highest worth control rod (or highest worth pair of control rods associated with the same HCU) during refueling does not result in criticality. This is verified experimentally by performing shutdown margin checks (see Section 4.3 for a description of the methods and results of the shutdown margin analysis). Additional reactivity insertion is precluded by refueling interlocks. Because no fuel damage can occur, no radioactive material is released from the fuel. Therefore, this event is not reanalyzed for specific core configurations.

15.3.7.4 Barrier Performance

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

15.3.7.5 Radiological Consequences

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

15.3.8 Control Rod Withdrawal Error During Startup

15.3.8.1 Identification of Causes

It is postulated that during a reactor startup, a gang of control rods or a single control rod is inadvertently withdrawn continuously due to a procedural error by the operator or a malfunction of the automated rod movement control system

The Rod Control and Information System (RC&IS) has a dual channel rod worth minimizer function that prevents withdrawal of any out-of-sequence rods from 100% control rod density to 50% control rod density (i.e., for Group 1 to Group 4 rods). It also has ganged withdrawal sequence restrictions at less than 50% control rod density such that, if the specified withdraw sequence constraints are violated, the rod worth minimizer function of the RC&IS initiates a rod block. These rod worth minimizer rod pattern constraints are in effect from 50% control rod density to the low power setpoint.

The startup range neutron monitor (SRNM) has a period-based trip function that stops continuous rod withdrawal by initiating a rod block if the flux excursion, caused by rod withdrawal, generates a period shorter than 20 seconds. The period-based trip function also initiates a scram if the flux excursion generates a period shorter than 10 seconds. Any single SRNM rod block trip initiates a rod block. Any two divisional scram trips out of four divisions initiates a scram. The SRNM also has upscale rod block and upscale scram functions as a double protection for flux excursion. A detailed description of the period-based trip function is presented in Chapter 7.

For this transient to happen, a large reactivity addition must be introduced. The reactor must be critical, with control rod density greater than 50%. Additionally, rod block logic of both rod worth minimizer channels must fail such that an out-of-sequence (i.e., in violation of Ganged Withdrawal Sequence Restriction (GWSR) rules) gang of rods (or a single rod) can be continuously withdrawn. The causes of the event are summarized in Figure 15.3-7b. The probability for this event to occur is considered low enough to warrant being categorized as an infrequent event. The frequency of this event is evaluated in Subsection 15A.3.12.

15.3.8.2 Sequence of Events and Systems Operation

15.3.8.2.1 Sequence of Events

The sequence of events of a typical continuous control rod withdrawal error during reactor startup is shown in Table 15.3-8.

15.3.8.2.2 Identification of Operator Actions

No operator actions are required to terminate this event, because the SRNM period-based trip functions initiate and terminate this event.

15.3.8.3 Core and System Performance

15.3.8.3.1 Analysis Method and Analysis Assumptions

The analysis uses the reactivity insertion model described in References 15.3-8, 15.3-9, and 15.3-10 then implemented in the PANACEA code. The analysis uses a three-dimensional adiabatic model and assumes that no heat is transferred to the coolant. An initial ESBWR core, Reference 15.3-5, with ganged control rods is considered with the error rods being continuously withdrawn from full-in to full-out, i.e. a continuous reactivity insertion. The code calculates the average power and period change as a function of time. Other assumptions used in the analysis are:

(1) The standard BWR data of the adiabatic model is used

(2) Six delayed neutron groups are assumed

15.3.8.3.2 Analysis Conditions and Results

(1) Analysis Conditions

a. The reactor is assumed to be in the critical condition before the control rod withdrawal, with an initial power of 0.001% rated, and a temperature of 271°C at the fuel cladding surface.

b. The worth of the withdrawn rods (gang) is 3% Δ k from full-in to full-out. Gang rod withdrawal is used during a normal startup to provide a larger reactivity change than a single rod withdrawal case.

c. The control rod withdrawal speed is 28.0 mm/s (1.1 in/s), the nominal ESBWR FMCRD withdrawal speed.

d. With the gang rod withdrawal, the reactor period monitored by any SRNM is relatively the same. Any single channel bypass of the SRNM does not affect the result.

(2) Analysis Result

There were two evaluations performed using the initial ESBWR core with no credit taken for the scram reactivity in the enthalpy calculation. For each case, the peak pin enthalpy is the calculated enthalpy at the time that corresponds to full control rod insertion following the scram. The first case examines a 10 second period scram trip, which is initiated as early as 15 seconds or as late as 28 seconds after the start of the transient, depending on the exposure. The peak pin enthalpy reached for this scram is then conservatively extracted at a time corresponding to a 10 second period with an additional 2.23 second scram time. The result for the 10 second period trip showed that the peak fuel enthalpy was approximately 145 J/g, much lower than the control rod withdrawal error criteria of 712 J/g.

The second case examines a 15% rated power high neutron flux scram, which is initiated as early as 26 seconds or as late as 40 seconds after the start of the transient, depending on the exposure. The core average enthalpy reached for this scram is then conservatively extracted at a time corresponding to 15% rated power high neutron flux scram with an additional 2.23 second scram time. The result for the 15% rated power high neutron flux scram showed that the peak fuel enthalpy was approximately 391 J/g, which is also much lower than the control rod withdrawal error criteria of 712 J/g.

The results are illustrated in Figure 15.3-7a. Table 15.3-8 contains a sequence of events for a continuous rod withdrawal error during reactor startup assuming both a period and high flux scram.

15.3.8.3.3 Evaluation Based On Criteria

Due to the effective protection function of the period-based trip function, the fuel enthalpy increase is small. The fuel enthalpy increase criterion of 712 J/g (170 cal/gm) for a control rod withdrawal error event is satisfied. An additional analysis was performed with the same assumptions and conditions as stated above, but without the SRNM protection function. Under this condition, the APRM startup mode scram trip at 15% power provides the protection function. Flux and power excursion caused by continuous rod withdrawal error reaches the 15% power scram level and the reactor scrams.

15.3.8.4 Barrier Performance

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

15.3.8.4.1 Radiological Consequences

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

15.3.8.4.2 COL Action Item (Deleted)

15.3.9 Control Rod Withdrawal Error During Power Operation

15.3.9.1 Identification of Causes

In ESBWR, the Automated Thermal Limit Monitor (ATLM) subsystem performs the associated rod block monitoring function. The ATLM is a dual channel subsystem of the RC&IS. Each ATLM channel has two independent thermal limit monitoring functions. One function monitors the Minimum Critical Power Ratio (MCPR) limit and protects the operating limit MCPR, another function monitors the Maximum Linear Heat Generation Rate (MLHGR) limit and protects the operating limit of the MLHGR. The rod block algorithm and setpoint of the ATLM are based on actual on-line core thermal limit information. If any operating limit protection setpoint limit is reached, such as due to control rod withdrawal, control rod withdrawal permissive is removed. Detailed description of the ATLM subsystem is presented in Chapter 7.

The causes of a potential control rod withdrawal error are either a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously, or a malfunction of the automated rod withdrawal sequence control logic during automated operation in which a gang of control rods is withdrawn continuously. In either case, the operating thermal limits rod block function blocks any further rod withdrawal when the operating thermal limit is reached. That is, the withdrawal of rods is stopped before the operating thermal limit is reached. Because there is no operating limit violation due to the preventive function of the ATLM, there is no rod withdrawal error transient event.

The frequency of this event is evaluated in Subsection 15A.3.13.

15.3.9.2 Sequence of Events and System Operation

A single control rod or a gang of control rods is withdrawn continuously due to an operator error or a malfunction of the automated rod withdrawal sequence control logic,. The ATLM operating thermal limit protection function of either the MCPR or MLHGR protection algorithms stops further control rod withdrawal when either operating limit is reached. As there are no operating limit violations, there is no basis for occurrence of the continuous control rod withdrawal error event in the power range.

No operator action is required to preclude this event, because the plant design as described above prevents its occurrence.

15.3.9.3 Core and System Performance

The performance of the ATLM subsystem of the RC&IS prevents the control rod withdrawal error event from occurring. The core and system performance are not affected by such an operator error or control logic malfunction. There is no need to analyze this event.

15.3.9.4 Barrier Performance

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

15.3.9.5 Radiological Consequences

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

15.3.10 Fuel Assembly Loading Error, Mislocated Bundle

15.3.10.1 Identification of Causes

The mislocated fuel bundle error involves the mislocation of at least two fuel bundles. The scenario includes: 1) one location loaded with a bundle that operates at a lower power than planned and 2) another location with a bundle operating at a higher power than planned. The frequency of this event is evaluated in Subsection 15A.3.14.

15.3.10.2 Sequence of Events and Systems Operation

There is a strong possibility that the core monitor will recognize the mislocated fuel bundle, thereby allowing the reactor operators to mitigate the consequences of this event. In the best situation where the high radial power mislocated bundle is adjacent to an instrument, the power adjustment in radially Automatic Fixed In-core Probe (AFIP) or local power range monitor (LPRM) adapting monitoring systems will detect the higher bundle power. The reactor will be operated such that the most limiting of the bundles near the mislocation will be maintained below the operating limit MCPR. A less effective situation is where the mislocated bundle has a bundle between it and an instrument.

An ineffective situation occurs when the core monitor does not recognize the mislocation because the monitoring system is not radially AFIP or LPRM adapted. The mislocated bundle sequence of events is discussed in Table 15.3-9.

15.3.10.3 Core and System Performance

Assuming the mislocated bundle is not monitored, one possible state of operation for the fuel bundle is that it operates through the cycle close to or above the fuel thermal mechanical limit.

15.3.10.4 Barrier Performance

The potential exists that if the fuel bundle operates above the thermal mechanical limit, one or more fuel rods may experience cladding failure. If this were to occur, the adverse consequence of operation are detectable and can be suppressed during operation in the same manner as leaking fuel rods resulting from other causes. In this context, the adverse consequence is the perforation of a small number of fuel rods in the mislocated fuel assembly. Any perforations that may result would be localized, there would be only a few perforations, and the perforations would not propagate to other fuel rods or fuel assemblies.

15.3.10.5 Radiological Consequences

The perforation of a small number of fuel rods leads to the release of fission products to the reactor coolant, which is detected by the offgas system. A control rod inserted in the vicinity of the leaking fuel rods suppresses the power in the leaking fuel rods, returns the thermal–hydraulic condition to normal heat transfer with its characteristic low temperature difference between the cladding and the coolant, and reduces the fission product release and offgas.

Further discussion on the analysis methods for the mislocated bundle event is given in Reference 15.3-3. Bounding radiological analysis of this event is contained in Reference 15.3-3. The GESTAR analysis bounds the ESBWR design using the generic atmospheric dispersion factors contained in Table 2.0-1 (Turbine Building release). The generic radiological analysis (Reference 15.3-11) requires licensees to utilize the methodology contained in Regulatory Guide 1.145, Revision 1 (Reference 15.3-12) for off-site dispersion factors and Regulatory Guide 1.194, Revision 0 Reference 15.3-13) for Control Room dispersion factors. Core verification requirements and confirmation of assumptions is discussed in Subsection 15.3.11.3.

Proper location of the fuel assembly in the reactor core is readily verified by visual observation and assured by verification procedures during core loading. GEH provides recommended fuel assembly loading instructions for the initial core as part of the Startup Test Instructions (STIs). It is expected that the plant owners use similar procedures during subsequent refueling operations. Verification procedures include inventory checks, current bundle location logs, serial number verifications and visual or photographic inspection of the loaded core. The verification procedures are designed to minimize the possibility of the occurrence of the mislocated bundle event.

15.3.11 Fuel Assembly Loading Error, Misoriented Bundle

15.3.11.1 Identification of Causes

The misoriented bundle event has been evaluated in Reference 15.3-3, on a generic bounding basis. The misoriented bundle sequence of events is discussed in Table 15.3-10.

Proper orientation of fuel assemblies in the reactor core is readily verified by visual observation and assured by verification procedures during core loading. Five separate visual indications of proper fuel assembly orientation exist:

- The channel fastener assemblies, including the spring and guard used to maintain clearances between channels, are located at one corner of each fuel assembly adjacent to the center of the control rod.
- The identification boss on the fuel assembly handle points toward the adjacent control rod.
- The channel spacing buttons are adjacent to the control rod passage area.

- The assembly identification numbers that are located on the fuel assembly handles are all readable from the direction of the center of the cell.
- There is cell–to–cell replication.

Experience has demonstrated that these design features are clearly visible so that any misoriented fuel assembly would be readily identifiable during core loading verification.

The frequency of this event is evaluated in Subsection 15A.3.15.

15.3.11.2 Core and Barrier Performance

The bounding analysis for the misoriented fuel assembly is discussed in detail in Reference 15.3-1.

15.3.11.3 Radiological Consequences

Bounding radiological analysis of these events is contained in Reference 15.3-3. The GESTAR analysis bounds the ESBWR design using the generic atmospheric dispersion factors contained in Table 2.0-1 (Turbine Building release). The generic radiological analysis (Reference 15.3-11) requires licensees to utilize the methodology contained in Regulatory Guide 1.145, Revision 1 (Reference 15.3-12) for off-site dispersion factors and Regulatory Guide 1.194, Revision 0 (Reference 15.3-13) for Control Room dispersion factors.

The NRC requires licensees to certify that core verification procedures (see Subsection 13.5.2 for COL applicant procedure requirements) have the following characteristics:

- During fuel movement, each move (location, orientation, and seating) is observed and checked at the time of completion by the operator and spotter.
- After completion of the core load, the core is verified by a video recording of the core using an underwater camera.
- Two independent reviewers perform the verification of the bundle serial number location, orientation, and seating. Each independent team records the bundle serial numbers on a core map, which is verified with the design core-loading pattern.

Should a bundle mislocation, misorientation, and seating occur and go undetected, the plant specific acceptance of the generic GESTAR analysis is revoked, and the classification of this event is changed from “infrequent incident” (infrequent event) classification to an “incident of moderate frequency” (AOO) classification immediately for that plant.

Assumptions Requiring Confirmation

Site parameter assumptions in the radiological analysis are confirmed in Subsection 2.0.1.

15.3.12 Inadvertent SDC Function Operation

15.3.12.1 Identification of Causes

A shutdown cooling malfunction leading to a moderate temperature decrease could result from misoperation of the cooling water controls for the RWCU/SDC system heat exchangers. The resulting temperature decrease would cause a slow insertion of positive reactivity into the core. The frequency of this event is evaluated in Subsection 15A.3.7.

15.3.12.2 Sequence of Events and Systems Operation

Sequence of Events

A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for RWCU/SDC system heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. During startup, a flux scram may occur. During full power operation or startup, no thermal limits are reached. The sequence of events for this event is a slow rise in reactor power. The operator can take action to limit the power rise.

System Operation

No unique safety actions are required to avoid unacceptable safety results for transients as a result of a reactor coolant temperature decrease induced by misoperation of the shutdown cooling heat exchangers. In startup or cooldown operation, where the reactor is at or near critical, the slow power increase resulting from the cooler moderator temperature is controlled by the operator in the same manner normally used to control power in the startup range.

15.3.12.3 Core and System Performance

The increased subcooling caused by misoperation of the RWCU/SDC shutdown cooling mode could result in a slow power increase due to the reactivity insertion. During power operation the reactor settles in a new steady state. During startup, if the power rises such that the neutron flux setpoint is reached, the power rise is terminated by a flux scram before approaching fuel thermal limits. Therefore, only a qualitative description is provided and this event is not analyzed for a specific core configuration.

15.3.12.4 Barrier Performance

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

15.3.12.5 Radiological Consequences

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

15.3.13 Inadvertent Opening of a Safety Relief Valve

15.3.13.1 Identification of Causes

Cause of inadvertent safety relief valve (SRV) opening is attributed to malfunction of the valve or an operator-initiated opening of the SRV. It is simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve design is provided in Chapter 5.

During normal operation, a spurious signal causes one SRV to open. The steam of this SRV is discharged in the suppression pool, if the subsequent manual closure of the SRV is not obtained, then the suppression pool temperature increases, reaching the scram set-point and finally scramming the reactor. The frequency of this event is evaluated in Subsection 15A.3.8.

15.3.13.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.3-11 lists the sequence of events for this event.

Identification of Operator Actions

The plant operator must re-close the valve as soon as possible and check that reactor and TG output return to normal. If the valve cannot be closed, plant shutdown should be initiated.

Systems Operation

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

15.3.13.3 Core and System Performance

The opening of one 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 SB&PC system senses the nuclear system pressure decrease and within a few seconds closes the TCVs enough to stabilize the reactor vessel pressure at a slightly lower value and the reactor settles at nearly the initial power level. Eventually, the plant automatically scrams on high suppression pool temperature.

Thermal margins decrease only slightly through the transient and no fuel damage results from the event.

15.3.13.4 Barrier Performance

The transient resulting from the inadvertent SRV opening is a mild depressurization, which is within the range of normal load following and has no significant effect on RCPB and containment design pressure limits.

15.3.13.5 Radiological Consequences

While the effect of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool. Because this activity is contained in the primary containment, there is no exposure to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity confined within containment, use the fuel and auxiliary pools cooling system (FAPCS) to remove radioactivity from the pool, or discharge it to the environment under controlled release conditions. If purging containment is chosen, the release is performed in accordance with Technical Specifications; therefore, this event, results in a small increase in the yearly integrated exposure level.

15.3.14 Inadvertent Opening of a Depressurization Valve

15.3.14.1 Identification of Causes

Potential causes of inadvertent Depressurization Valve (DPV) opening are malfunction of the valve or an operator-initiated DPV opening. It is simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve design is provided in Subsection 6.3.2.8 and 7.3.1. The frequency of this event is evaluated in Subsection 15A.3.9. The discussion of inadvertent opening of a DPV also applies to a stuck open safety valve which is piped to the drywell. The frequency of a stuck open safety valve is evaluated in Subsection 15A.3.10.

15.3.14.2 Systems Operation and Sequence of Events

15.3.14.2.1 Sequence of Events

If auxiliary power remains available, the sequence of events is similar to the stuck open relief valve sequence given in Table 15.3-12, except that scram occurs on high drywell (DW) pressure within a few seconds. If auxiliary power is not available, the sequence of events is similar to the Main Steam Line Break sequence given in Table 6.3-8.

Identification of Operator Actions

The operator should monitor the reactor water level and control makeup systems. Because the valve cannot be closed remotely, plant shutdown should be initiated. The operator should confirm the plant scrams on drywell pressure, or scram before the high drywell pressure is reached if a DPV is confirmed to be open, and monitor RPV and DW pressure and control with the bypass valves, if necessary.

Systems Operation

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

15.3.14.3 Core and System Performance

The opening of one DPV allows steam to be discharged into the drywell. The sudden increase in the rate of steam flow leaving the reactor vessel causes a depressurization transient.

The SB&PC system senses the nuclear system pressure decrease and within a few seconds closes the TCVs sufficiently to stabilize the reactor vessel pressure at a slightly lower value with the reactor returning to nearly the initial power level. The plant automatically scrams on high drywell pressure. After scram, depressurization of the RPV will resume.

Thermal margins decrease only slightly through the transient and no fuel damage results from the event.

15.3.14.4 Barrier Performance

The transient resulting from the inadvertent DPV open is a depressurization which is bounded by the spectrum of loss of coolant accidents analyzed in Chapter 6. It does not approach the RCPB and containment design pressure limits.

15.3.14.5 Radiological Consequences

While the effect of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the drywell. After the drywell pressurizes, and the DW to WW vents clear, the steam will be vented to the suppression pool and condenses in the pool. Because the activity is contained in the primary containment, there is no exposure to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity confined inside the containment, use FAPCS to remove radioactivity from the pool, or discharge it to the environment under controlled release conditions. If purging containment is chosen, the release is performed in accordance Technical Specifications; therefore, this event, results in a small increase in the yearly integrated exposure level.

15.3.15 Stuck Open Safety Relief Valve

15.3.15.1 Identification of Causes

Cause of a stuck open safety relief valve is attributed to the malfunction of the valve after it has opened either inadvertently or in response to a high pressure signal. It is simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve design is provided in Chapter 5.

In this analysis, after any event that produces a reactor scram, it is assumed that a SRV remains open without any possibility of closure. The operations of the ICs produce a depressurization, with the HP_CRD operating to recover the level after the scram. The event is analyzed with 4 ICs available and with bounding capacity, to observe the maximum possible depressurization rate. Finally the reactor reaches near atmospheric pressure. The frequency of this event is evaluated in Subsection 15A.3.10.

15.3.15.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.3-12 lists the sequence of events for this event. If auxiliary power is not available, the sequence of events is similar to the Main Steam Line Break sequence given in Table 6.3-8.

Identification of Operator Actions

The plant operator must re-close the valve as soon as possible and check that the reactor and TG output return to normal. If the valve cannot be closed and the reactor has scrammed because of some other reason (if the SRVs are in the open condition, the reactor has scrammed previously, except for the Inadvertent SRV opening analyzed previously), manual activation of the IC and other systems can be initialized to reduce the amount of steam reaching the suppression pool.

Systems Operation

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

15.3.15.3 Core and System Performance

The opening of one SRV allows steam to be discharged to the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a depressurization transient, with the vessel pressure slowly decreasing until reaching atmospheric pressure. The SRV steam discharge also results in slight heating of the suppression pool.

Thermal margins decrease slightly through the transient and no fuel damage is predicted for this event.

15.3.15.4 Barrier Performance

As presented previously, the transient resulting from a stuck open relief valve is the total depressurization of the pressure vessel, which is within the range of normal plant operation and therefore has no significant effect on RCPB and containment design pressure limits.

15.3.15.5 Radiological Consequences

While the effect of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool. Because this activity is contained in the primary containment, there is no exposure to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity confined inside containment, use FAPCS to remove radioactivity from the pool, or discharge it to the environment in a controlled manner. If purging of the containment is chosen, the release is performed in accordance with Technical Specifications. Consequently, this event results in a small increase in the yearly integrated exposure level.

15.3.16 Liquid-Containing Tank Failure

15.3.16.1 Identification of Causes

An unspecified event causes the complete release of the radioactive inventory in all tanks containing radionuclides in the liquid radwaste system. Postulated events that could cause a release of the inventory of a tank are sudden unmonitored cracks in the vessel or operator error. Small cracks and consequent low level releases are bounded by this analysis.

The ESBWR Radwaste Building is designed to seismic requirements as specified in Subsection 3.8.4. Because of these design capabilities, it is considered remote that any major event involving the release of liquid radwaste into these volumes would result in the release of these liquids to the environment via the liquid pathway. Releases as a result of major cracks would instead result in the release of the liquid radwaste to the compartment and then to the building sump system for containment in other tanks or emergency tanks. A complete description of the liquid radwaste system is found in Section 11.2, except for the tank inventories, which are found in Section 12.2.

A liquid radwaste release caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instructions. A positive action interlock system is also provided to prevent inadvertent opening of a drain valve. Should a release of liquid wastes occur, the sealed concrete walls and the steel tank cubicle liners contain the liquid waste thereby preventing liquid release into the environment. The liquid waste would then be transferred from the tank cubicle to the radwaste sumps for processing.

The probability of a complete tank release is considered low enough to warrant this event as an Infrequent Event. The frequency of this event is evaluated in Subsection 15A.3.16.

15.3.16.2 Sequence of Events and Systems Operations

Following a failure, the area radiation alarms would be expected to alarm at one minute with operator intervention following at approximately five minutes after release. However, the rupture of a waste tank would be contained and allow the operator time to develop and setup a means to process the contained waste. Gases would be processed through the Radwaste Building HVAC system as described in Subsections 9.4.3, 11.5.3.2.8 and 12.3.3.2.4.

Liquid releases would be contained within the sealed concrete walls and steel liquid waste management system tank cubicle liners, and would present no immediate threat to the environment leaving the operator sufficient time (on the order of hours) in which to recover systems to pump the release into holding tanks or emergency tanks.

15.3.16.3 Results

A single pathway is considered for release of fission products to the environment via airborne releases. The liquid pathway is not considered because of the mitigation capabilities of the Radwaste Building, following the guidance of SRP Section 15.7.3.III.1.b. General Design Criterion (GDC) 60 is met, as the release of radioactive materials in this case is suitably controlled.

For the airborne pathway, volatile iodine species in the tank using the cumulative inventories in Tables 12.2-13a through 12.2-13g are considered. Although isolation is expected within minutes of the occurrence, release of 100% of the iodine inventory is conservatively assumed instantaneously with no holdup or plateout. Specific parameters for this analysis are found in Tables 15.3-17 and 15.3-18.

For the Radwaste Building tanks analyzed, no liquid or significant (from airborne species) ground contamination is expected. Airborne doses are given in Table 15.3-19 and are a fraction of the 2.5 rem TEDE offsite and 5 rem onsite criteria. The effluent concentration limits of 10 CFR 20 Appendix B are met, as no liquid effluent is released to the environment as a result of the tank failure.

15.3.17 COL Information

15.3-1-A Potentially Limiting Infrequent Events for Initial Core (Deleted)

15.3-2-H Potentially Limiting Infrequent Events for Reload Core Design Changes (Deleted)

15.3-3-A Control Rod Withdrawal Error During Startup (Deleted)

15.3-4-A EAB X/Q Value (Deleted)

15.3-5-A LPZ X/Q Values (Deleted)

15.3-6-A Control Room X/Q Values (Deleted)

15.3.18 References

15.3-1 GE Nuclear Energy, "General Electric Standard Application for Reactor Fuel--United States Supplement," NEDE-24011-P-A-US, (Latest approved revision).

15.3-2 Deleted

15.3-3 FLN–2004–026, "GESTAR I Amendment 28 Revision 1, Misloaded Fuel Bundle Event Licensing Basis Change to Comply with Standard Review Plan 15.4.7," Margaret E. Harding to Mel B. Fields August 23, 2004

15.3-4 Global Nuclear Fuel, "GE14 for ESBWR Nuclear Design Report", NEDC-33239-P, Class III (Proprietary), Revision 2, April 2007, NEDO-33239, Class I (Non-proprietary), Revision 2, April 2007.

15.3-5 Global Nuclear Fuel, "ESBWR Initial Core Nuclear Design Report", NEDC-33326-P, Class III (Proprietary), Revision 0, July 2007, NEDO-33326, Class I (Non-proprietary), Revision 0, July 2007.

15.3-6 GE-Hitachi Nuclear Energy, "ESBWR Initial Core Transient Analyses", NEDO-33337 Class I, Revision 0, Scheduled September 2007.

15.3-7 GE-Hitachi Nuclear Energy, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis", NEDO-33338 Class I, Revision 0, Scheduled September 2007.

15.3-8 General Electric Co., "Steady State Nuclear Methods," NEDE-30130-P-A, April 1985.

15.3-9 R. J. Reda (GE) to R. C. Jones, Jr. (NRC), "Implementation of Improved Steady-State Nuclear Methods," MFN-098-96, July 2, 1996.

15.3-10 G. A. Watford (GE) to E. D. Kendrick (NRC), "Implementation of Improved Steady-State Nuclear Methods," MFN-003-98, January 8, 1998.

15.3-11 NRC "Final Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report GESTAR II Amendment 28, 'Misloaded Fuel Bundle Event Licensing Basis Change to Comply with Standard Review Plan 15.4.7,' Global Nuclear Fuel Project 712."

15.3-12 Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1.

15.3-13 Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Revision 0.

Table 15.3-1

Results Summary of Infrequent Events [1]

Sub-section I.D.	Description	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPaG (psig)	Max. Vessel Bottom Pressure, MPaG (psig)	Max. Steamline Pressure, MPaG (psig)	Max. Core Average Surface Heat Flux, % of Initial	ΔCPR/ ICPR
15.3.1	Loss of Feedwater Heating with SCRRI failure	122	7.13 (1034)	7.27 (1054)	7.09 (1028)	121	0.11
15.3.2	FWCF – Maximum Demand	117	7.29 (1057)	7.43 (1078)	7.25 (1052)	109	0.04
15.3.3	Pressure Regulator Failure – Opening of all TCVs and BPVs	100	7.08 (1027)	7.21 (1046)	7.04 (1021)	100	0.00
15.3.4	Pressure Regulator Failure – Closing of all TCVs and BPVs	137	8.06 (1169)	8.19 (1188)	8.06 (1169)	104	0.05
15.3.5	Load Rejection with total bypass failure	339	8.14 (1181)	8.27 (1199)	8.15 (1182)	108	0.11
15.3.6	Turbine Trip with total bypass failure	295	8.13 (1179)	8.26 (1198)	8.13 (1179)	108	0.11
15.3.13	Inadvertent SRV open	101	7.08 (1027)	7.21 (1046)	6.99 (1014)	101	<0.01
15.3.15	Stuck open SRV [2]	100.0	7.08 (1027)	7.21 (1046)	7.04 (1021)	100.0	<0.1

[1] The input parameters and initial conditions used to perform the analysis in this table are located in Table 15.2-1.

[2] The initiating event can produce some over power, but the Stuck SRV open should not produce any appreciable overpower or MCPR reduction.

Table 15.3-2

Sequence of Events for Loss of Feedwater Heating With Failure of Selected Control Rod Run-In

Time (s)	Event*
0	Initiate a 55.6°C (100°F) temperature reduction in the FW system.
25 (est.)	Initial effect of unheated FW starts to raise core power level.
80	High thermal simulated Scram is reached but it is not credited.
300 (est.)	New Steady State Reached.

* See Figure 15.3-1

Table 15.3-3

Sequence of Events for Feedwater Controller Failure – Maximum Demand

Time (sec)	Event *
0	Initiate simulated runout of all FW pumps (170% at rated vessel pressure).
12.4	Main turbine bypass valves opened to control vessel pressure.
15.4	L8 vessel level setpoint is reached.
16.3	Scram, trip of main turbine and FW pump runback is activated.
16.43	Turbine Bypass fast opening activation limits the pressurization of the vessel.
16.5	The rods begin to enter inside the core.
Later	L2 is reached because no FW availability, activating IC and HP_CRD to recover the level and isolating MSIV's.

* See Figure 15.3-2

Table 15.3-4

Sequence of Events for Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

Time (sec)	**Event***
0	Simulate all turbine control valves and bypass valves to open.
19.3	Low turbine inlet pressure trip initiates main steamline isolation.
20.5	MSIV position switch at 85% initiates scram and activates the IC.
24.1	Main steam isolation valves closed. Bypass valves remain open, exhausting steam in steamlines downstream of isolation valves.
31.6	L2 setpoint is reached.
36.5	The IC begins to remove heat from the vessel.
41.8	HP_CRD is activated, this recovers the level.

* See Figure 15.3-3

Table 15.3-5

Sequence of Events for Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

Time (sec)	**Event***
0	Simulate zero steam flow demand to main turbine and bypass valves.
0	Turbine control valves start to close.
1.78	Neutron flux reaches high flux scram setpoint and initiates a reactor scram.
2.03	The rods begin to enter inside the core
2.5	TCV is closed
Long term	HP_CRD is activated on L2 to recover the level

* See Figure 15.3-4

Table 15.3-6a

Sequence of Events for Generator Load Rejection With Total Turbine Bypass Failure

Time (sec)	**Event***
(-)0.015	Turbine-generator detection of loss of electrical load.
0.0	Turbine-generator load rejection sensing devices trip to initiate turbine control valves fast closure.
0.0	Turbine bypass valves fail to operate.
0.08	Turbine control valves closed.
0.20	After detection of not enough bypass availability the RPS initiates a reactor scram.
0.45	The rods begin to enter inside the core
Long term	HP_CRD is activated on L2 to recover the level

Table 15.3-6b

Causes of Control Rod Withdrawal Error (Deleted)

Table 15.3-6c

Sequence of Events for Continuous Control Rod Withdrawal Error During Reactor Startup (Deleted)

Table 15.3-7

Sequence of Events for Turbine Trip With Total Turbine Bypass Failure

Time (sec)	**Event***
0.0	Turbine trip initiates closure of main stop valves.
0.0	Turbine bypass valves fail to operate.
0.10	Turbine stop valves close.
0.20	After detection of not enough bypass availability the RPS initiates a reactor scram.
0.45	The rods begin to enter inside the core.
Long term	HP_CRD is activated on L2 to recover the level.

* See Figure 15.3-6. This figure shows a short term response, for longer term response of a similar transient see Figure 15.3-5

Table 15.3-8

Sequence of Events for Continuous Control Rod Withdrawal Error During Reactor Startup

Time (sec)	Events
0	Operator withdraws a gang of rods (or a single rod) continuously; or a gang of rods (or a single rod) is withdrawn continuously due to a malfunction of the Automated Rod Movement Control System
~15-28	The SRNM Period-Based Scram Trip initiates reactor scram due to short period (less than the 10-second setpoint)
~26-40	The SRNM Flux-Based Scram Trip initiates reactor scram due to high flux (greater than 15% rated power setpoint)
~15-31	Reactor is assumed to be scrammed (all rods inserted) due to short period
~26-43	Reactor is assumed to be scrammed (all rods inserted) due to high flux

Table 15.3-9

Sequence of Events for the Mislocated Bundle

(1)	During the core loading operation, a bundle is loaded into the wrong core location.
(2)	Subsequently, the bundle designated for this location is incorrectly loaded into the location of the previous bundle.
(3)	During the core verification procedure, the two errors are not observed.
(4)	The plant is brought to full power operation without detecting misplaced bundles.
(5)	The plant continues to operate throughout the cycle.

Table 15.3-10

Sequence of Events for the Misoriented Bundle

(1)	During the core loading operation, a bundle is rotated and loaded with incorrect orientation.
(2)	During the core verification procedure, the orientation error is not observed.
(3)	The plant is brought to full power operation without detecting the misoriented bundle.
(4)	The plant continues to operate throughout the cycle.

Table 15.3-11

Sequence of Events for Inadvertent SRV Opening

Time (s)	Event*
0	Spurious opening of one SRV
1.0	Relief valve flow reaches full flow.
30.0	System establishes new steady-state operation.
412.5	Suppression pool temperature reaches the setpoint; suppression pool cooling function is initiated. (Not Credited)
412.5	Suppression pool temperature reaches setpoint; reactor scram is automatically initiated. (scram is conservatively assumed at pool cooling initiation temperature)

* See Figure 15.3-8

Table 15.3-12

Sequence of Events for Stuck Open Safety Relief Valve

Time (s)	**Event***
0	Event happens, the reactor is scrammed and one SRV stuck open
10.0	Relief valve flow reaches one SRV flow.
10.0	The vessel begins depressurization
29.0	HP_CRD is activated on L2
103	Low steamline pressure is activated
103.8	MSIV at 85%
106.0	MSIV is closed
134.8	ICs discharge valves fully open[(1)]
320.0	HP_CRD is deactivated because of L8
Long term	Suppression pool temperature reaches the setpoint; suppression pool cooling function is initiated. (Not Credited) Atmospheric pressure is reached

* See Figure 15.3-9

[(1)] Four ICS at their maximum capacity are actuating during the depressurization.

Table 15.3-13

1000 Fuel Rod Failure Parameters

I. Data and assumptions used to estimate source terms		
A.	Power level, MWt	4590
B.	Number of bundles in core	1132
C.	Core fission product inventory released to coolant	Table 15.3-14
D.	Equivalent full length fuel rods per bundle	87.33
E.	Fuel rods damaged	1000
F.	Radial peaking factor for failed rods	1.5
II. Data and assumptions used to estimate activity released		
A.	Iodine released from failed fuel rods	10%
	Noble gases released from failed fuel rods	10%
	Alkali metals released from failed fuel rods	12%
B.	Iodine released from reactor coolant	10%
	Noble gases released from reactor coolant	100%
	Alkali metals released from reactor coolant	1%
C.	Iodine released from condenser	10%
	Noble gases released from condenser	100%
	Alkali metals released from condenser	1%
	Fission product inventory released to environment	Table 15.3-15
III. Dispersion and Dose Data		
A.	Meteorology:	
	EAB	2.00E-03 s/m^3
	LPZ	
	0 – 8 hours	1.90E-04 s/m^3
	8 – 24 hours	1.40E-04 s/m^3
	1 – 4 days	7.50E-05 s/m^3
	4 – 30 days	3.00E-05 s/m^3

Table 15.3-13

1000 Fuel Rod Failure Parameters

B. Control Room	
0 – 2 hours	1.20E-03 s/m^3
2 – 8 hours	9.8E-04 s/m^3
8 – 24 hours	3.90E-04 s/m^3
1 – 4 days	3.80E-04 s/m^3
4 – 30 days	3.20E-04 s/m^3
C. Dose evaluations	Table 15.4-19

Table 15.3-14

1000 Fuel Rod Failure Fission Product Activity Released to Coolant

Isotope	Activity Released to Primary Coolant (MBq)
Kr-85	8.47E+07
Kr-85m	1.88E+09
Kr-87	3.63E+09
Kr-88	5.11E+09
Rb-86	1.94E+07
I-131	6.82E+09
I-132	9.92E+09
I-133	1.40E+10
I-134	1.55E+10
I-135	1.32E+10
Kr-85	8.47E+07

Table 15.3-15
1000 Fuel Rod Failure Fission Product Activity
Cumulative Release to Environment

	Cumulative Release to Environment (MBq)		
Isotope	**2 Hours**	**8 Hours**	**24 Hours**
Kr-85	7.07E+04	2.82E+05	8.44E+05
Kr-85m	1.34E+06	3.56E+06	4.77E+06
Kr-87	1.79E+06	2.65E+06	2.69E+06
Rb-86	1.61E+00	6.42E+00	1.89E+01
I-131	5.67E+04	2.24E+05	6.49E+05
I-132	6.12E+04	1.23E+05	1.34E+05
I-133	1.13E+05	4.09E+05	9.42E+05
I-134	6.24E+04	7.84E+04	7.86E+04
I-135	9.85E+04	2.95E+05	4.63E+05
Xe-133	1.16E+07	4.54E+07	1.30E+08
Xe-135	3.57E+06	1.15E+07	2.05E+07
Cs-134	1.36E+02	5.44E+02	1.63E+03
Cs-136	4.73E+01	1.88E+02	5.50E+02
Cs-137	8.84E+01	3.53E+02	1.06E+03

Table 15.3-16

1000 Fuel Rod Failure Dose Results

Exposure Location and Time Period/Duration	**Maximum Calculated TEDE** (rem)	**Acceptance Criterion TEDE** (rem)
Exclusion Area Boundary (EAB) for the Entire Period of the Radioactive Cloud Passage	1.56E-01	2.5
Outer Boundary of Low Population Zone (LPZ) for the Entire Period of the Radioactive Cloud Passage	5.94E-02	2.5
Control Room Operator Dose for the Entire Period of the Accident		5.0

Table 15.3-17

Radwaste System Failure Accident Parameters

I	**Data and Assumptions Used to Estimate Source Terms**	
A.	Source inventory	Tables 12.2-13a through 12.2-13g (combined)
B.	Fraction of iodine released	100%
C.	Duration of accident	Instantaneous
II.	**Control Room Parameters**	
A.	Control Room Volume, m^3 (ft^3)	2.2E+03 (7.8E+04)
B.	Unfiltered intake, liters/sec (gpm)	200 (3170)
C.	Occupancy Factors	RG 1.183
III	**Dispersion and Dose Data**	
A.	Atmospheric Dispersion Factors	
	Offsite, s/m^3 (s/ft^3)	2.0E-03 (5.7E-05)
	Control Room	Table 2.0-1
B.	Dose conversion assumptions	RG 1.183
C.	Activity released	Table 15.3-18
D.	Dose consequences	Table 15.3-19

Table 15.3-18

Radwaste System Failure Accident Isotopic Airborne Release to Environment (megabecquerel)

Isotope	Activity (MBq)
I-131	9.7E+04
I-132	9.4E+03
I-133	7.8E+04
I-134	6.2E+03
I-135	3.1E+04
Total I	2.2E+05

Table 15.3-19

Radwaste System Failure Accident Dose Results

Exposure Location	**Maximum Calculated TEDE** (rem)	**Acceptance Criterion TEDE** (rem)
Exclusion Area Boundary (EAB)	7.2E-02	2.5
Low Population Zone (LPZ)	7.2E-02	2.5
Control Room	5.1E-02	5.0

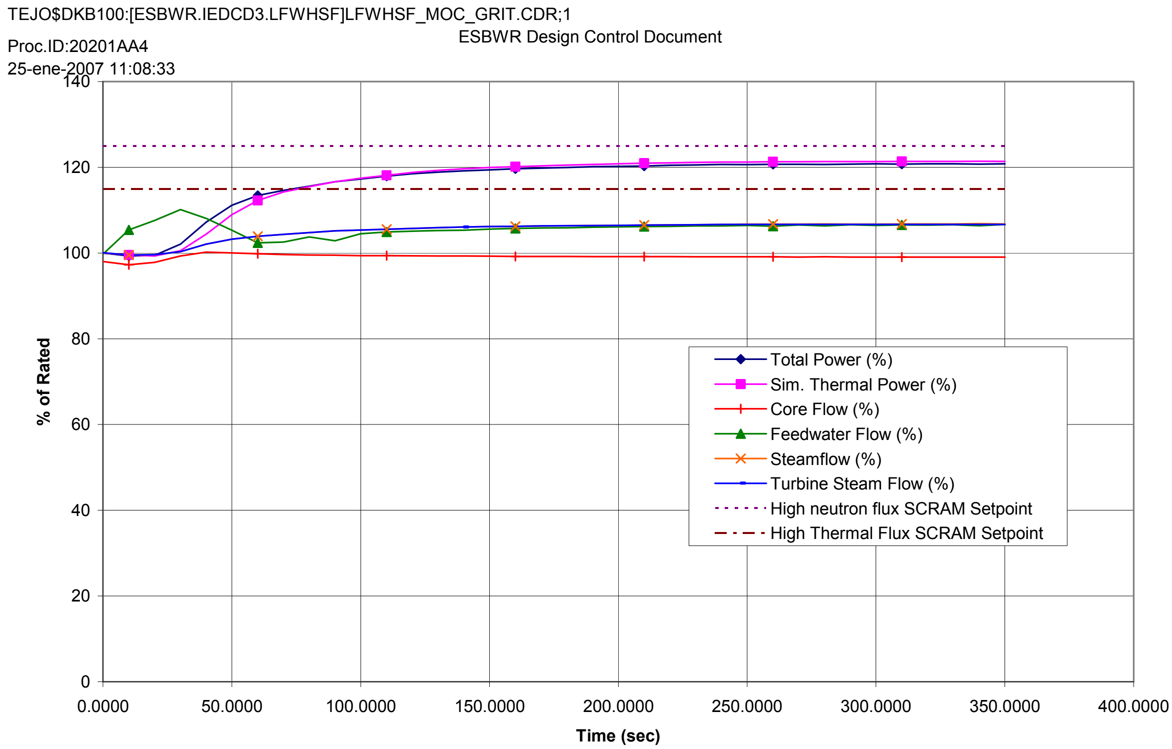


Figure 15.3-1a. Loss of Feedwater Heating with SCRRI/SRI Failure

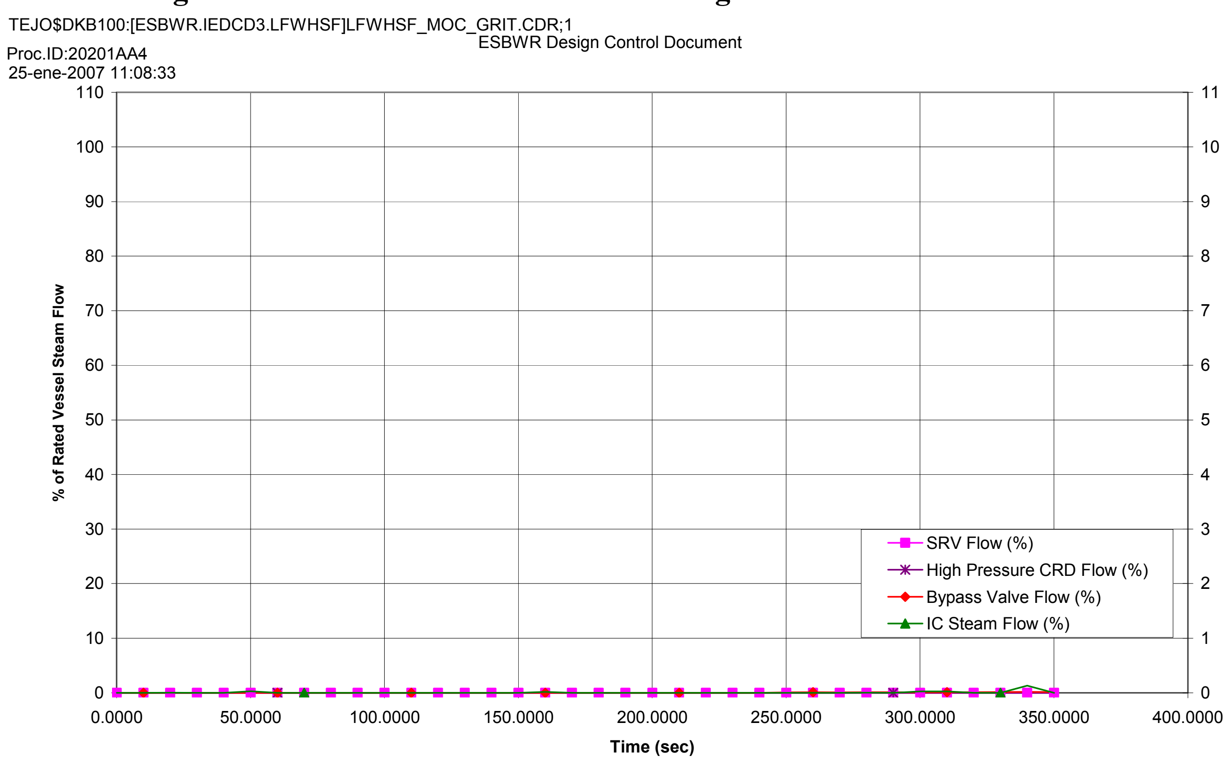


Figure 15.3-1b. Loss of Feedwater Heating with SCRRI/SRI Failure

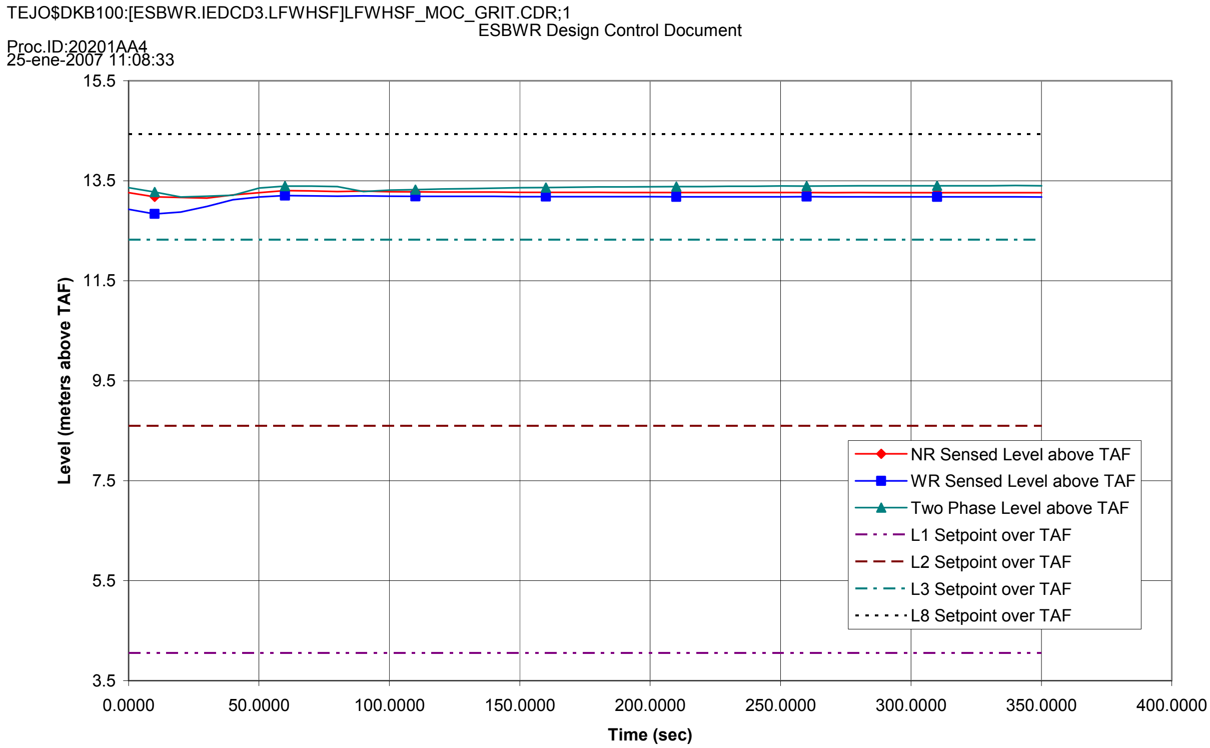


Figure 15.3-1c. Loss of Feedwater Heating with SCRRI/SRI Failure

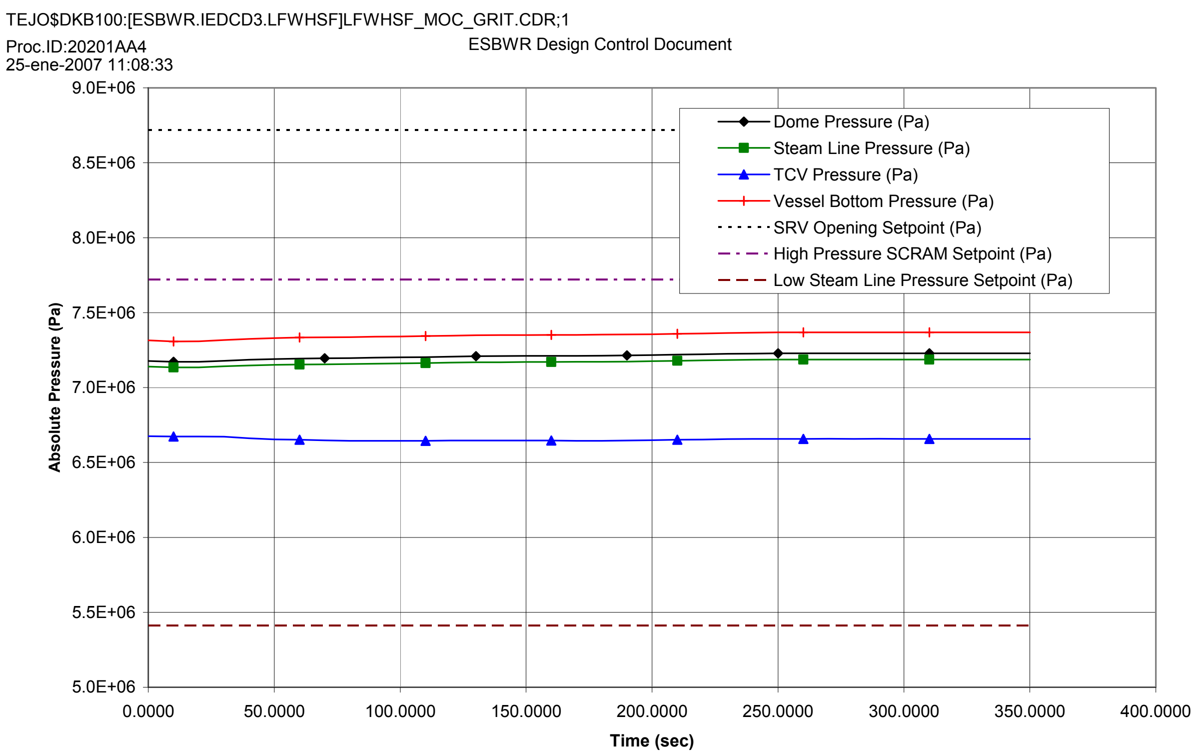


Figure 15.3-1d. Loss of Feedwater Heating with SCRRI/SRI Failure

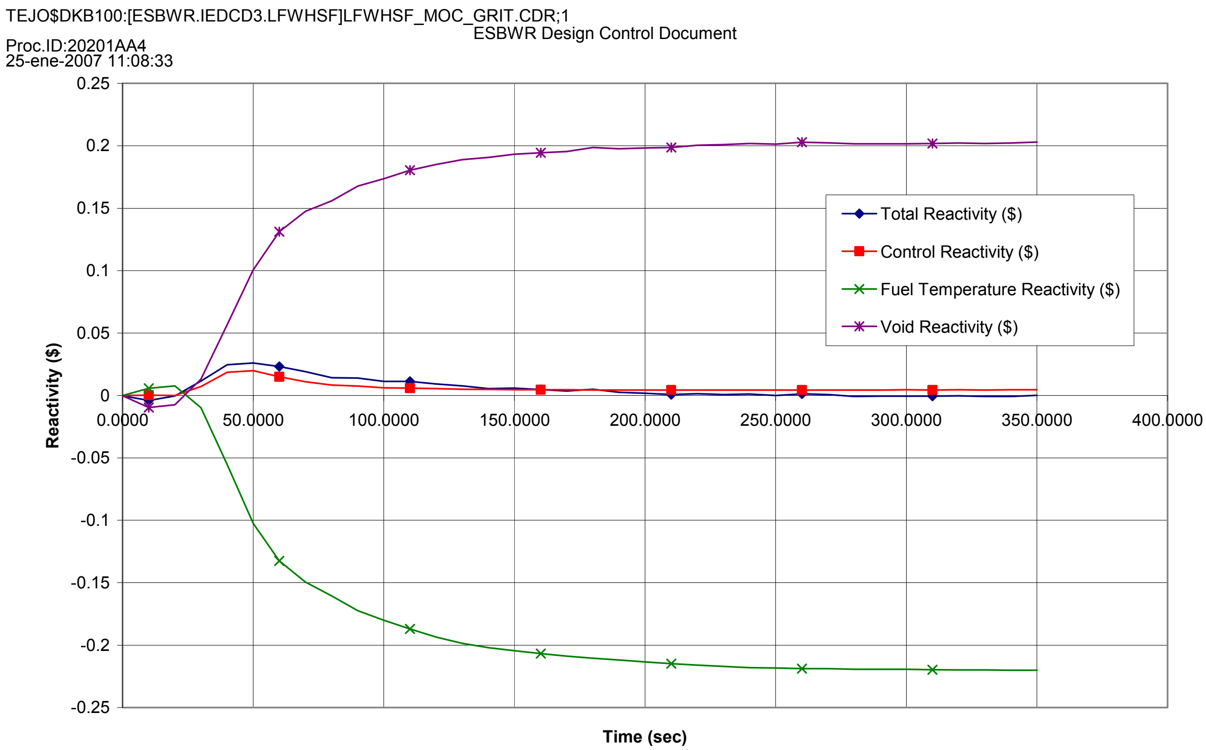


Figure 15.3-1e. Loss of Feedwater Heating with SCRRI/SRI Failure

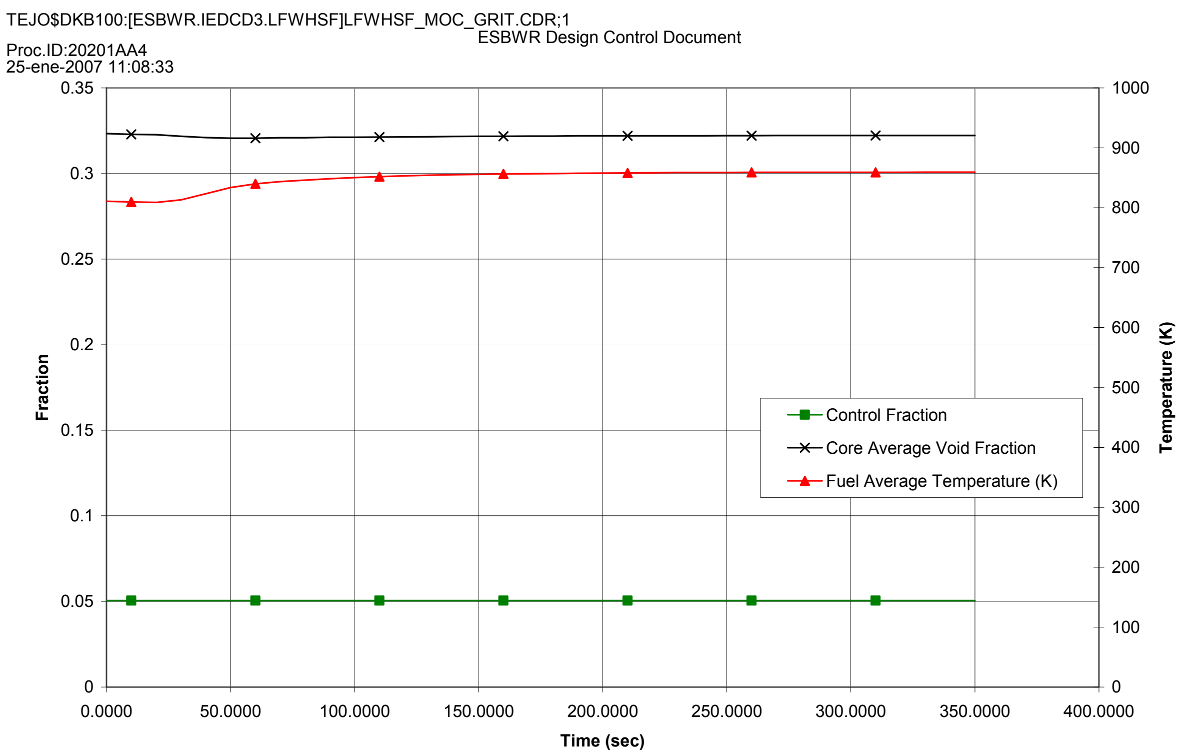


Figure 15.3-1f. Loss of Feedwater Heating with SCRRI/SRI Failure

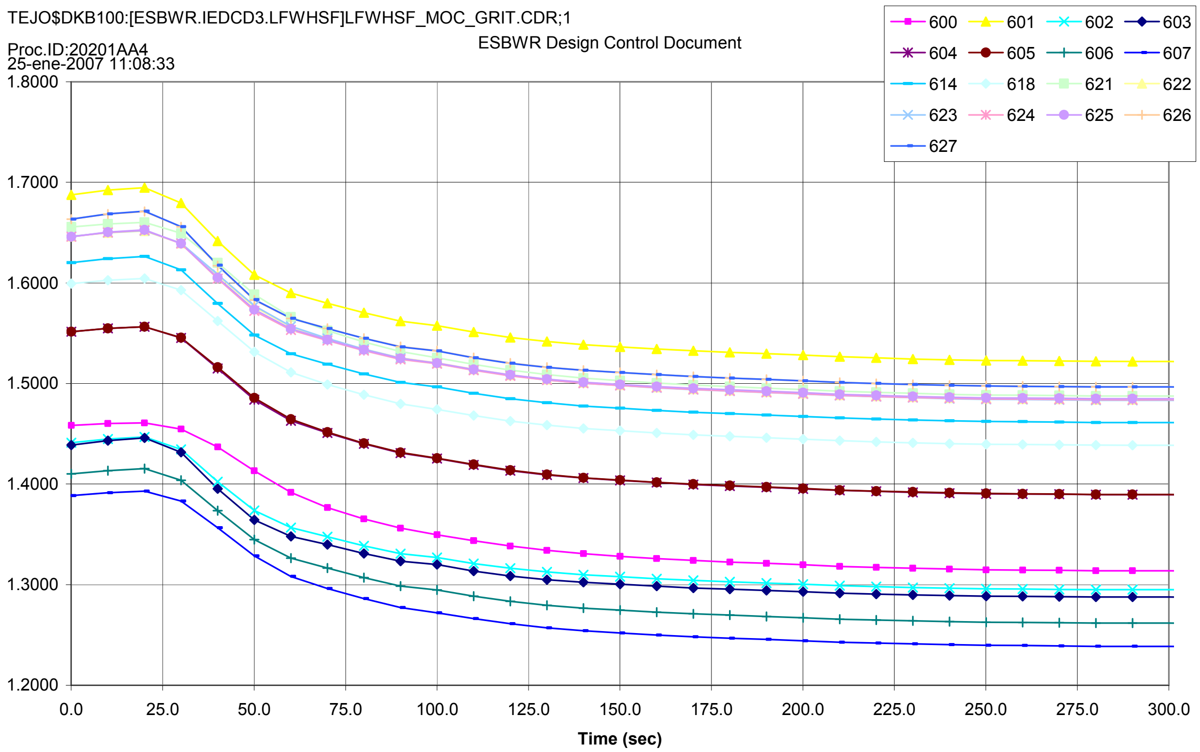


Figure 15.3-1g. Loss of Feedwater Heating with SCRRI/SRI Failure

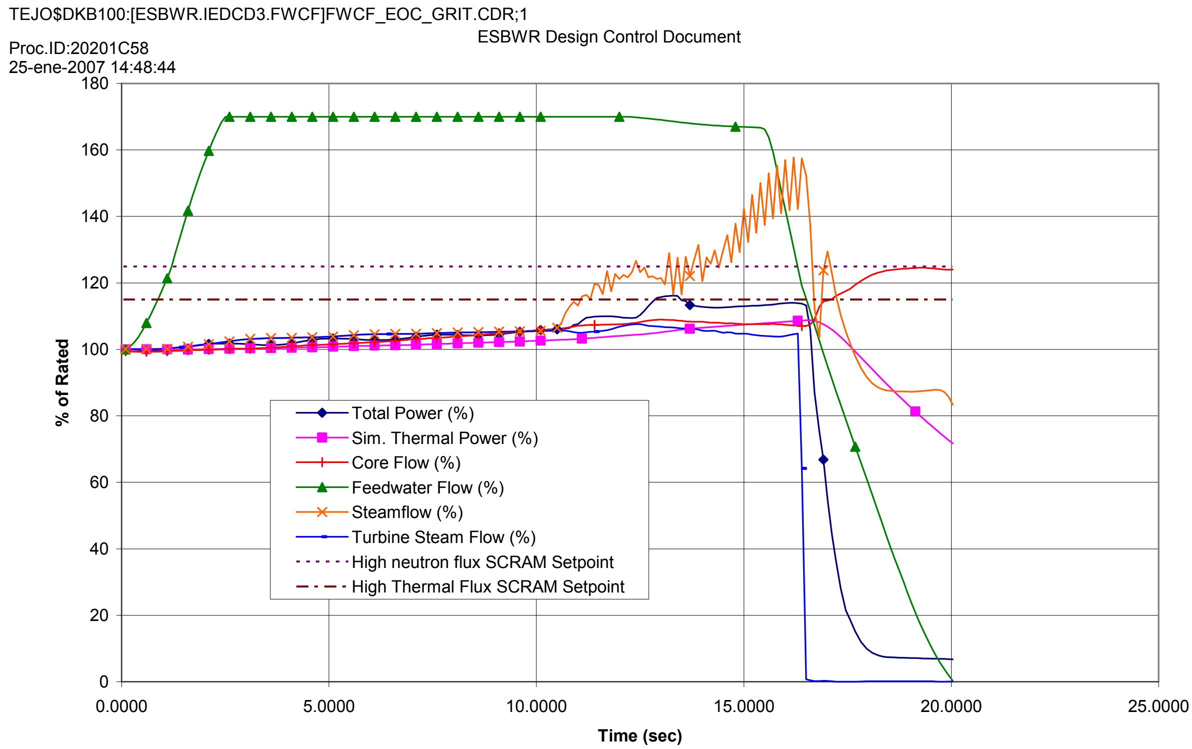


Figure 15.3-2a. Feedwater Controller Failure – Maximum Demand

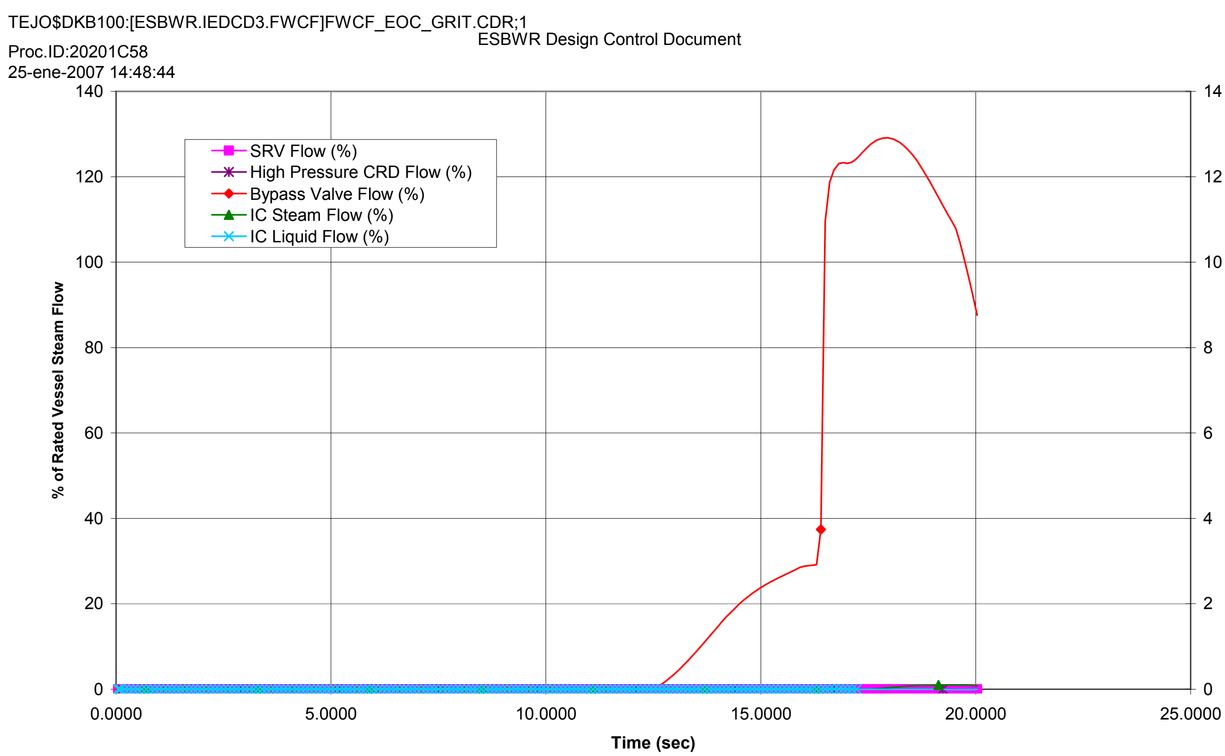


Figure 15.3-2b. Feedwater Controller Failure – Maximum Demand

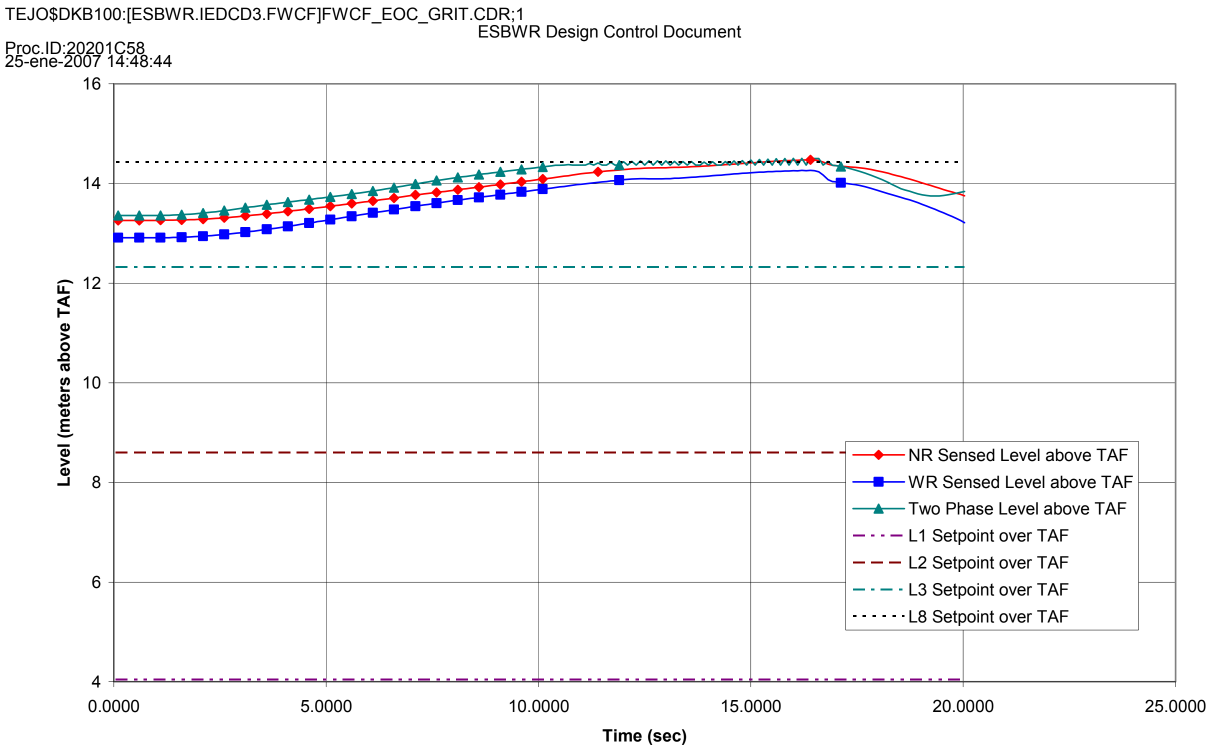


Figure 15.3-2c. Feedwater Controller Failure – Maximum Demand

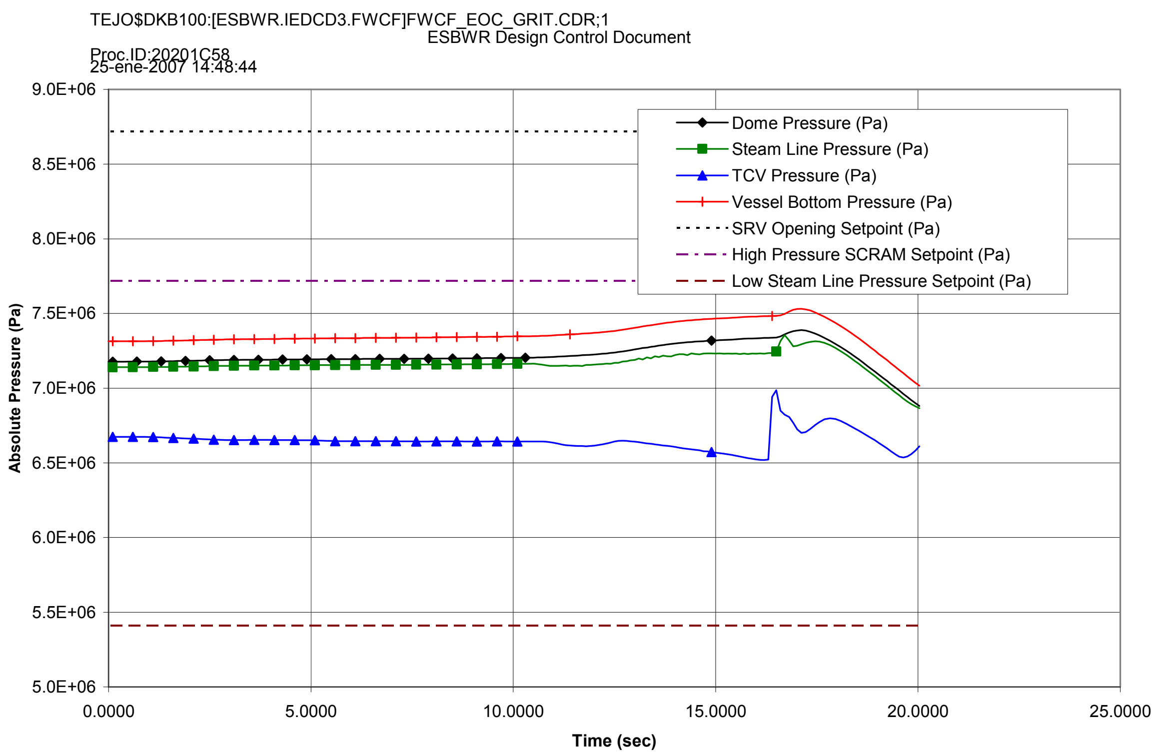


Figure 15.3-2d. Feedwater Controller Failure – Maximum Demand

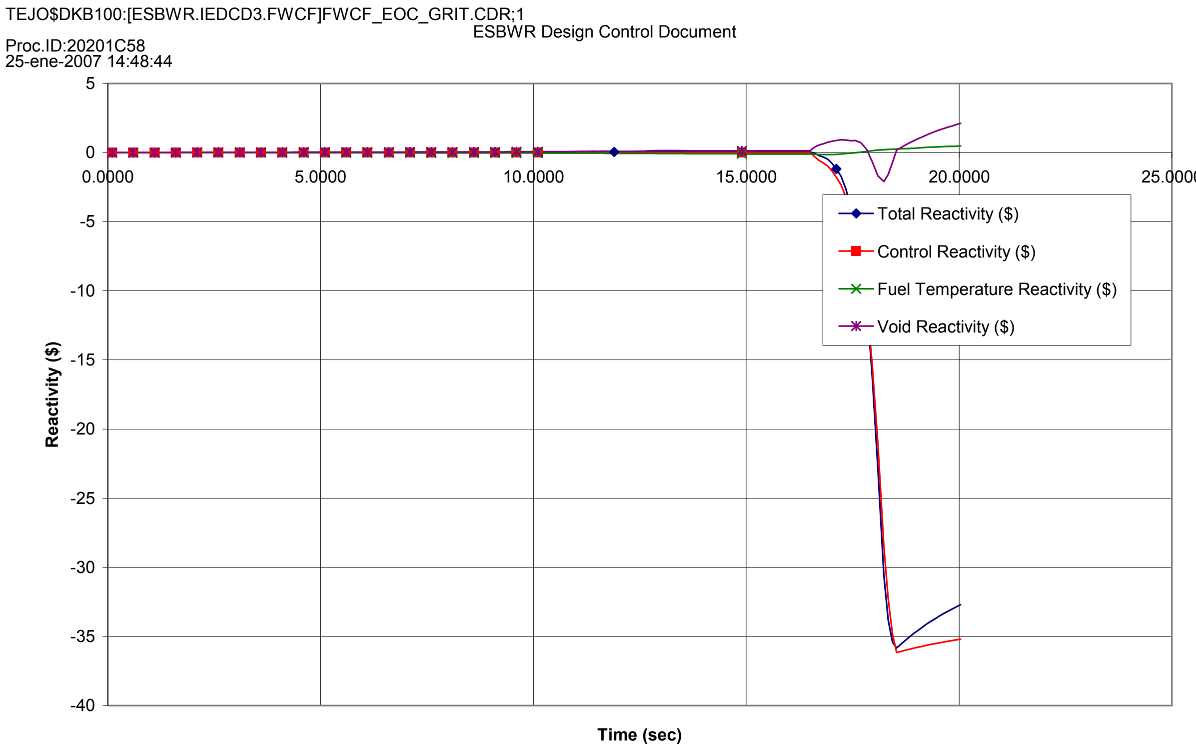


Figure 15.3-2e. Feedwater Controller Failure – Maximum Demand

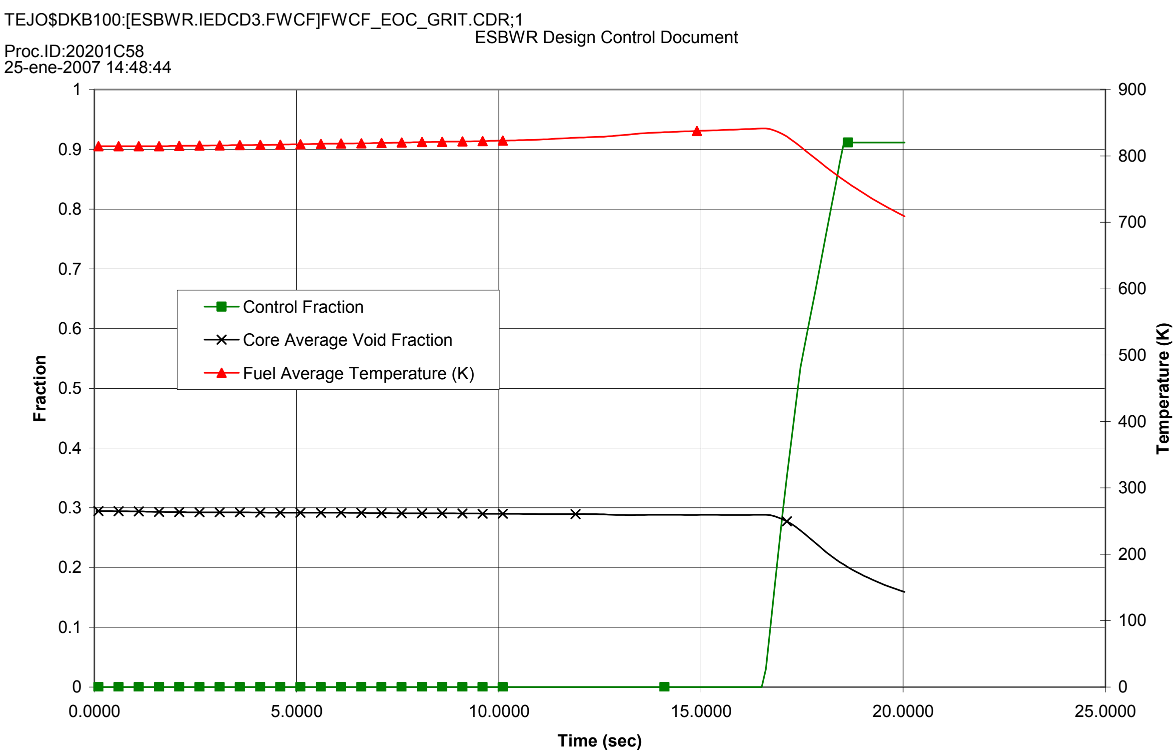


Figure 15.3-2f. Feedwater Controller Failure – Maximum Demand

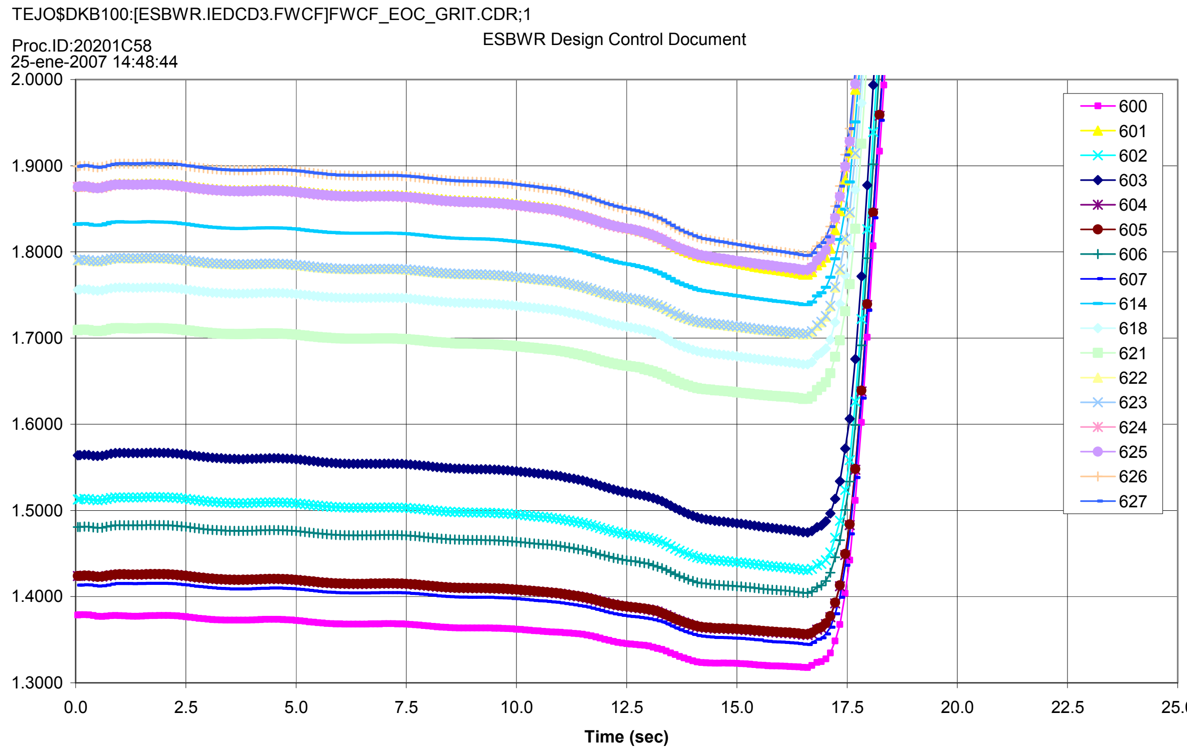


Figure 15.3-2g. Feedwater Controller Failure – Maximum Demand

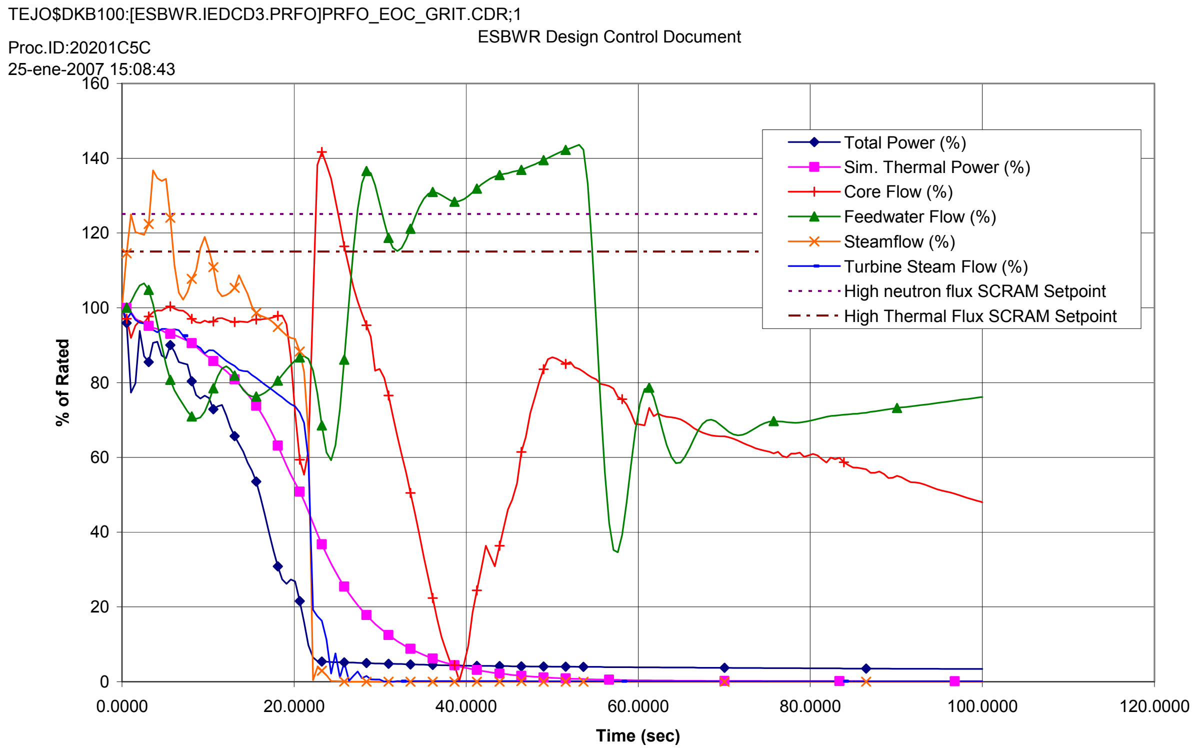


Figure 15.3-3a. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

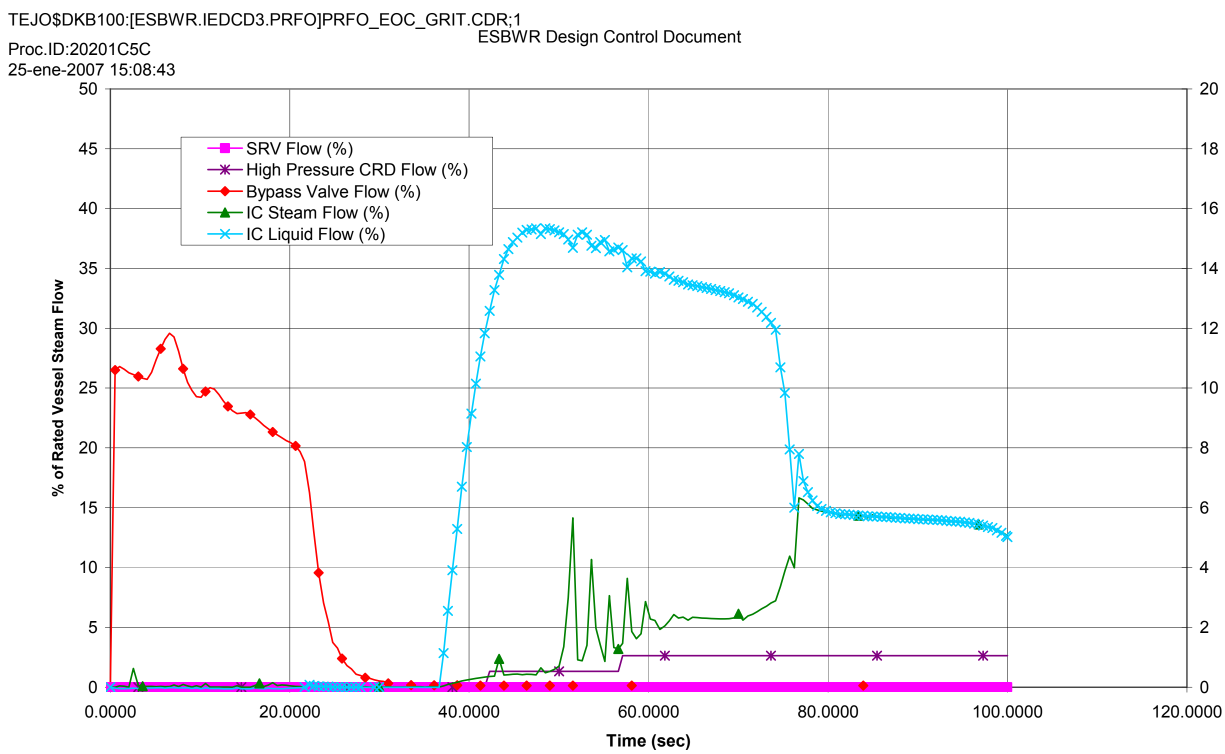


Figure 15.3-3b. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

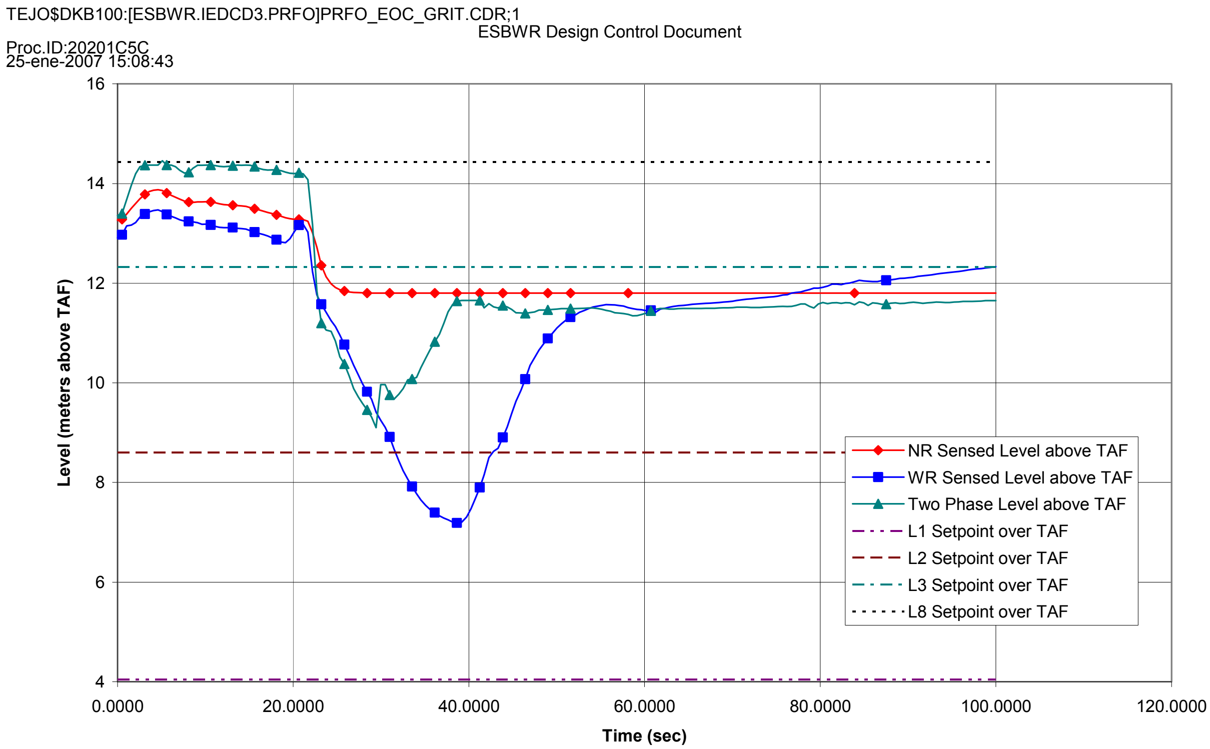


Figure 15.3-3c. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

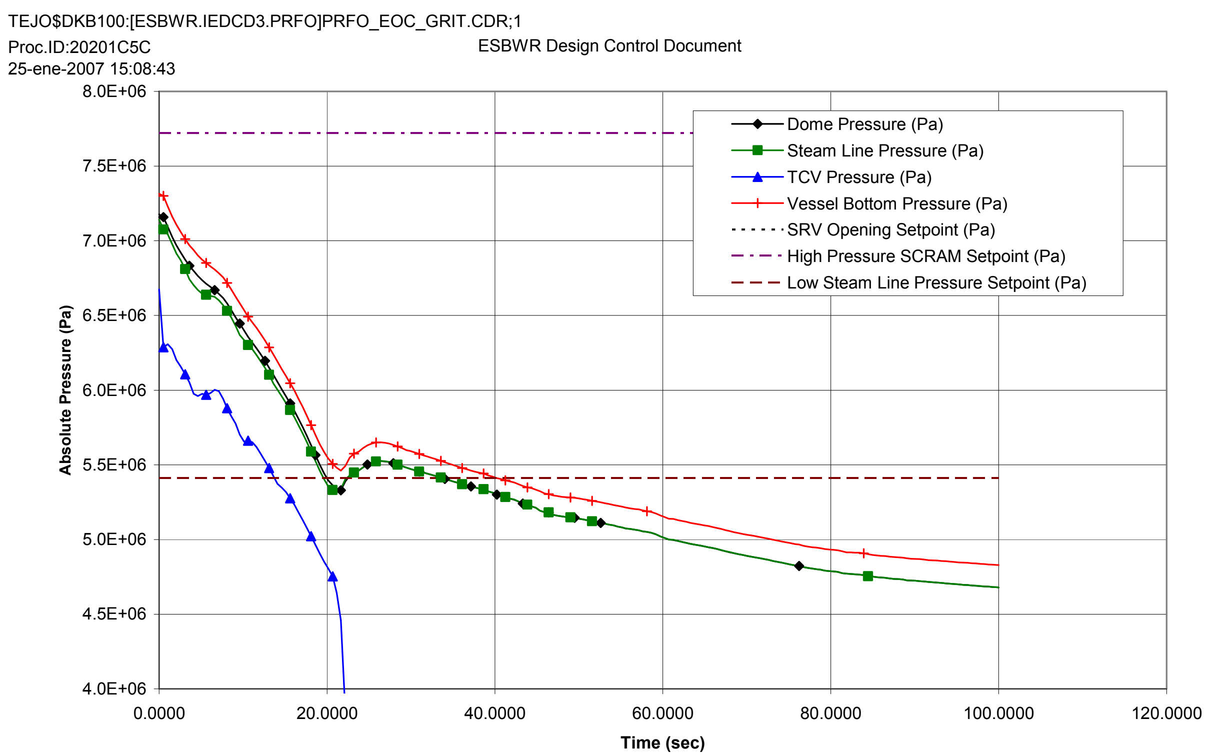


Figure 15.3-3d. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

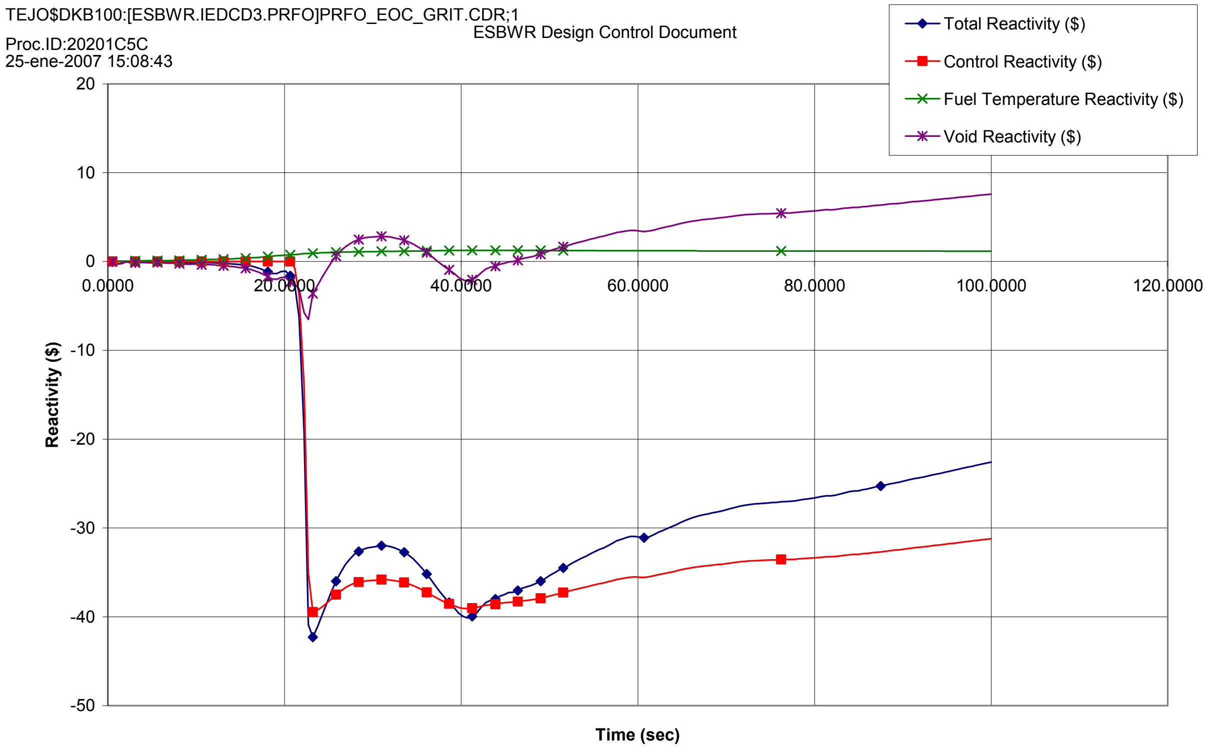


Figure 15.3-3e. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

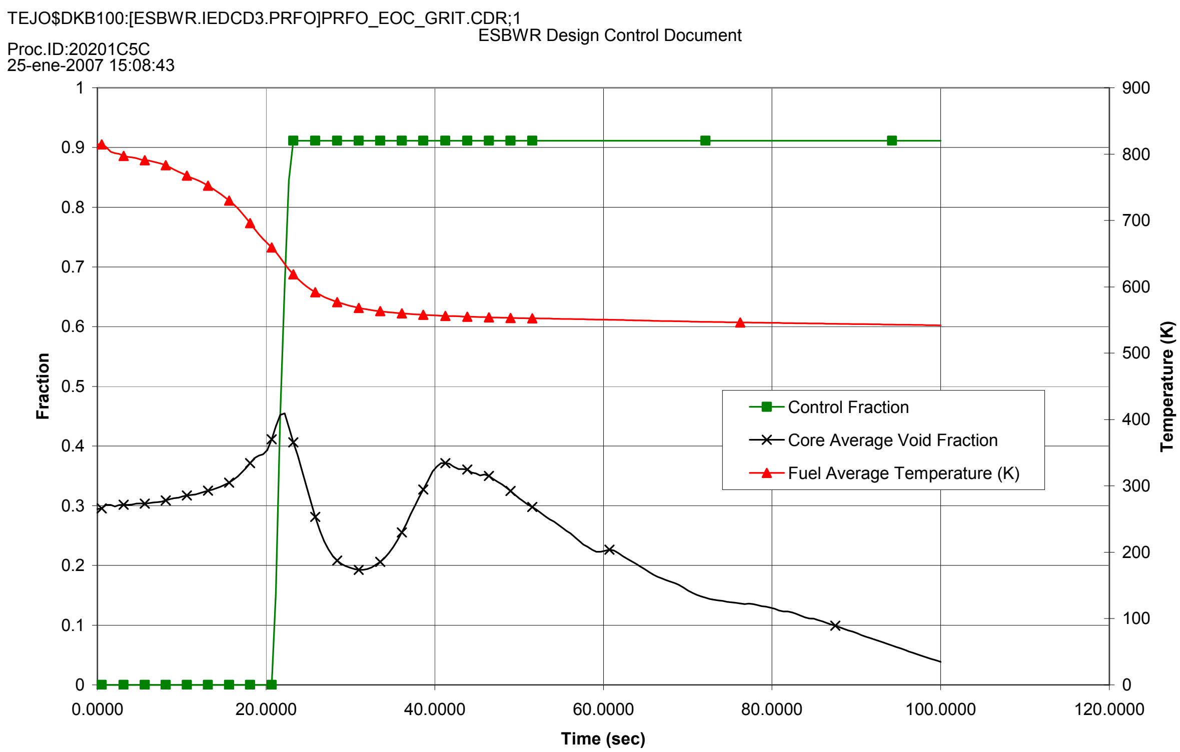


Figure 15.3-3f. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

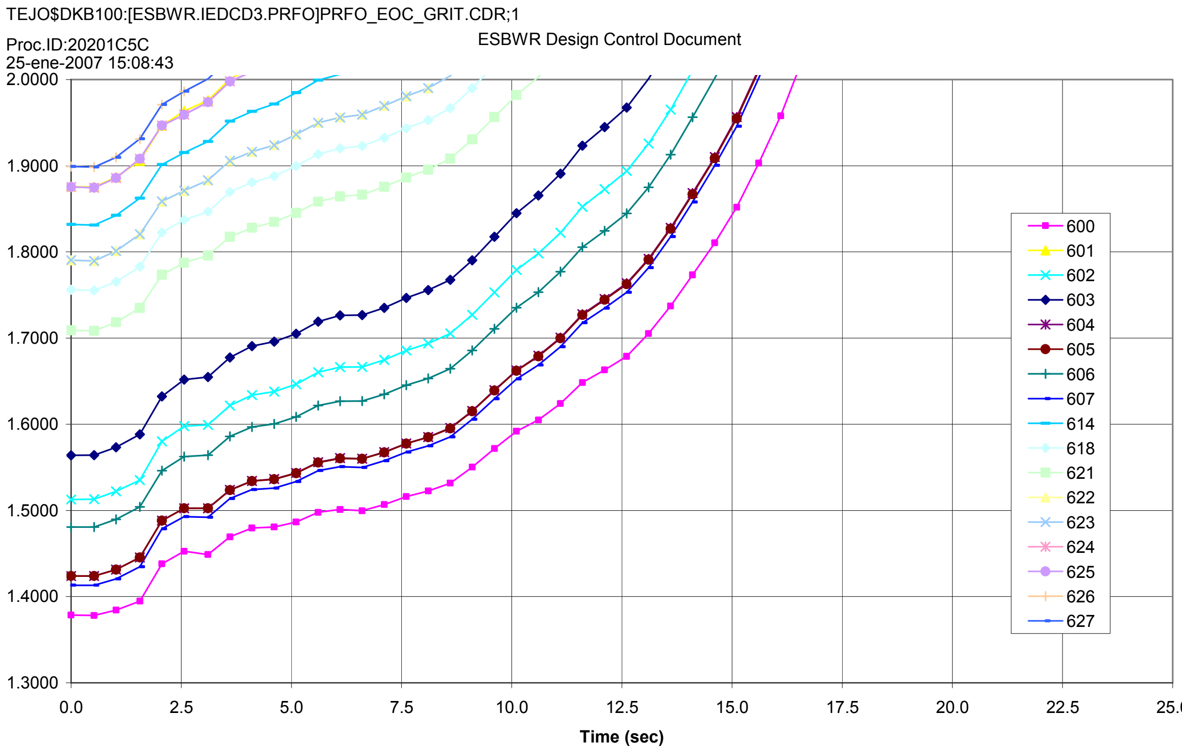


Figure 15.3-3g. Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

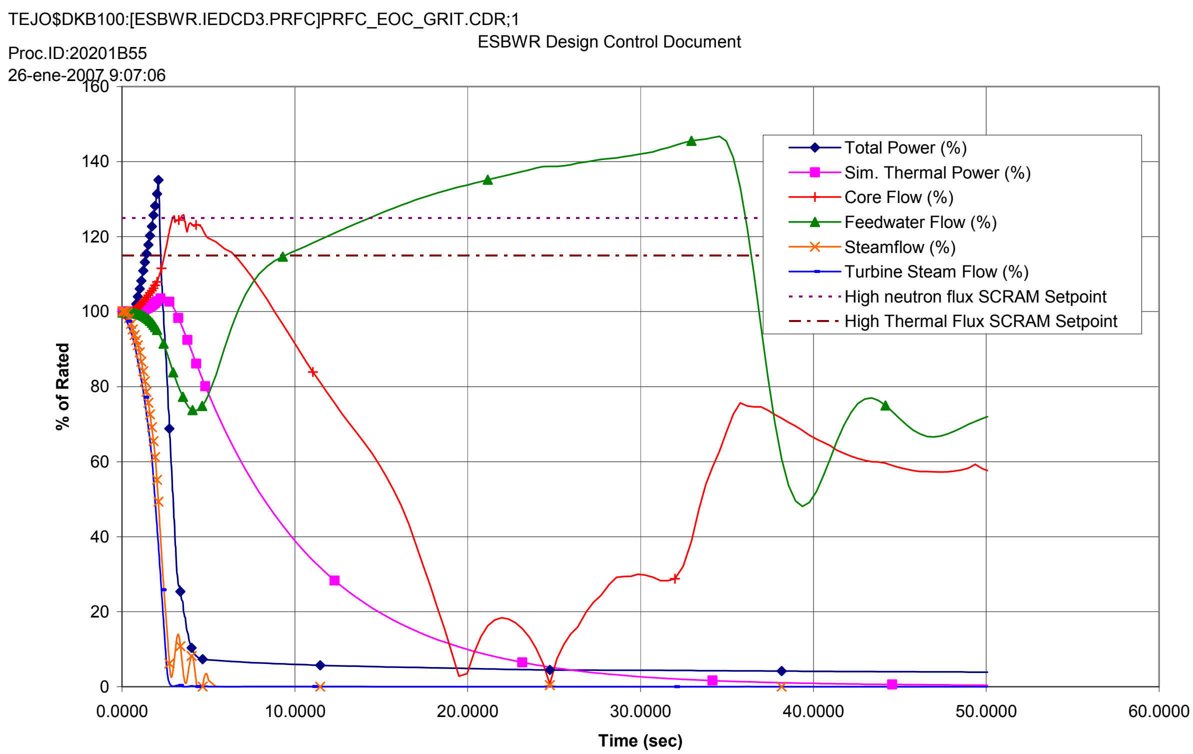


Figure 15.3-4a. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

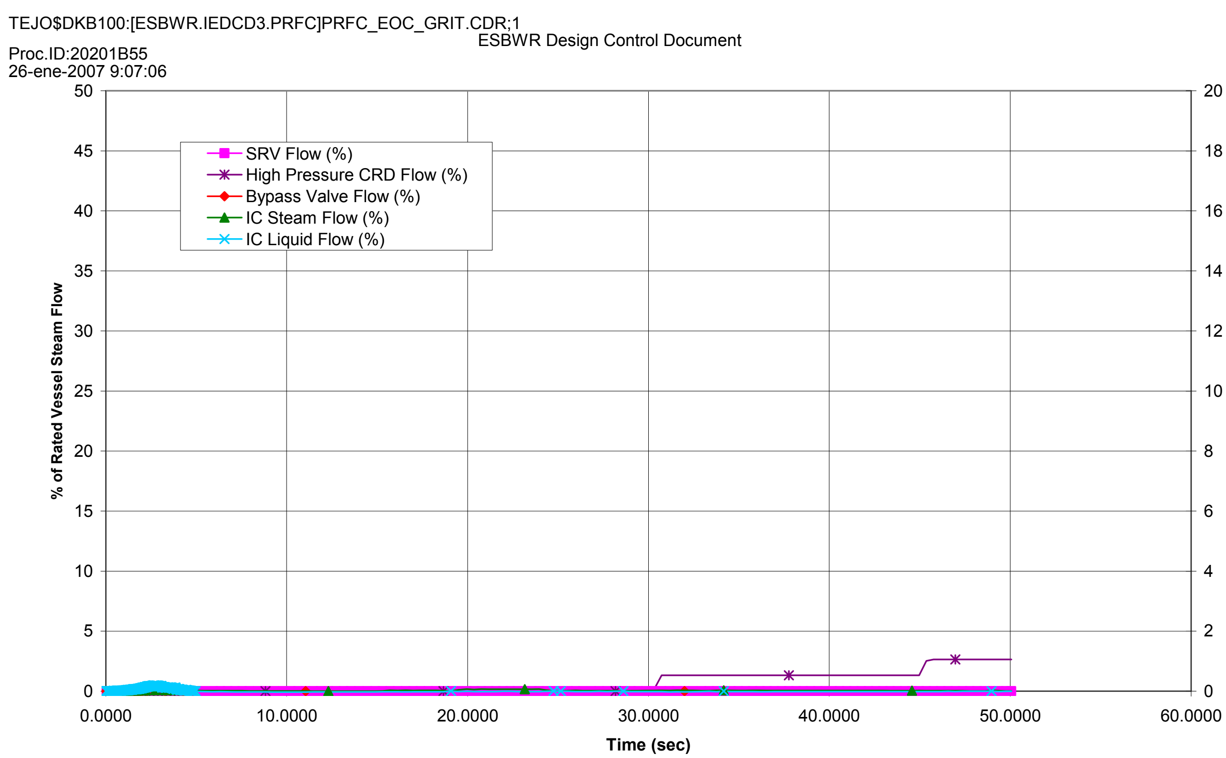


Figure 15.3-4b. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

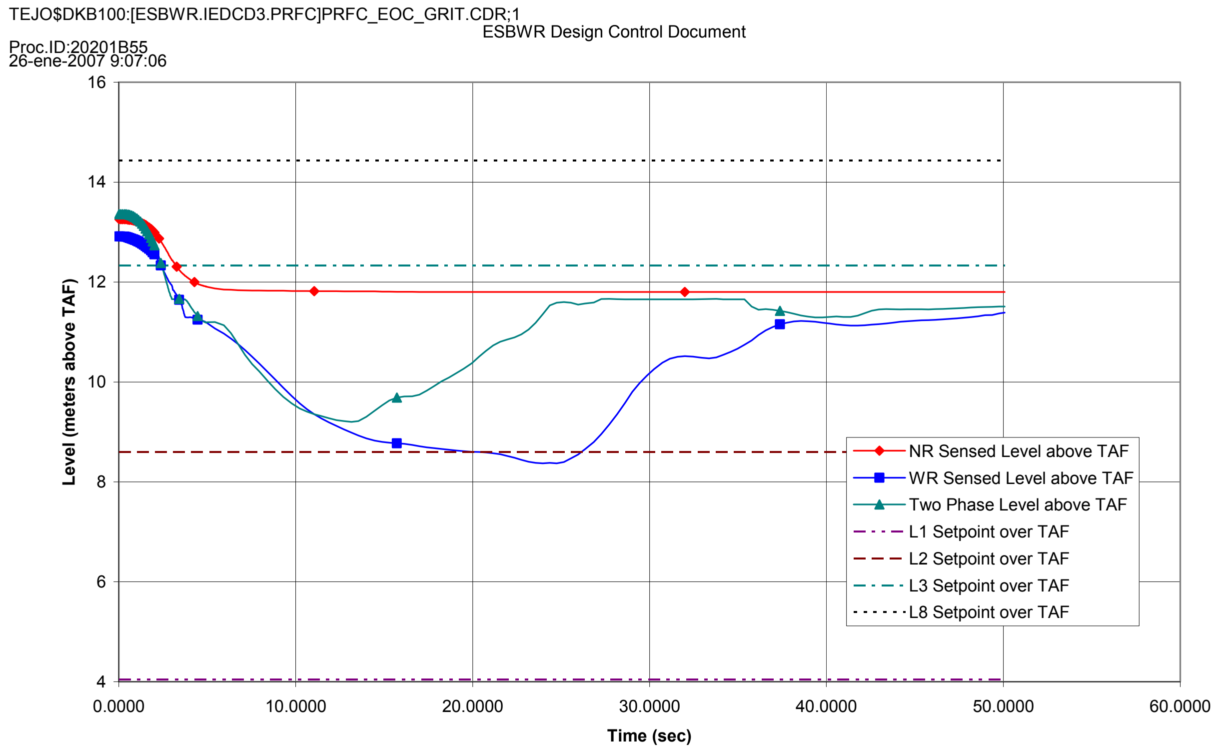


Figure 15.3-4c. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

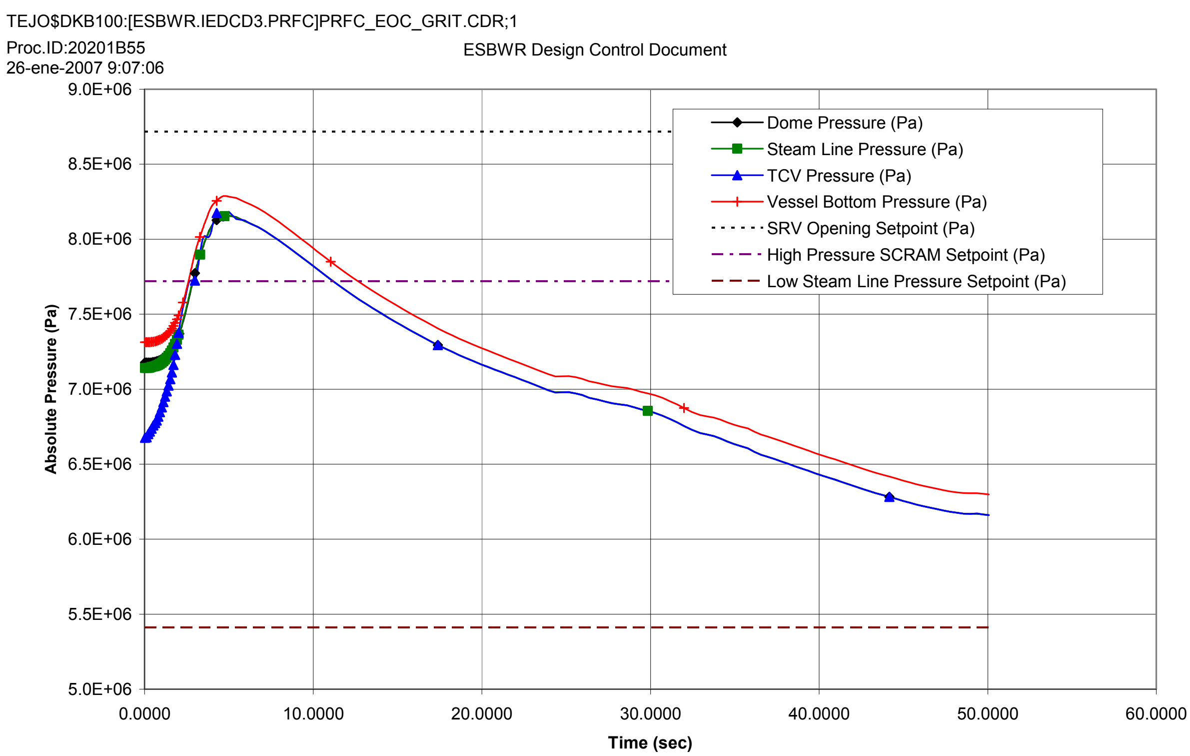


Figure 15.3-4d. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

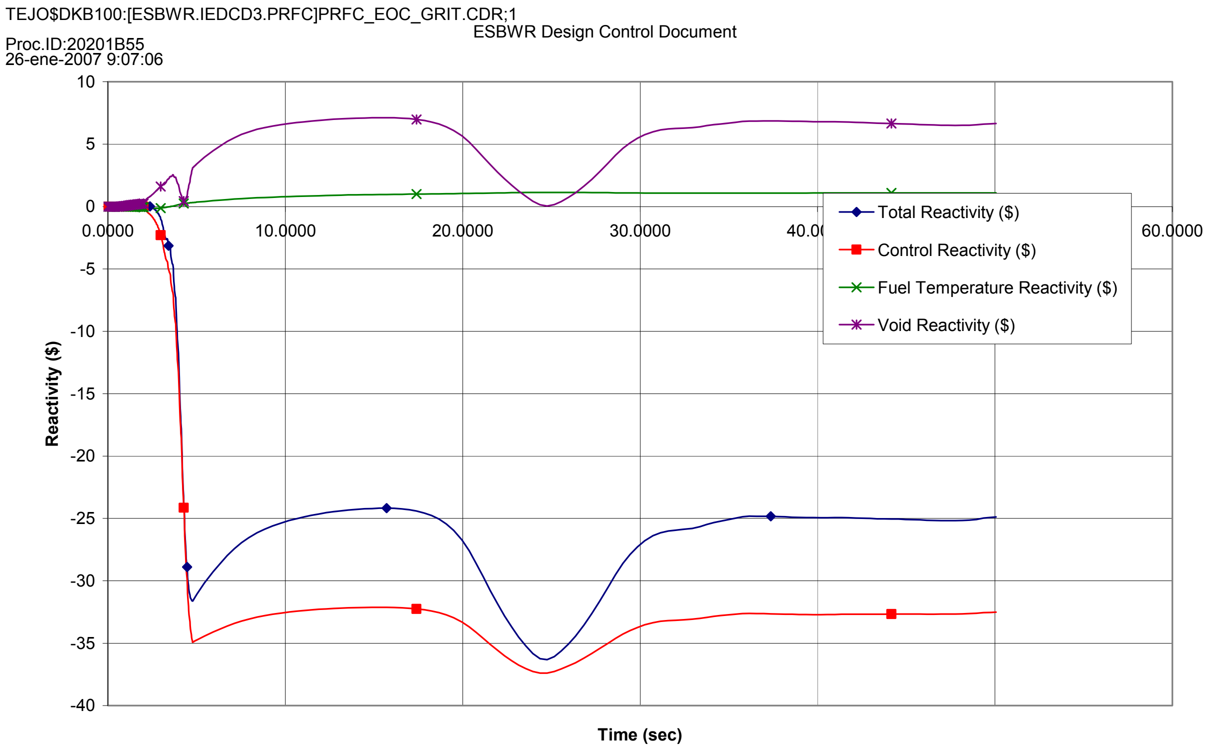


Figure 15.3-4e. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

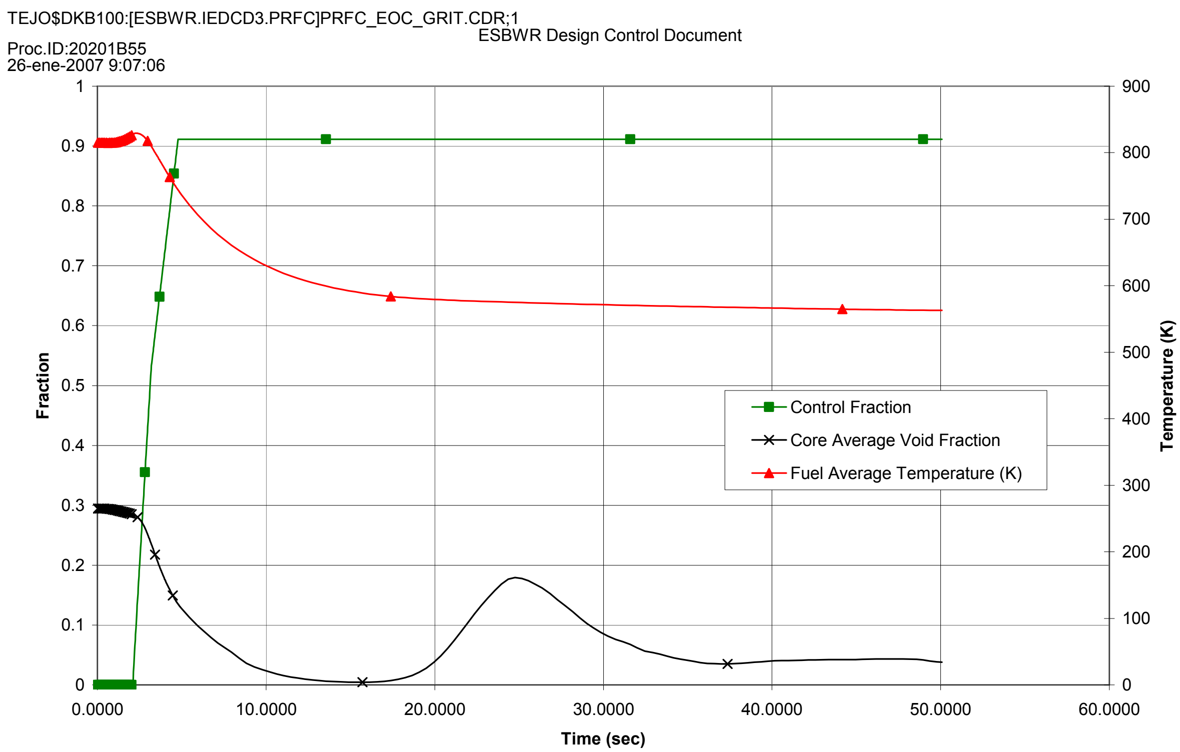


Figure 15.3-4f. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

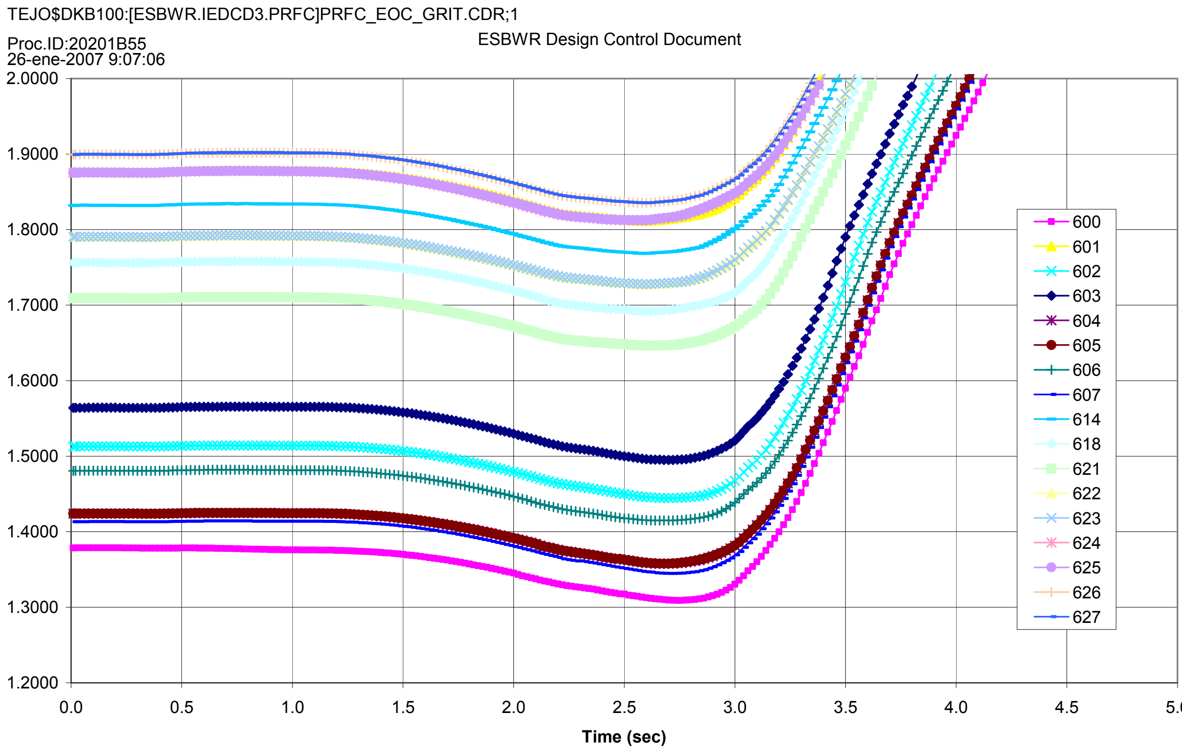


Figure 15.3-4g. Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

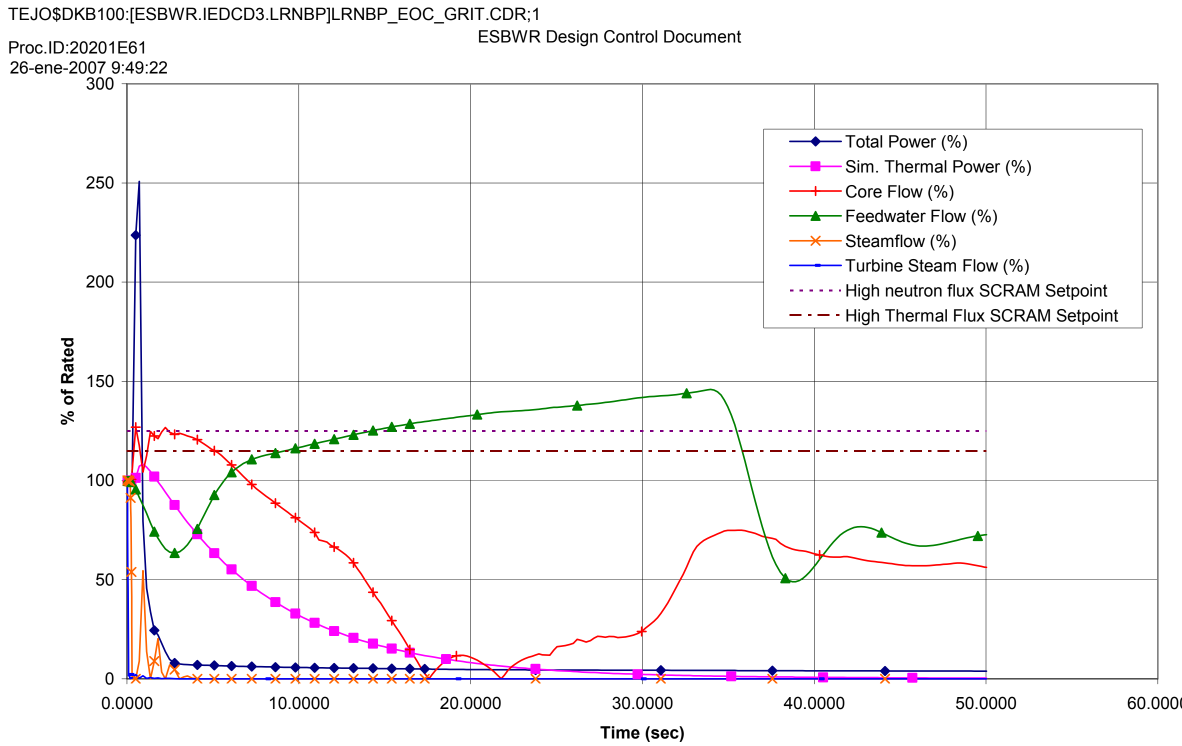


Figure 15.3-5a. Generator Load Rejection With Total Turbine Bypass Failure

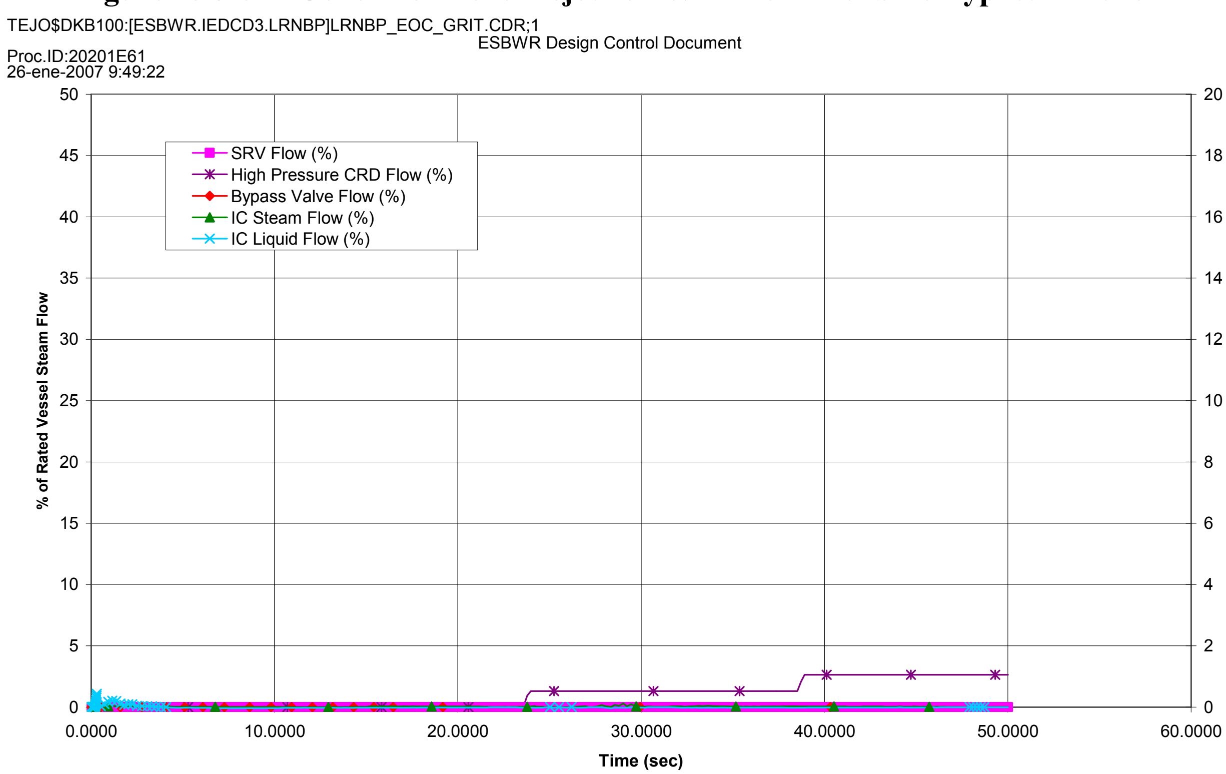


Figure 15.3-5b. Generator Load Rejection With Total Turbine Bypass Failure

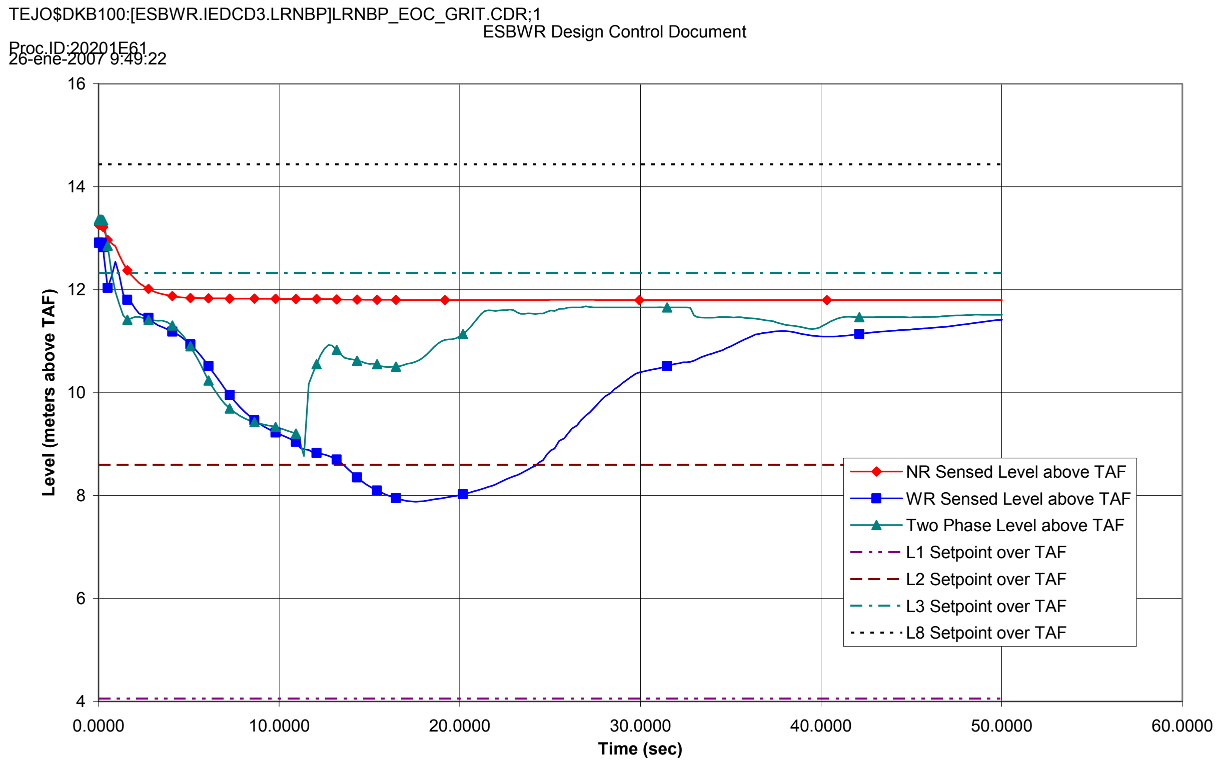


Figure 15.3-5c. Generator Load Rejection With Total Turbine Bypass Failure

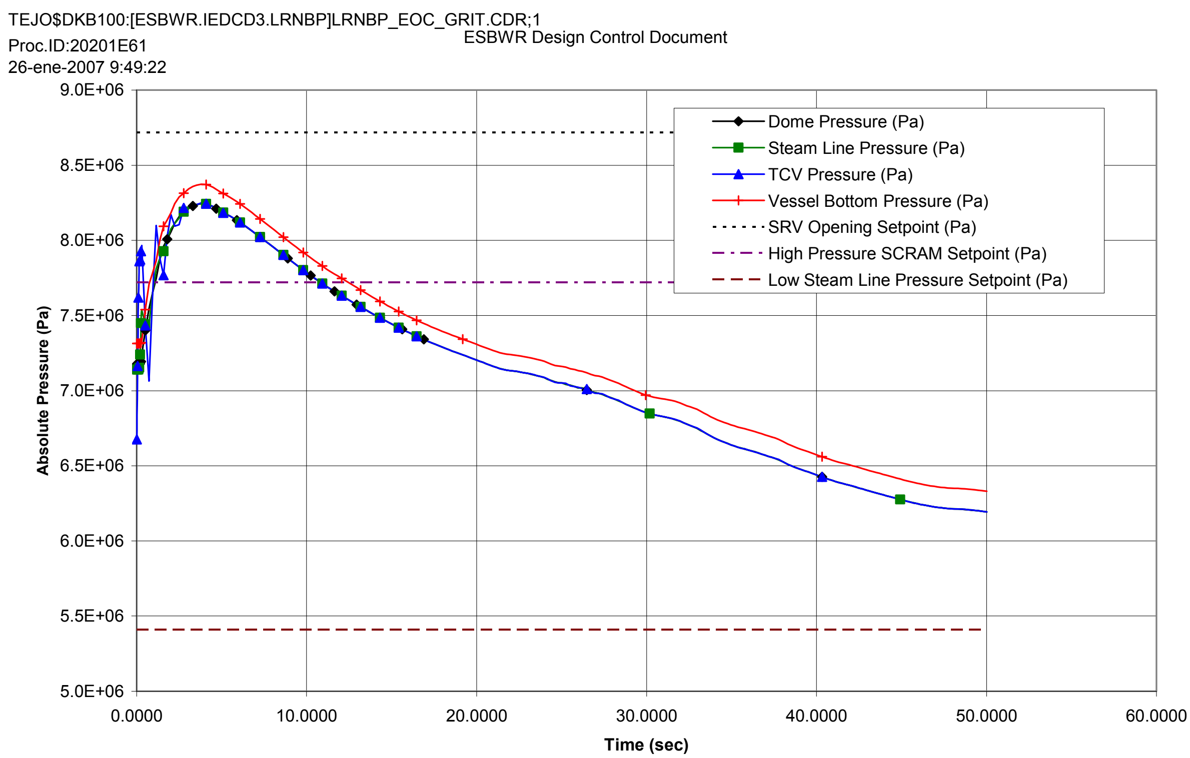


Figure 15.3-5d. Generator Load Rejection With Total Turbine Bypass Failure

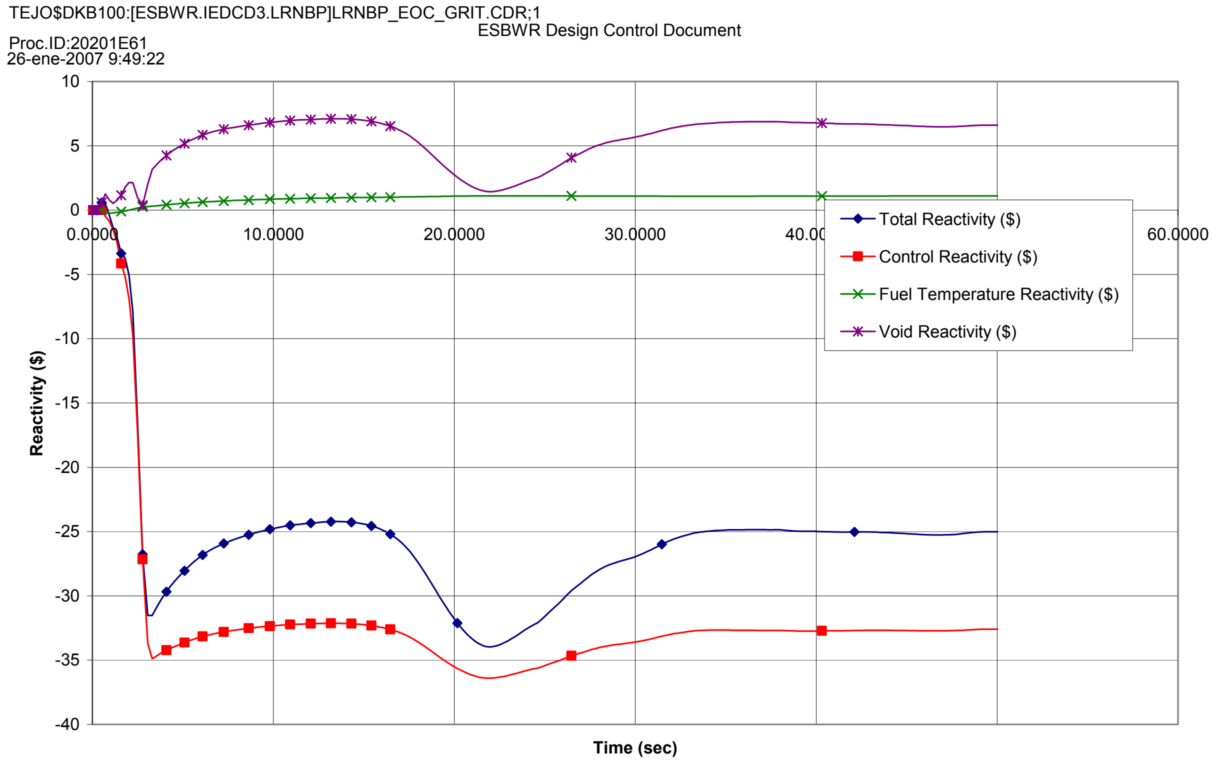


Figure 15.3-5e. Generator Load Rejection With Total Turbine Bypass Failure

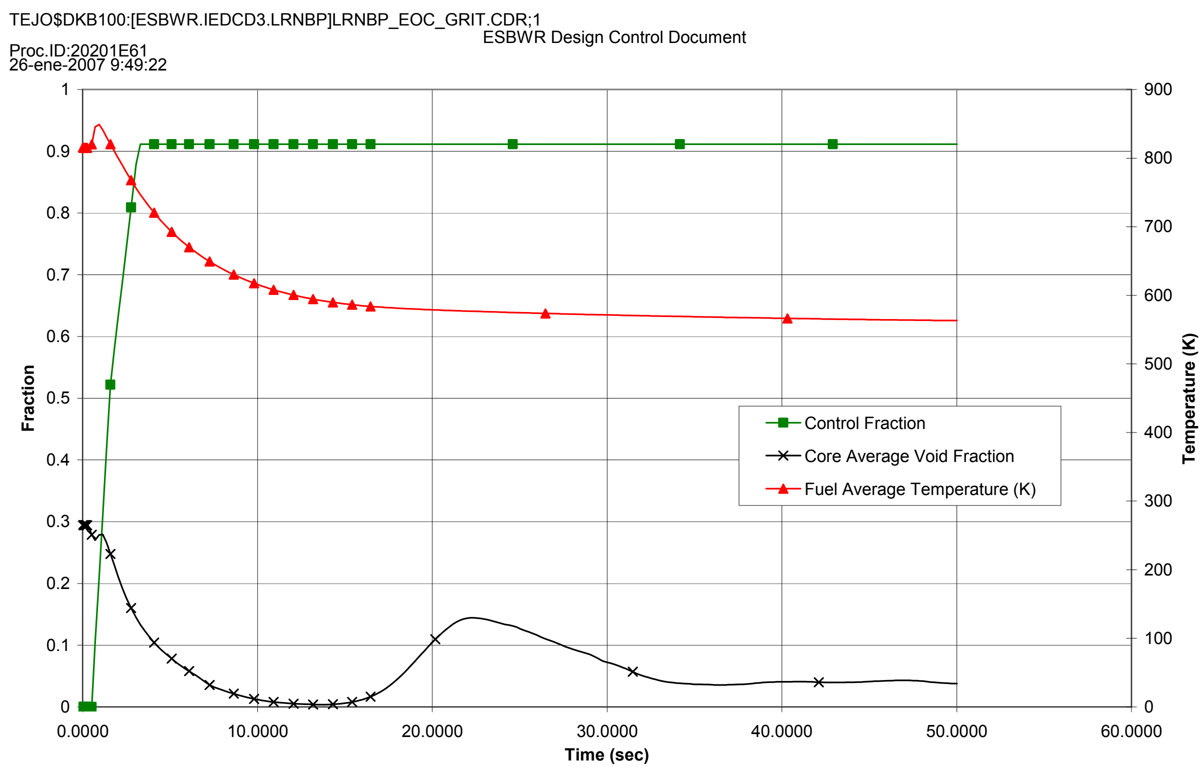


Figure 15.3-5f. Generator Load Rejection With Total Turbine Bypass Failure

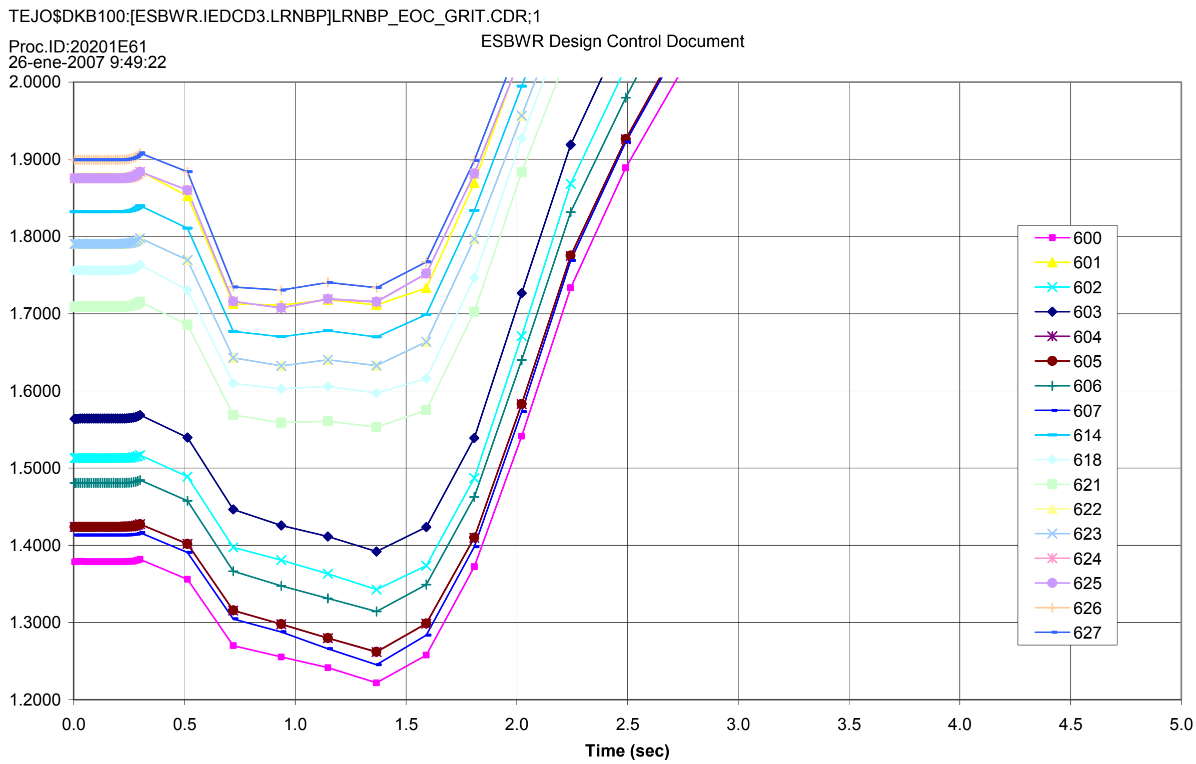


Figure 15.3-5g. Generator Load Rejection With Total Turbine Bypass Failure

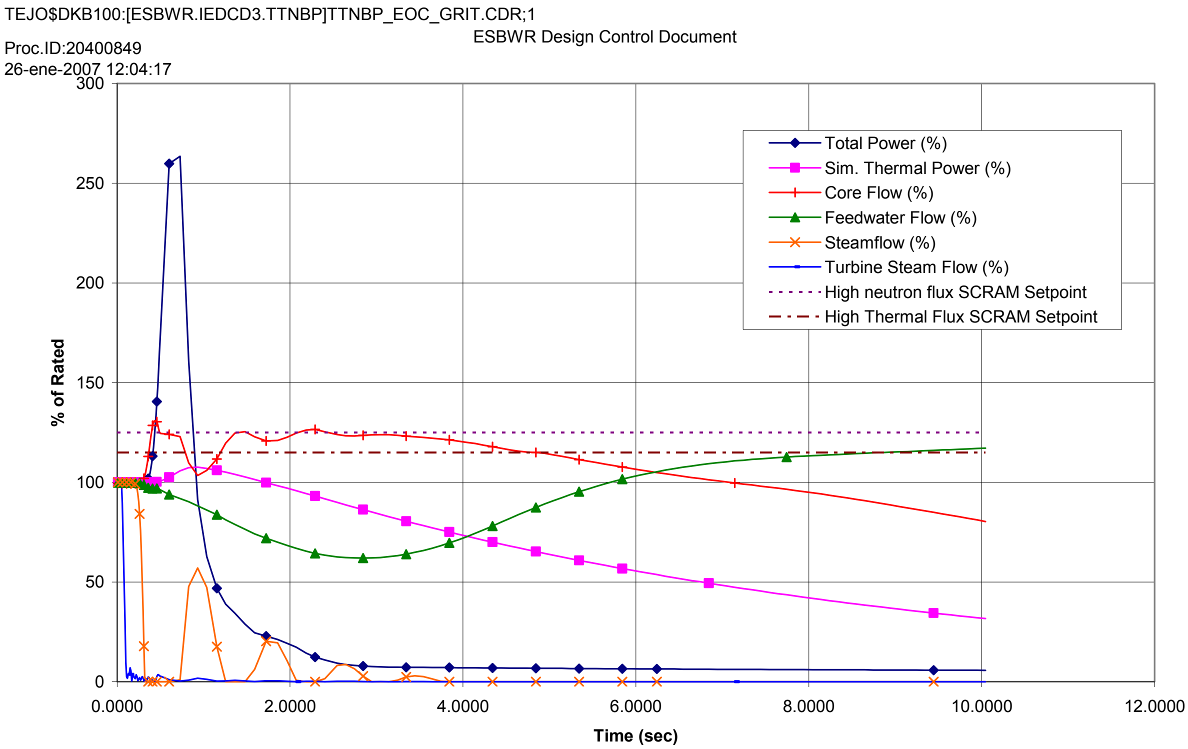


Figure 15.3-6a. Turbine Trip With Total Turbine Bypass Failure

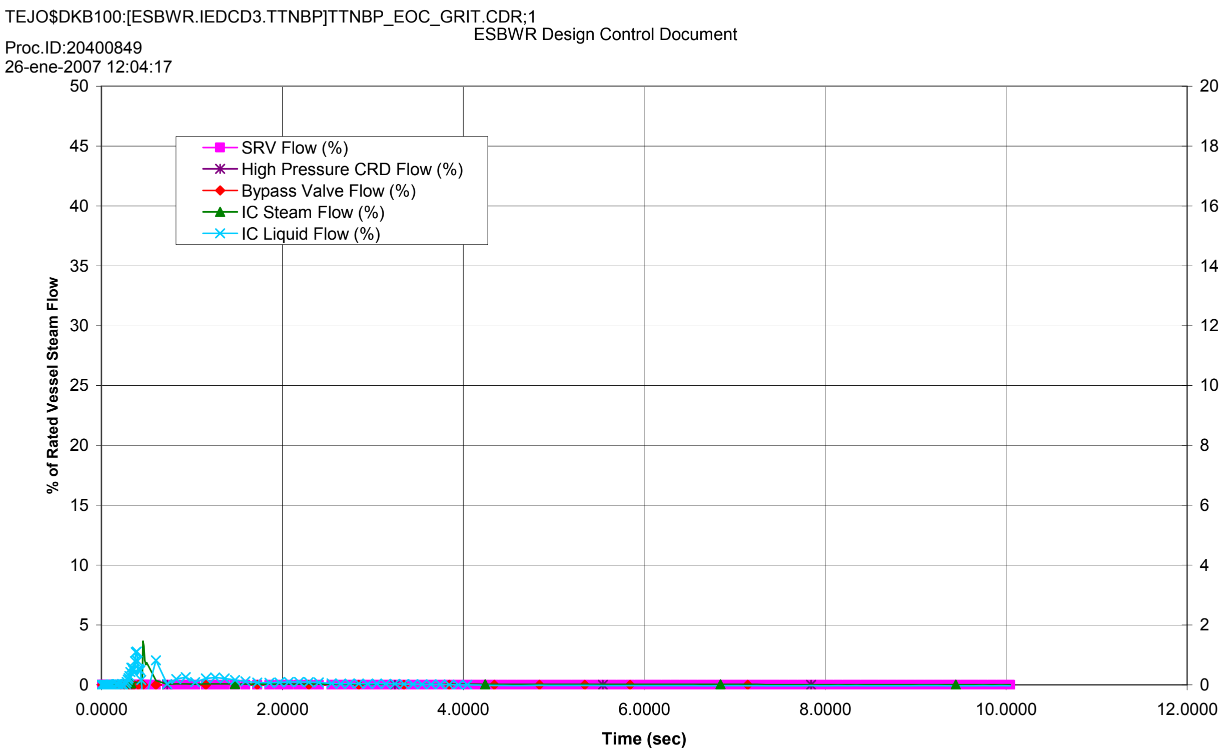


Figure 15.3-6b. Turbine Trip With Total Turbine Bypass Failure

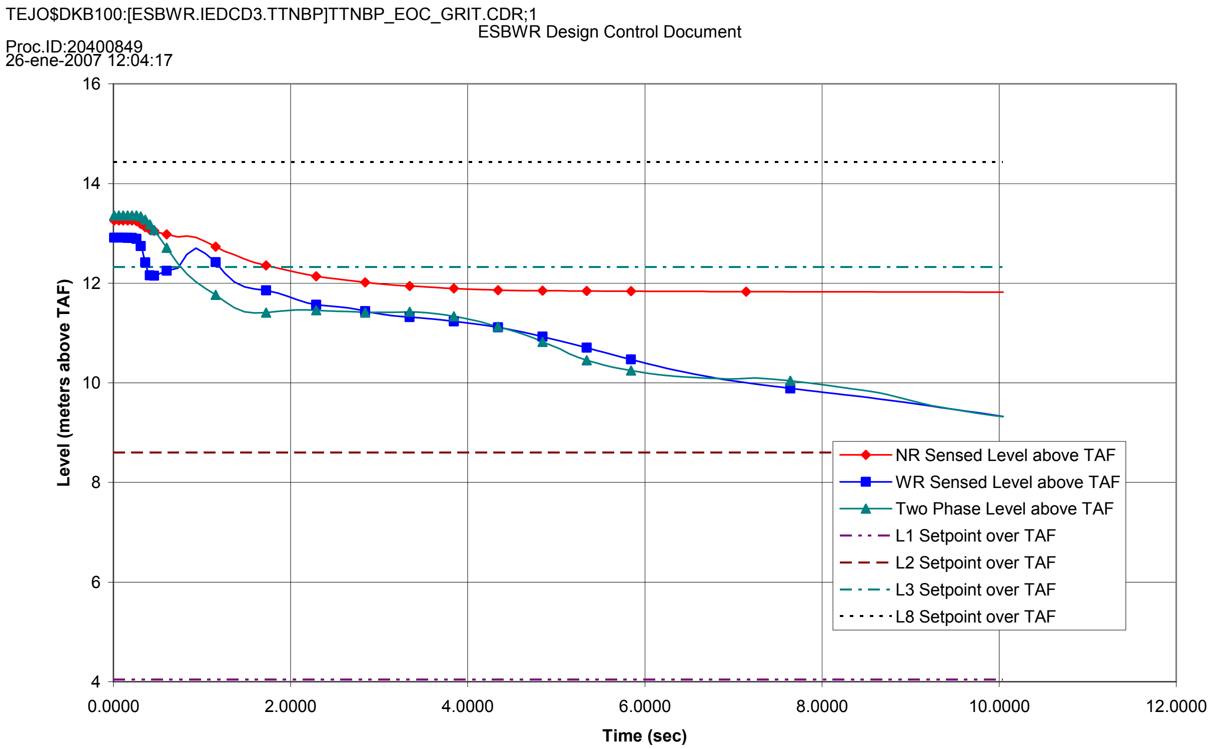


Figure 15.3-6c. Turbine Trip With Total Turbine Bypass Failure

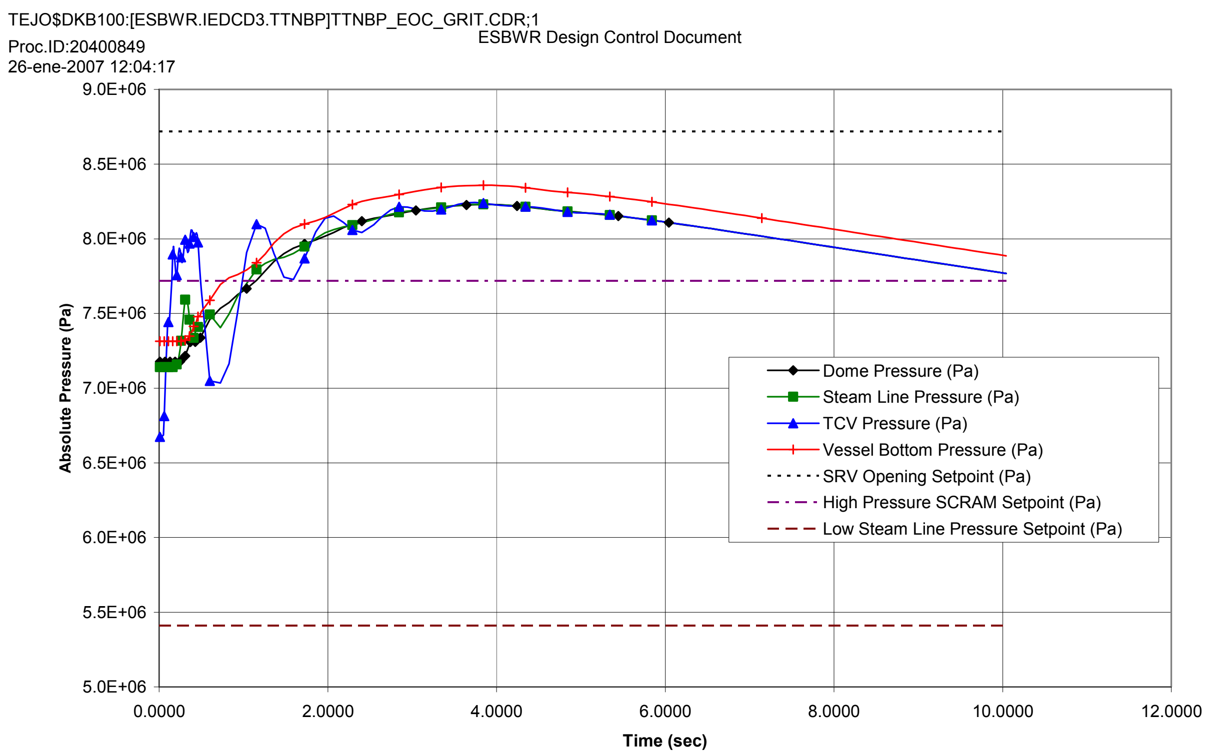


Figure 15.3-6d. Turbine Trip With Total Turbine Bypass Failure

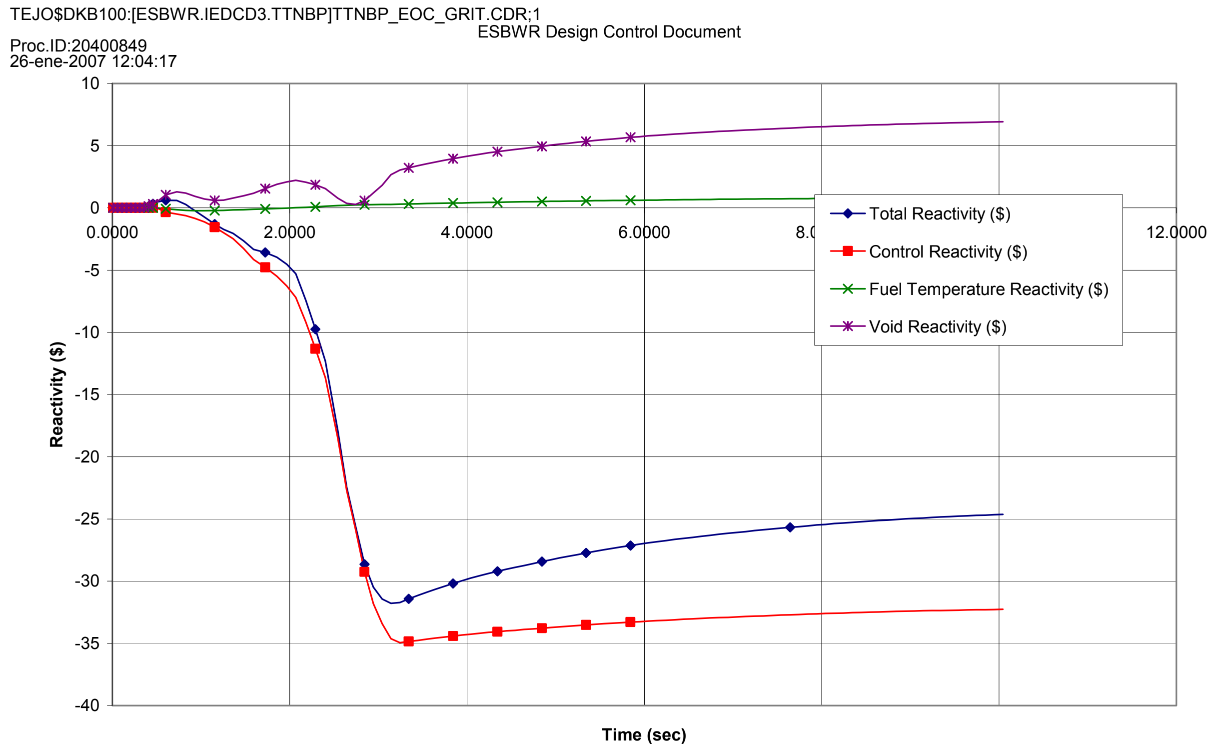


Figure 15.3-6e. Turbine Trip With Total Turbine Bypass Failure

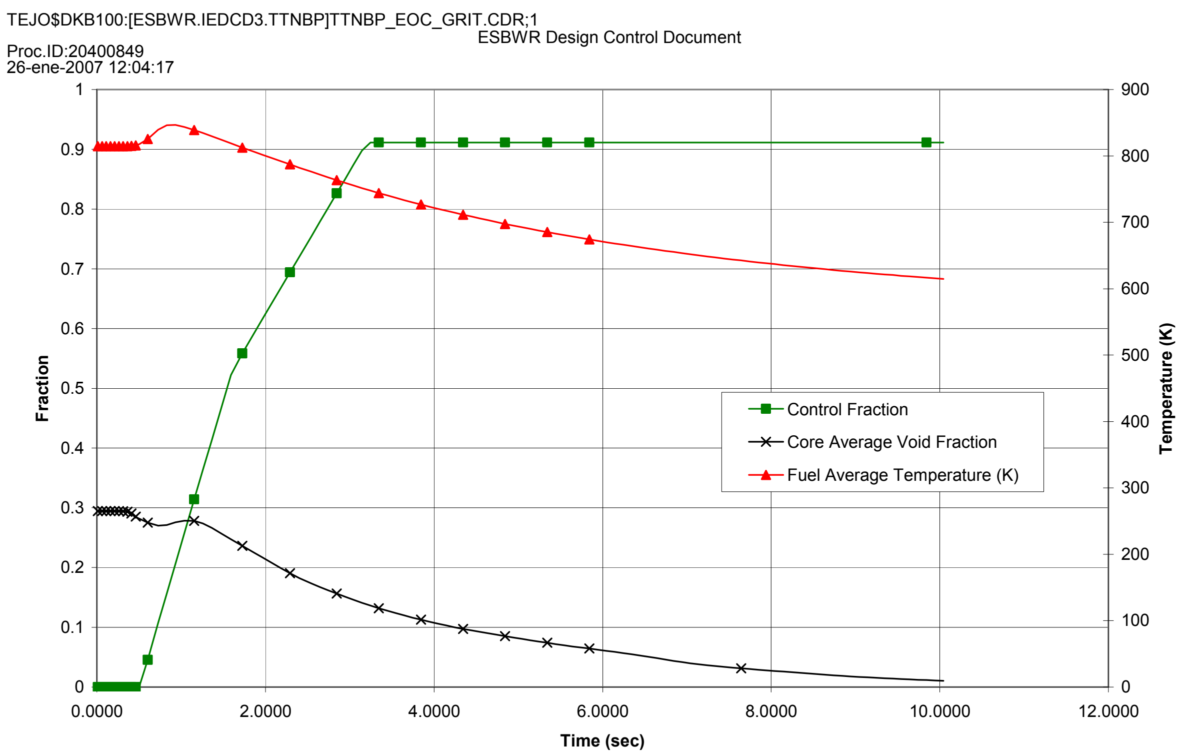


Figure 15.3-6f. Turbine Trip With Total Turbine Bypass Failure

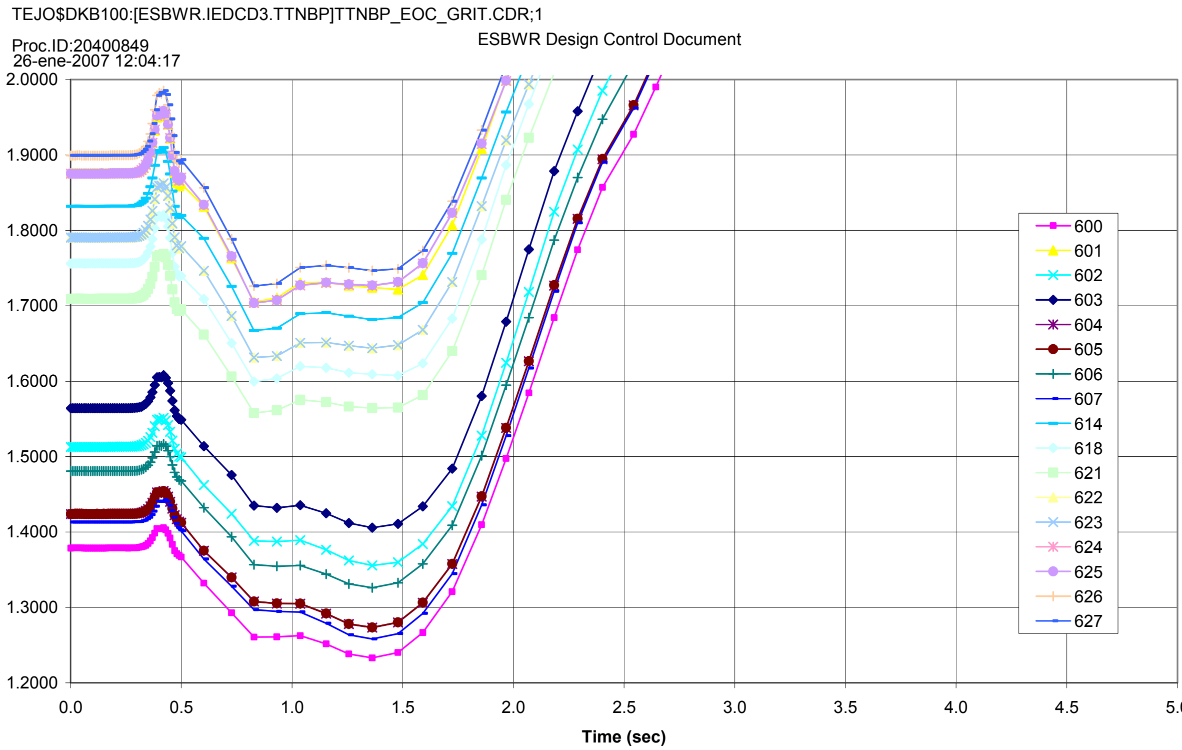


Figure 15.3-6g. Turbine Trip With Total Turbine Bypass Failure

Figure 15.3-7. (Deleted)

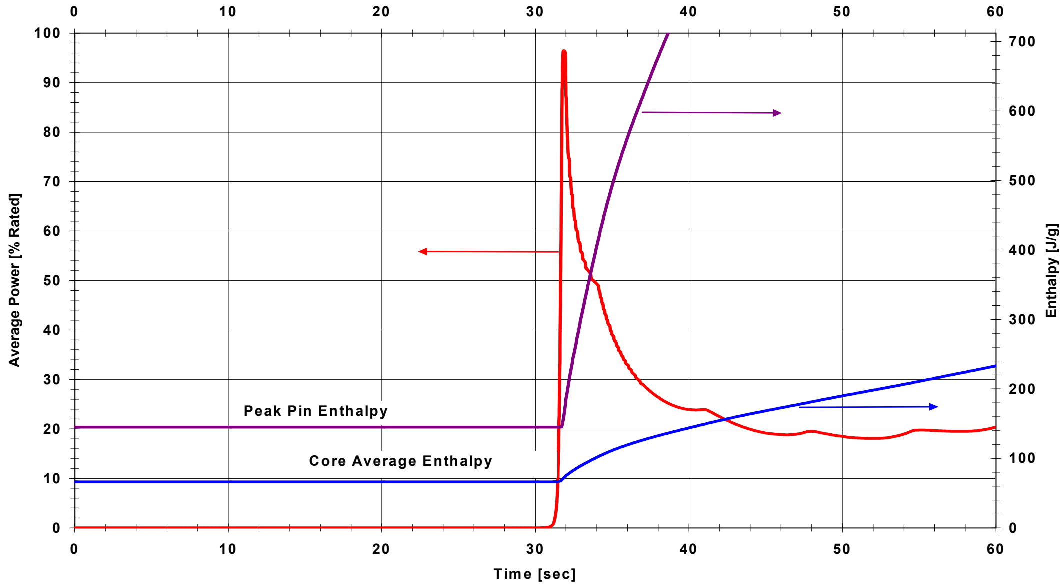


Figure 15.3-7a. Transient Changes for Control Rod Withdrawal Error During Startup

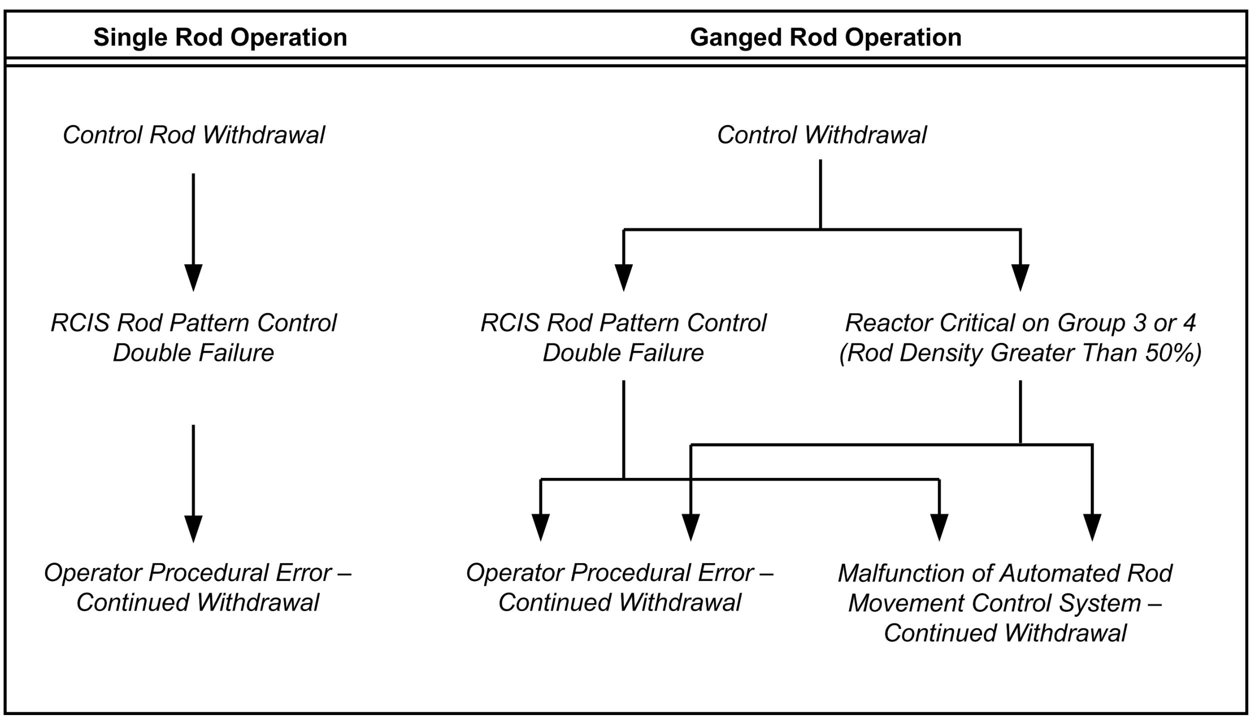


Figure 15.3-7b. Causes of Control Rod Withdrawal Error During Startup

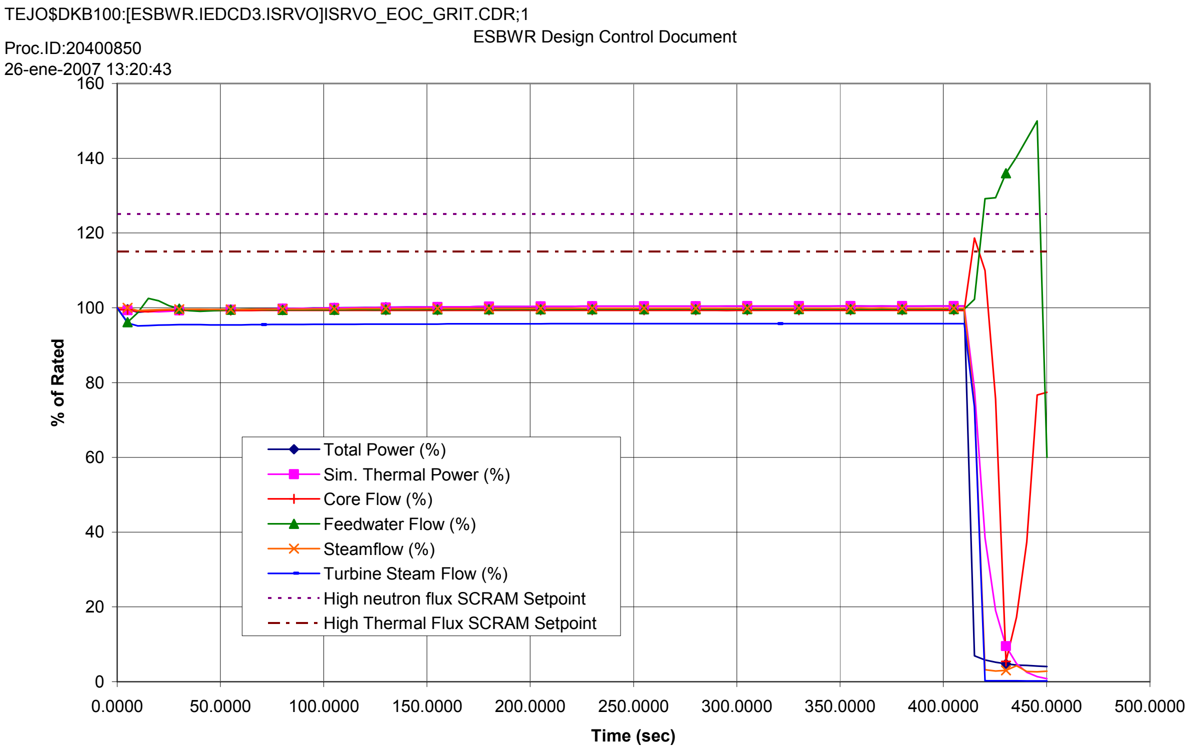


Figure 15.3-8a. Inadvertent SRV opening

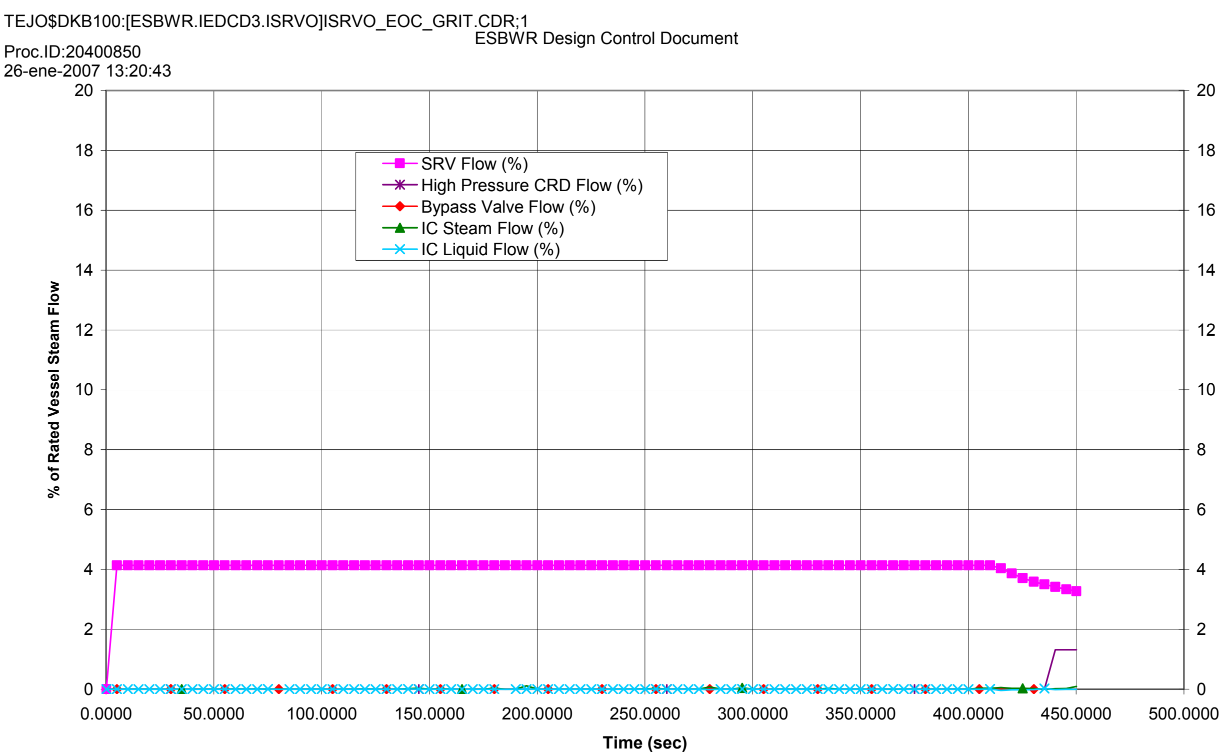


Figure 15.3-8b. Inadvertent SRV opening

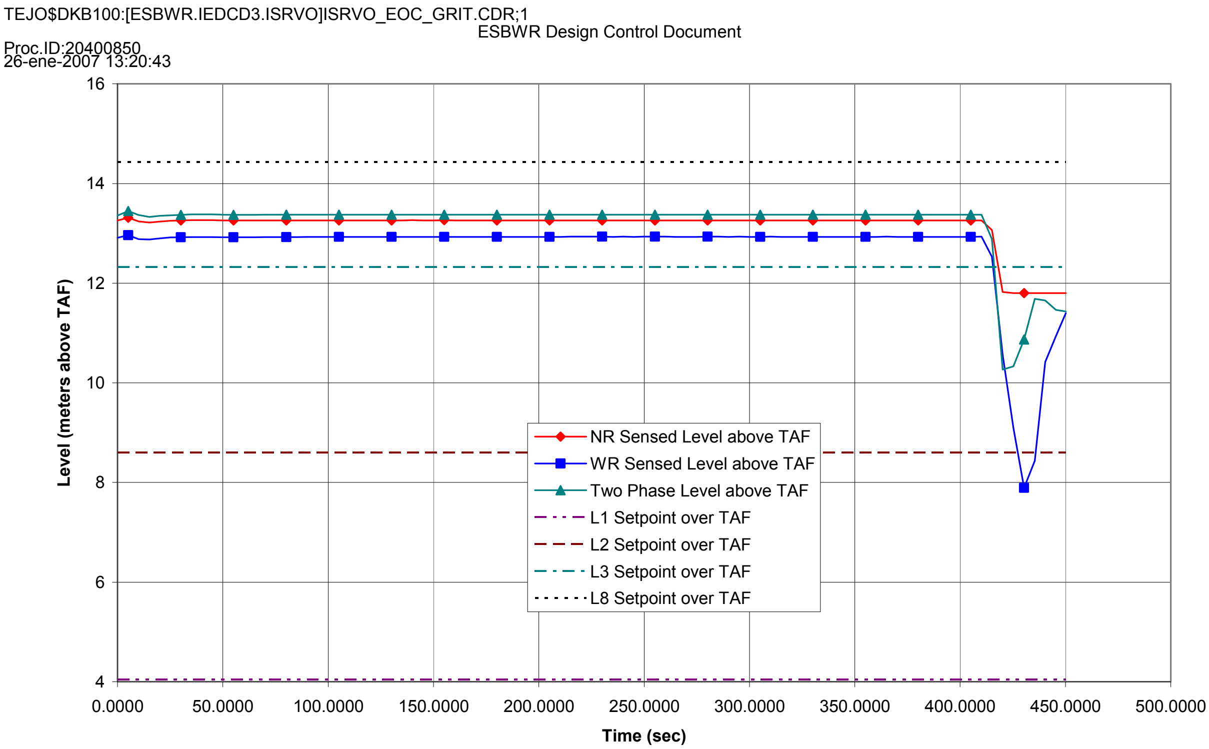


Figure 15.3-8c. Inadvertent SRV opening

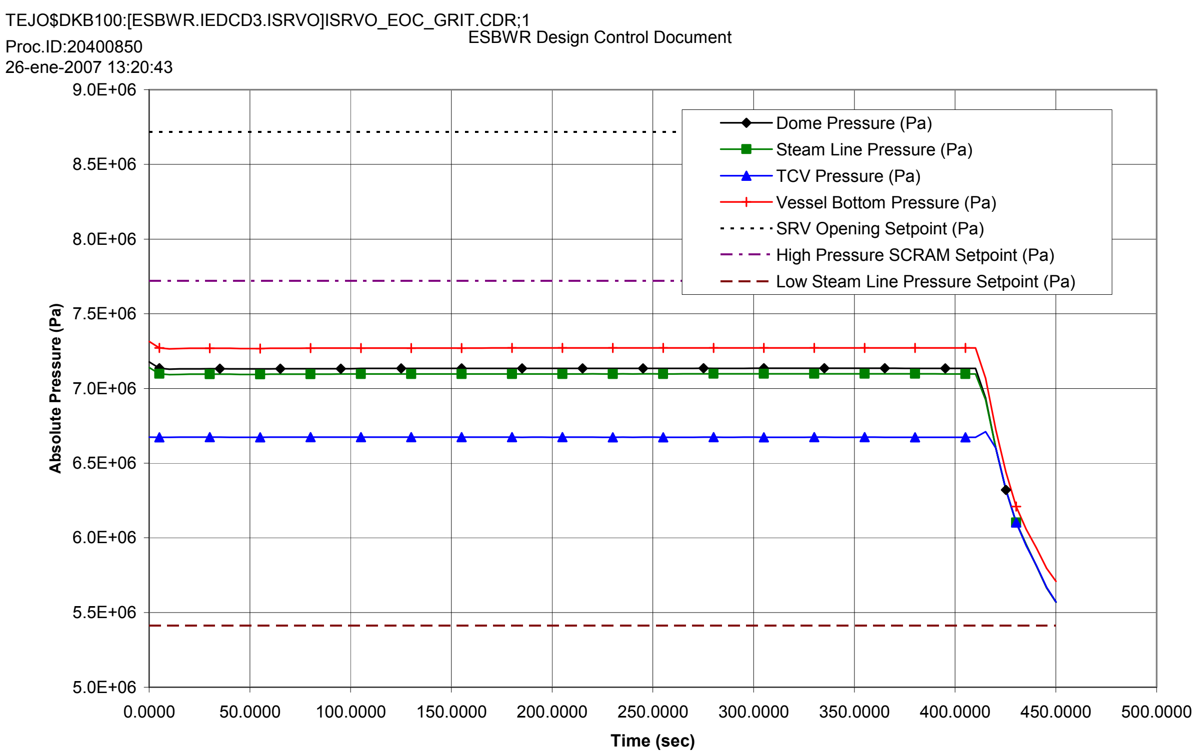


Figure 15.3-8d. Inadvertent SRV opening

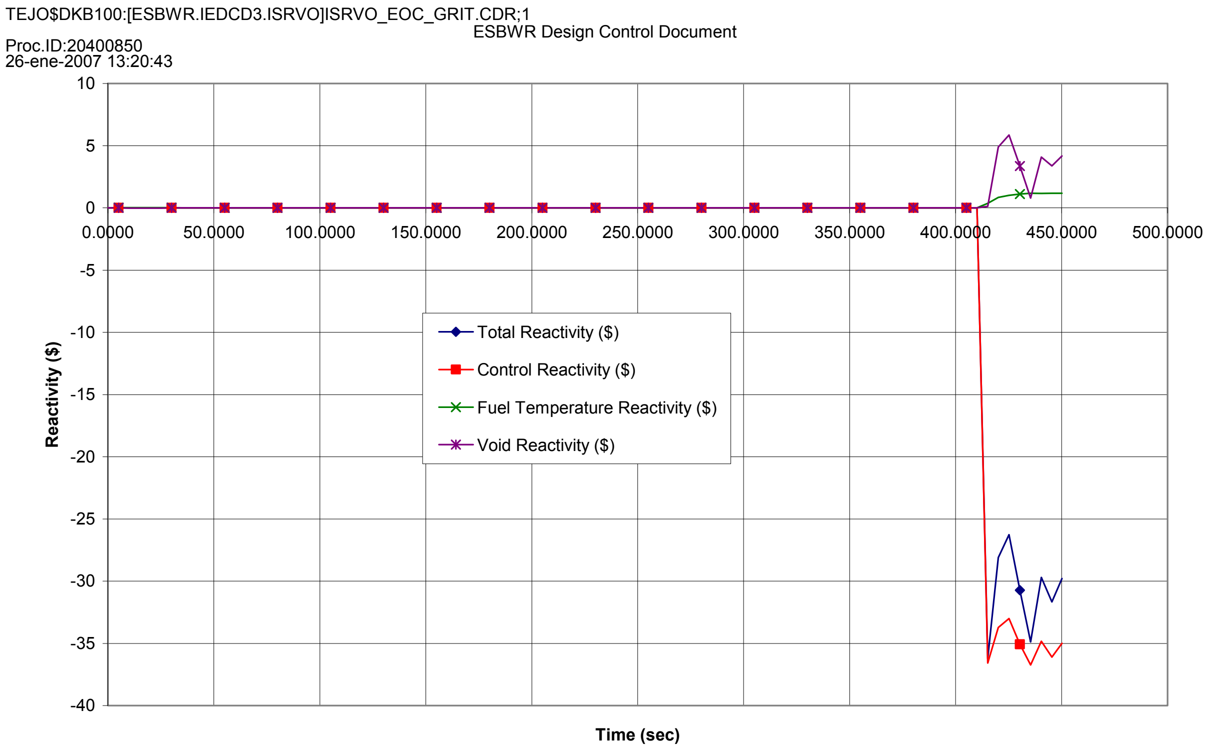


Figure 15.3-8e. Inadvertent SRV opening

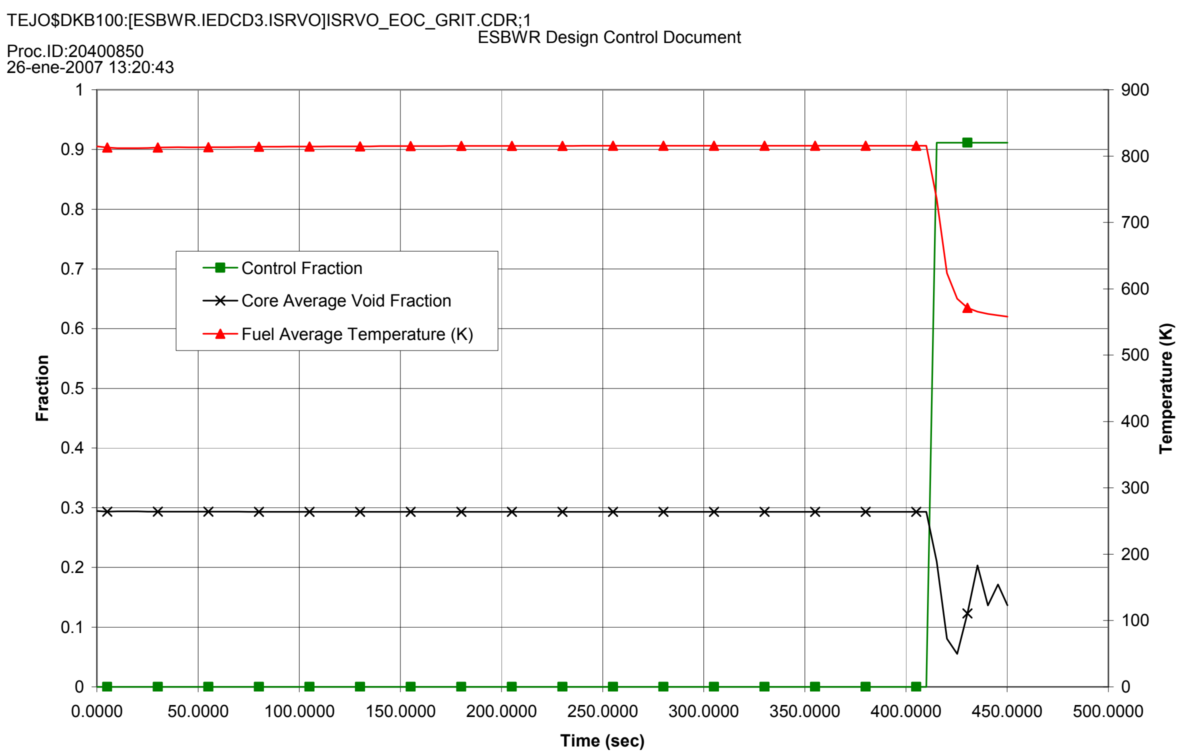


Figure 15.3-8f. Inadvertent SRV opening

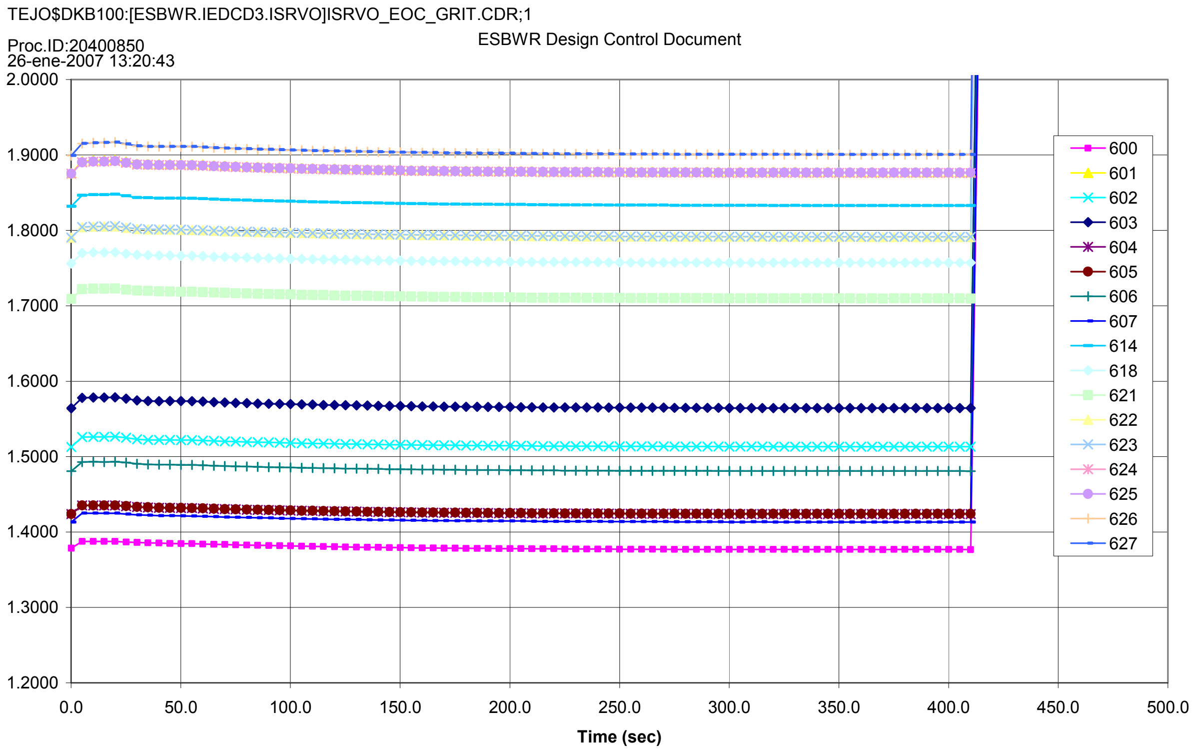


Figure 15.3-8g. Inadvertent SRV opening

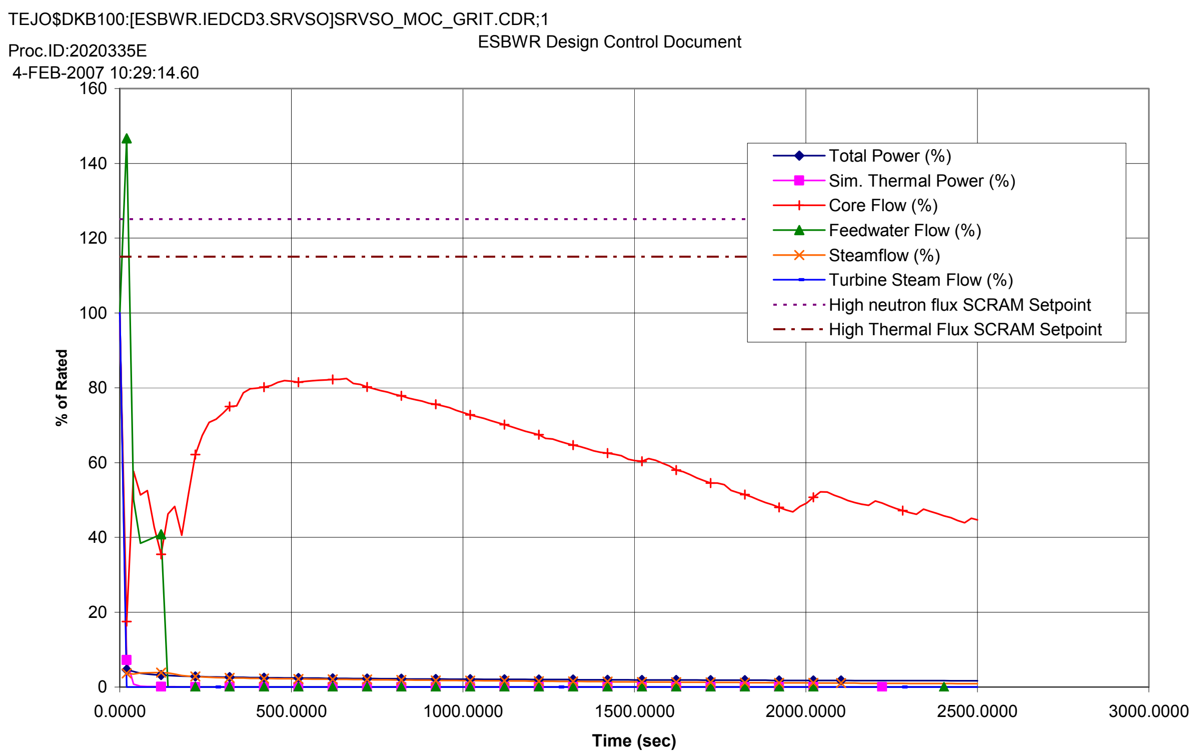


Figure 15.3-9a. Stuck Open Safety Relief Valve

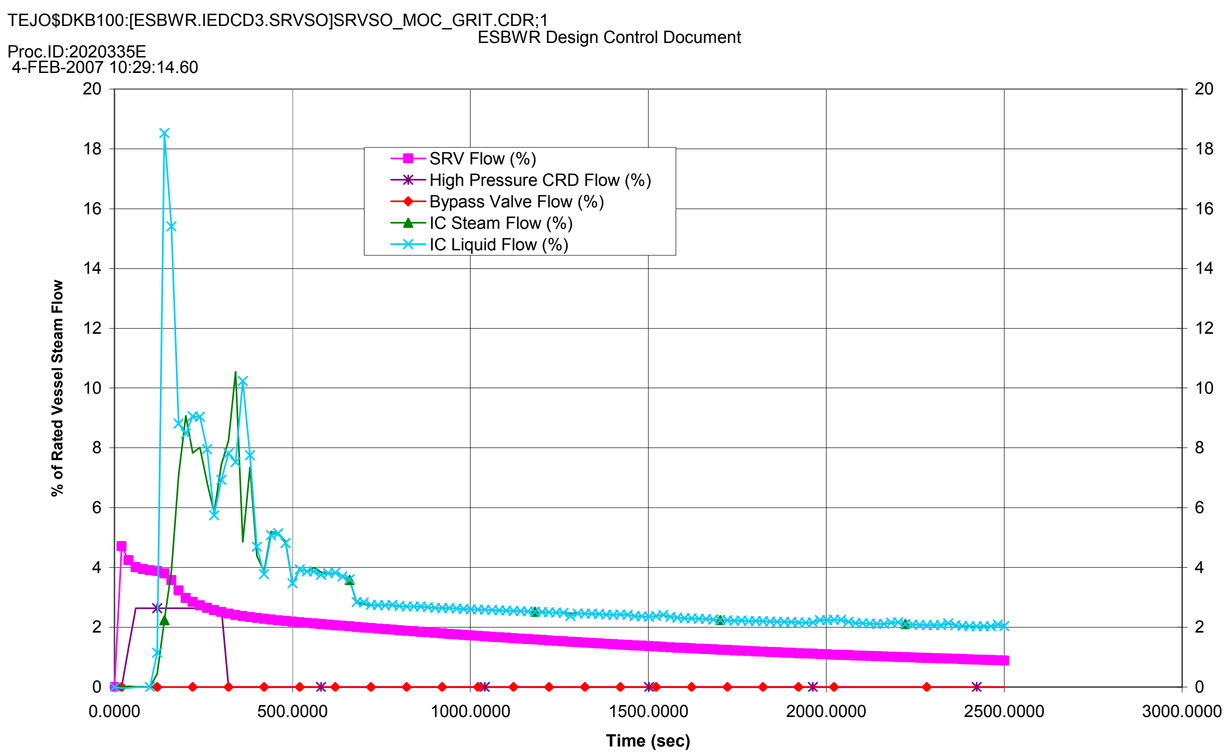


Figure 15.3-9b. Stuck Open Safety Relief Valve

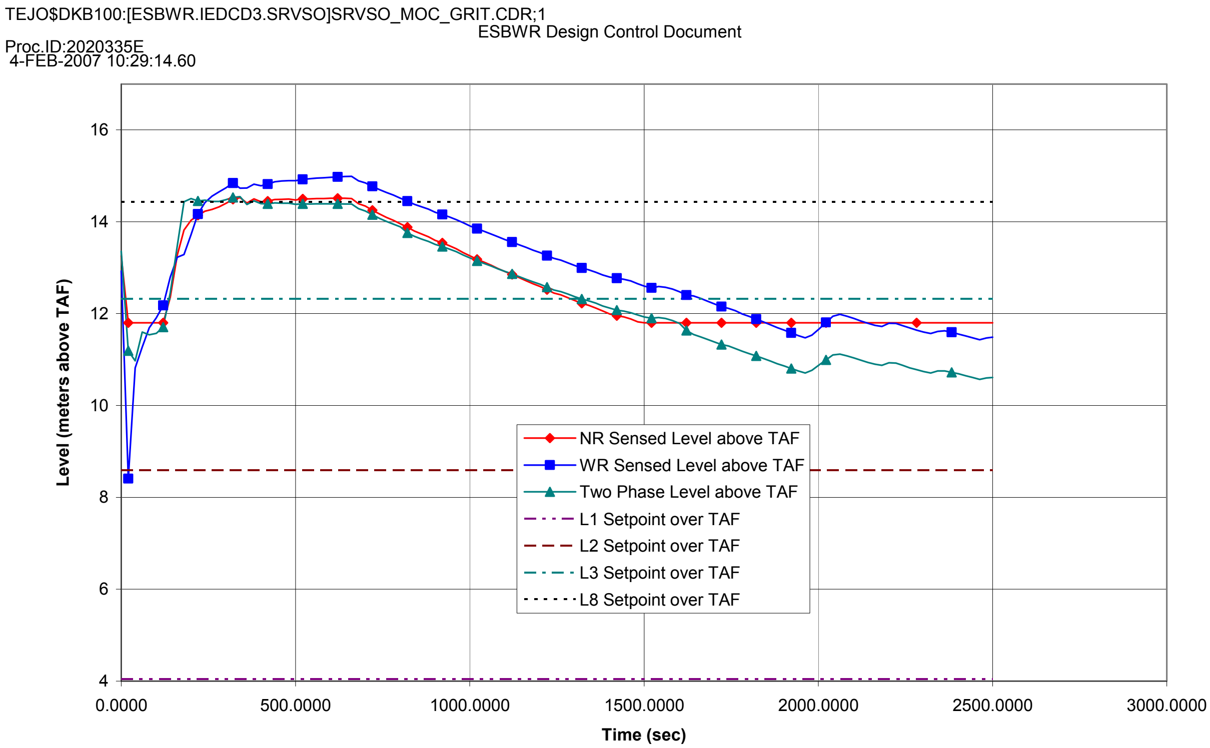


Figure 15.3-9c. Stuck Open Safety Relief Valve

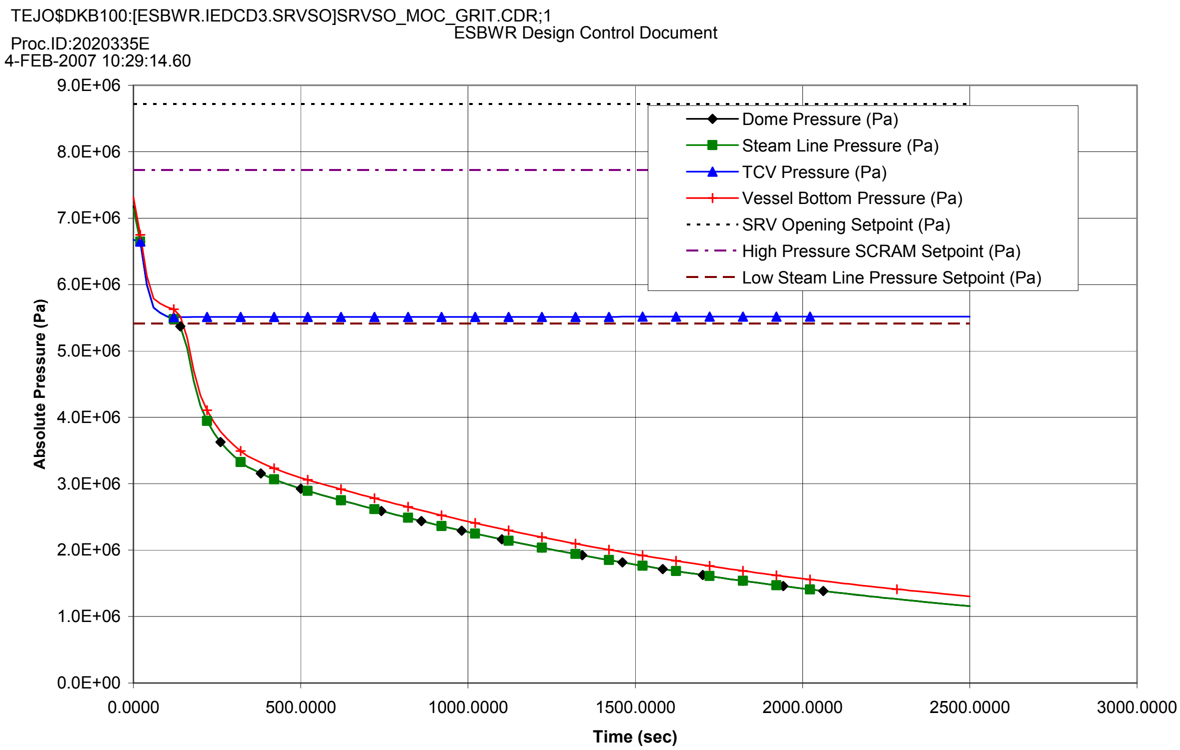


Figure 15.3-9d. Stuck Open Safety Relief Valve

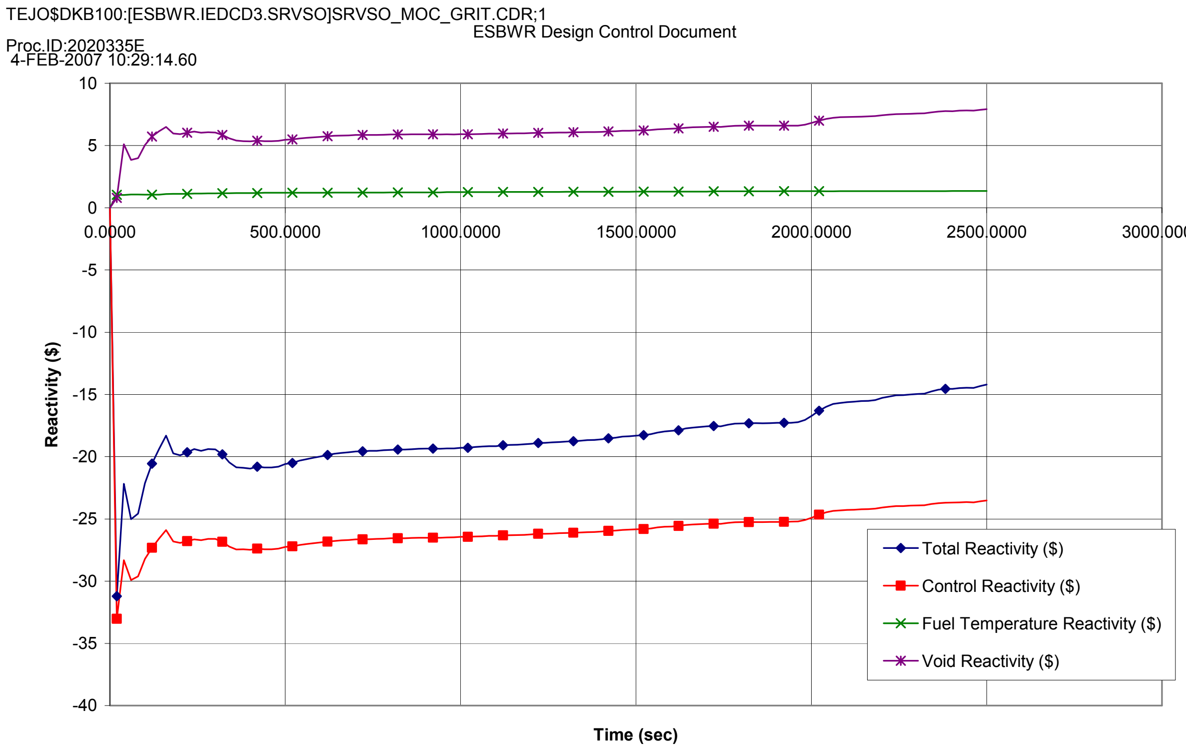


Figure 15.3-9e. Stuck Open Safety Relief Valve

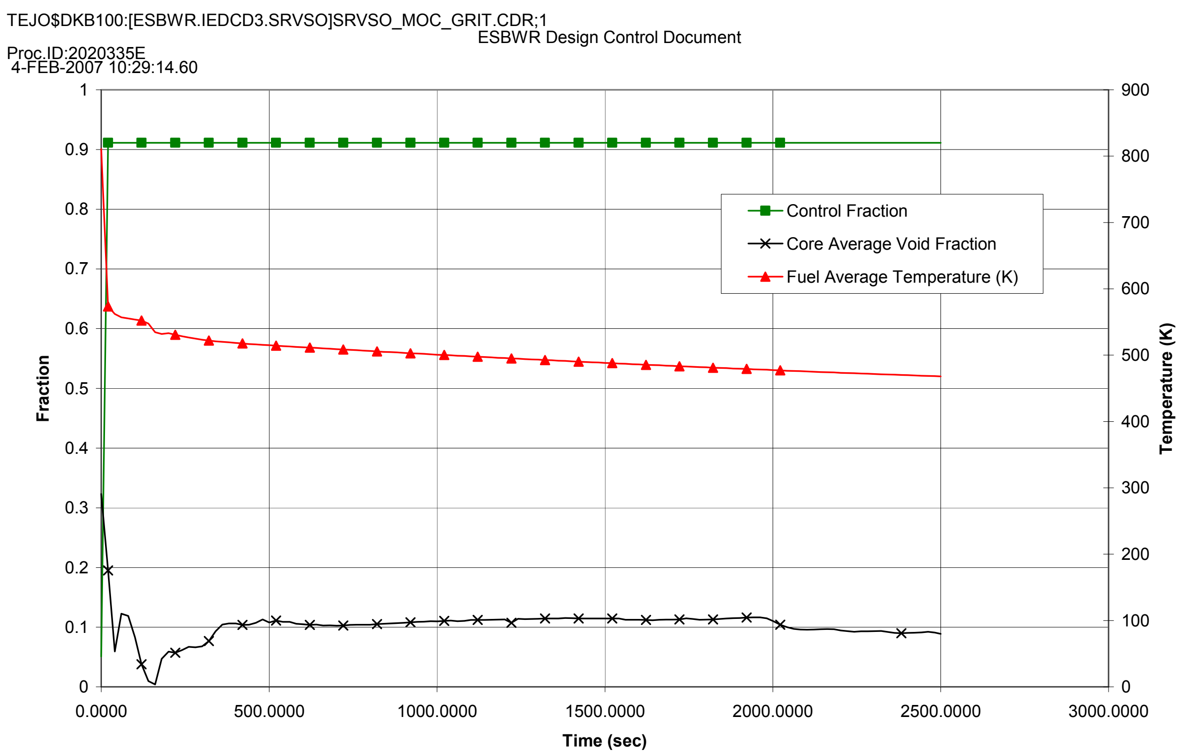


Figure 15.3-9f. Stuck Open Safety Relief Valve

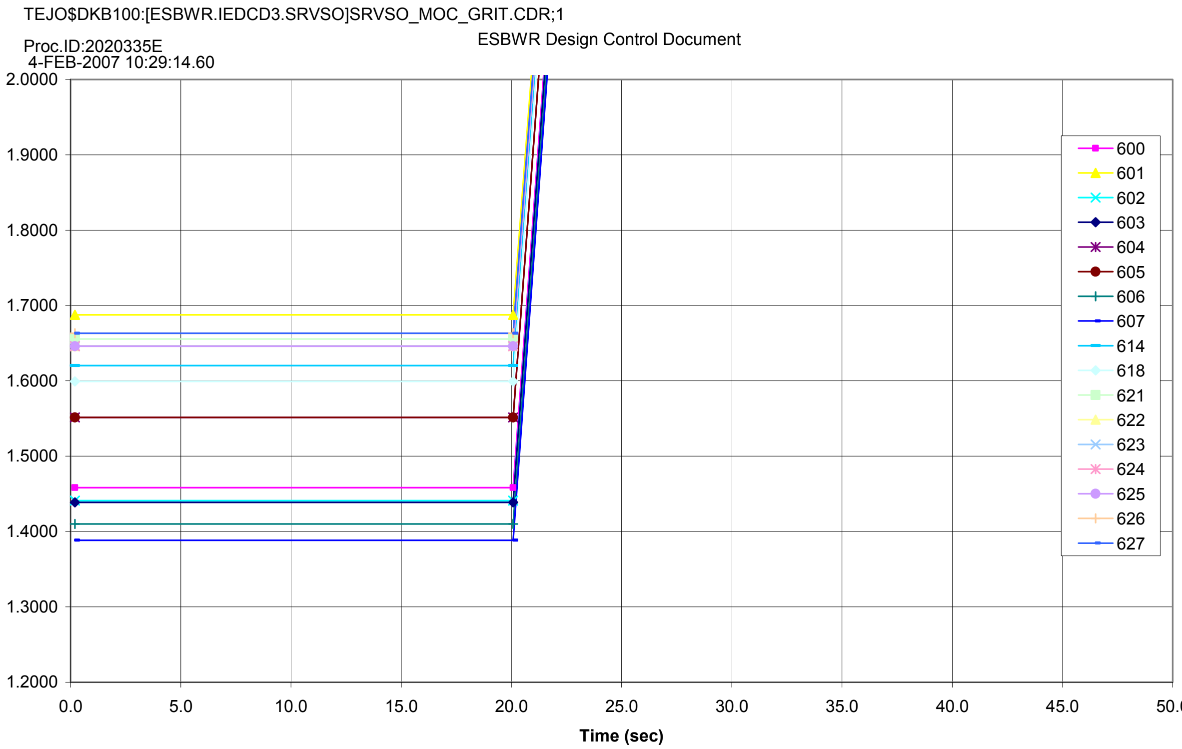


Figure 15.3-9g. Stuck Open Safety Relief Valve

15.4 ANALYSIS OF ACCIDENTS

15.4.1 Fuel Handling Accident

15.4.1.1 Identification of Causes

The fuel-handling accident is assumed to occur as a result of a failure of the fuel assembly lifting mechanism, resulting in dropping a raised fuel assembly onto the reactor core or into the spent fuel storage pool.

15.4.1.2 Sequence of Events and Systems Operation

Sequence of Events

The sequence of events is provided in Table 15.4-1.

Identification of Operator Actions

The following actions are carried out:

- Initiate the evacuation of the Reactor Building or Fuel Building fuel handling area and the locking of the fuel building doors;
- The fuel-handling foreman gives instructions to go immediately to the radiation protection decontamination area;
- The fuel-handling foreman makes the operations shift engineer aware of the accident;
- The shift engineer determines if the normal ventilation system has isolated;
- The shift engineer initiates action to determine the extent of potential radiation doses by measuring the radiation levels in the vicinity of or close to the Reactor Building or Fuel Building;
- The duty shift engineer posts the appropriate radiological control signs at the entrance of the Reactor Building or Fuel Building; and
- Before entry to the fuel handling area is made, a careful study of conditions, radiation levels, etc., is performed.

15.4.1.2.1 System Operation

Normally operating plant instrumentation and controls are assumed to function. No credit is taken for the control room charcoal filter trains or the integrity of the Reactor Building or the Fuel Handling Building. Control room ventilation is assumed to operate in normal operation mode for the duration of the event. Operation of other plant reactor protection or engineered safety feature (ESF) systems is not expected.

15.4.1.3 Core and System Performance

15.4.1.3.1 Mathematical Model

The analytical methods and associated assumptions used to evaluate the radiological consequences of this accident are based on NUREG-1465 alternative source terms (AST) and the methodology in Regulatory Guide (RG) 1.183, to demonstrate compliance with the 10 CFR 50.34(a)(1), SRP 15.0.1 and RG 1.183 total effective dose equivalent (TEDE) acceptance criteria.

15.4.1.3.2 Input Parameters and Initial Conditions

Regulatory Guide (RG) 1.183 provides assumptions acceptable to the NRC that may be used in evaluating the radiological consequences of a postulated fuel-handling accident resulting in damage to the fuel cladding and subsequent release of radioactive materials.

15.4.1.3.3 Number of Failed Fuel Rods

The bounding event with respect to the number of fuel rods damaged occurs in the Reactor Building. Failure of the fuel rod is assumed at 1% circumferential strain. The associated axial strain is (.01)/ υ, where υ, Poisson's ratio, is 0.5 for plastic deformation, and thus the energy per rod failure is

$$Ef = \sigma y \times \varepsilon \times Vol$$

The kinetic energy of the dropped fuel bundle accounts for the effects of buoyancy and the resistance of water. Finite Element Analysis (FEA) simulations determined that when the drop distance of a fuel bundle is greater than 2.3 m (7.5 ft), the kinetic energy of the bundle is less than 50% in water than in air. When the bundle reaches a drop height of 10.36 m (34 ft), the energy is only ~22% of that in air.

The fuel assembly wet weight is assumed to be 215 kg (474 lbs), and the mast wet weight is 195 kg (430 lbs). For conservatism in the analysis for an ESBWR (a drop height of 23.038 m [75.6 ft]), a factor of 2 reduction is applied to obtain the available energy in a fuel assembly drop through water. Therefore the kinetic energy as a result of the drop is

$$KE = (215kg + 195kg) \times (23.038m) \times 50\% = 4722.8kg - m\ .$$

Half of the energy is assumed to be absorbed by the impacted assemblies. The ratio of the cladding to the non-fuel mass is 0.485. The calculated yield strength using the methodology described above is 35.515 kg-m/rod (256.88 ft-lb/rod). Therefore the number of failed rods from the initial drop is calculated as follows

$$\frac{(50\%)(4722.8)(0.485)}{35.515\,{}^{kg-m}\!/_{rod}} = 32.25rods \Rightarrow 33rods$$

The fuel bundle is assumed to have a height of 3.6 m (141.7 in). One again accounting for a 50% reduction in water

$$KE_2 = 50\% \times \left[h_{fuel} W_{mast} + \tfrac{1}{2} h_{fuel} W_{fuel} \right]$$

$$KE_2 = 0.5\left[(3.6m)(195) + \tfrac{1}{2}(3.6m)(215lb) \right] = 544.5kg - m$$

Once again 50% is assumed to be absorbed by the impacted assemblies, therefore the number of failed rods from the secondary impact is

$$\frac{(50\%)(544.5kg - m)(0.485)}{35.515\,{}^{kg-m}\!/_{rod}} = 3.7rods \Rightarrow 4rods$$

All of the 92 rods in the dropped assembly are assumed to fail, therefore the total number of rods (and bundles) failed are

$$92rods + 33rods + 4rods = 129rods$$

$$\left(\frac{129rods}{92\, {}^{rods}\!/_{bundle}} \right) = 1.4bundles \Rightarrow 2.0bundles$$

15.4.1.4 Radiological Consequences

Radiological analysis is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10 CFR 50.34 and 10 CFR 50, Appendix A, General Design Criterion 19 guidelines.

The fission product inventory in the fuel rods that are assumed to be damaged is based on the days of continuous operation at full power. Due to plant cool down and disassembly operations, there is a time delay following initiation of reactor shutdown before fuel movement operations can be initiated. The analysis is based on Regulatory Guide 1.183. Specific values or parameters used in the evaluation are presented in Table 15.4-2.

15.4.1.4.1 Fission Product Transport to the Environment

Emergency procedures require that under FHA conditions the HVAC system be shut down and the fuel-handling area of the Reactor Building or Fuel Building isolated. Following isolation, the operator determines the extent of contamination and time for resuming operation of the HVAC. Gases are released to the environment over a 2-hour period in accordance with Regulatory Guide 1.183 guidance. The flow rate assumed in the dose consequence analysis exceeds the design flow rate for the Fuel Building ventilation system and the Reactor Building ventilation refueling floor subsystem (REPAVS). The Control Room ventilation is assumed to operate in normal mode. No credit is taken for Control Room emergency filter unit (EFU) mitigation nor is the Reactor Building or Fuel Handling Building integrity assumed. The total activity released to the environment is presented in Table 15.4-3a.

15.4.1.4.2 Assumptions Requiring Confirmation

Site parameter assumptions in the radiological analysis are confirmed per Subsection 2.0.1.

15.4.1.5 Results

Calculations are performed for releases from both the Reactor Building and the Fuel Building. The results indicate that the Fuel Building release point is bounding due to the higher atmospheric dispersion factor. The results of this analysis are presented in Table 15.4-4 for both offsite and control room dose evaluations and are within 10 CFR 50.34(a)(1) and 10 CFR 50, Appendix A, GDC 19 limits, and RG 1.183 regulatory guidelines.

15.4.2 Loss-of-Coolant Accident Containment Analysis

The containment performance analysis is provided within Section 6.2, and demonstrates that containment systems meet their design limits for all postulated design basis events.

15.4.3 Loss-of-Coolant Accident ECCS Performance Analysis

The emergency core cooling system (ECCS) performance analysis evaluates the full spectrum of pipe breaks, including the worst case of piping break inside containment. This analysis is provided within Section 6.3, and demonstrates compliance with the 10 CFR 50.46 ECCS acceptance criteria.

15.4.4 Loss-of-Coolant Accident Inside Containment Radiological Analysis

This event assumes a worst case of piping break inside containment. This event is in part based on the fact that the ECCS performance analysis demonstrates to what level that the 10 CFR 50.46 ECCS acceptance criteria are met, and that the containment analysis demonstrates that containment systems meet their design limits.

The postulated event represents the envelope evaluation for liquid or steam line failures inside containment.

The following analysis is based on NUREG-1465 alternative source terms (AST) and the methodology in Regulatory Guide (RG) 1.183, and demonstrates compliance with the 10 CFR 50.34(a)(1), SRP 15.0.1 and RG 1.183 total effective dose equivalent (TEDE) acceptance criteria.

15.4.4.1 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 a LOCA coincident with a Safe Shutdown Earthquake (SSE). The subject piping is of high quality, designed to nuclear construction industry codes and standards, and for seismic and environmental conditions. However, because such an accident provides an upper limit estimate for the resultant effects for this category of pipe breaks, it is evaluated without the causes being identified.

15.4.4.2 Sequence of Events and Systems Operation

15.4.4.2.1 Sequence of Events

The sequence of events associated with this accident is presented in Section 6.3 for ECCS performance and Section 6.2 for barrier (containment) performance.

Following the pipe break and scram, the MSIVs close on the reactor water low level trip signal (Level 2). Some moments later, the reactor low water (Level 1) signal initiates the ADS and GDCS. The core remains covered throughout the accident and there is no fuel damage.

15.4.4.2.2 Identification of Operator Actions

Because automatic actuation and operation of the ECCS is a system design basis, no operator actions are required. However, the operator should perform the following:

- Verify that all rods have inserted;
- Monitor reactor water level and pressure;
- Verify ADS actuation on Level 1;

- Verify GDCS flow at low vessel pressure by observing check valve at open position and GDCS pool level decreasing; and
- Periodically monitor the oxygen concentration in the drywell and wetwell.

15.4.4.2.3 Systems Operations

For all design basis LOCA events described within Section 6.3, there is no core uncovery or heatup. Therefore, no fission products other than spiking terms associated with rapid depressurization occur. Multiple failures of safety-related systems are required to cause significant core damage. Nevertheless, a fission product release is assumed without regard to mechanistic causes to evaluate the ability of the design to mitigate potential fission product releases to the containment.

Accidents that could result in the release of radioactive fission products directly into the containment are the results of postulated reactor coolant pressure boundary (RCPB) pipe breaks. All pipe breaks, sizes and locations are presented in Sections 6.2 and 6.3, including the severance of main steam lines, emergency core cooling system lines, feedwater lines and other process system lines. The minimum required functions of any reactor and plant protection system are presented in Sections 6.2, 6.3, 7.3, 7.6 and 8.3.

15.4.4.3 Core and System Performance

15.4.4.3.1 Mathematical Model

The analytical methods and associated assumptions that are used in evaluating the consequences of this accident are considered to provide a conservative assessment of the consequences of this improbable event. The details of these calculations, their justification and bases for the models are developed to comply with RG 1.183.

15.4.4.3.2 Input Parameters and Initial Condition

Input parameters and initial conditions used for the analysis of this event are presented in Section 6.3.

15.4.4.3.3 Results

Results of this event are presented in detail within Section 6.3. The temperature and pressure transients resulting from this accident are insufficient to cause perforation of the fuel cladding. Therefore, no fuel damage results from this accident. Post-accident tracking instrumentation and control is assured. Continued long-term core cooling is demonstrated. Radiological effects are minimized and within limits. Continued operator control and surveillance is examined and provided.

15.4.4.4 Barrier Performance

The structural design basis for the containment is to maintain its integrity and experience normal stresses after the instantaneous rupture of any primary system piping within the structure, while also accommodating the dynamic effects of the pipe break and an SSE. Therefore, any postulated LOCA does not result in exceeding the containment design limit (see Subsection 3.8.2.3, Section 3.6 and Section 6.2 for details and results of the analyses).

15.4.4.5 Radiological Consequences

The evaluation of the radiological consequences of a design basis loss-of-coolant accident (LOCA), for both the offsite dose evaluations and the control room dose evaluations utilizes the Alternate Source Term (AST) dose methodology. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," was developed to provide guidance for the use of AST. Generally, this analysis follows the methodology prescribed by RG 1.183 and any exceptions are discussed. The NRC developed the computer code RADTRAD specifically for use in AST applications. RADTRAD 3.03 (Reference 15.4-7) is used for the analysis. RADTRAD utilizes "compartments" to represent specific volumes of the plant (and the environment). To estimate the dose consequences of a LOCA, a model consisting of these 6 compartments is used:

(1) Drywell

(2) Reactor Building

(3) Passive Containment Cooling System/Isolation Condenser Pools Air Volumes

(4) Main Condenser

(5) Control Room

(6) Environment

The analysis is based upon a process flow diagram shown in Figure 15.4-1 and accident parameters specified in Table 15.4-5.

15.4.4.5.1 Source Term Assumptions

15.4.4.5.1.1 Chemical Release Fractions

RG 1.183, Appendix A, Section 3.1 states that the radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the drywell. It also states that the release into the drywell should be terminated at the end of the Early In-Vessel (EIV) release phase. As such, the AST dose methodology assumes a 2-hour phased release. Three phases are assumed: coolant, gap, and EIV. The coolant release phase is assumed to last 20 minutes based on an ESBWR specific model using the MELCOR computer code (Reference 15.4-12). During the coolant release phase, no fuel damage occurs. The fission gases in the plenum of the fuel rods are assumed to be released during the gap release phase. This gap release phase is assumed to last for 30 minutes, from 20 to 50 minutes following the break. The final release phase for DBA considerations is the EIV phase. During this phase the core is assumed to melt, thus core geometry is compromised. This phase is assumed to last for 90 minutes, or from 50 minutes to 2 hours 20 minutes. Thus core damage is assumed to occur over a period of 2 hours, after the initial coolant release phase of 20 minutes.

The release fractions listed in RG 1.183 are divided into eight groups. The release fractions for each group are taken from Table 1 of RG 1.183. The release timing is taken from Table 4 of Reg. Guide 1.183.

15.4.4.5.1.2 Core Inventory

35 GWd/MT burnup fuel was used with no decay time assumed.

15.4.4.5.1.3 Reactor Power

The rated core thermal power of the ESBWR is 4500 MWth. Adding an additional 2% to account for instrument uncertainty (Regulatory Guide 1.49) yields a core thermal power for this analysis of 4590 MWth.

15.4.4.5.1.4 Iodine Chemical Distribution

RG 1.183, Appendix A, Section 2 states that "If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodine (CsI), 4.85% elemental iodine, and 0.15% organic iodine." Based on the application of the systems identified in Subsection 15.4.4.5.2.2 to control pH after 72 hrs, this chemical distribution for pH controlled pools is assumed in the analysis.

15.4.4.5.1.5 Radiation Decay and Daughter Products

The computer code RADTRAD allows users to track radiation decay for the duration of the event. It also has an option to account for the buildup of daughter products. Both options are used in this analysis.

15.4.4.5.2 Radionuclide Releases and Pathways

Two specific pathways are analyzed in releasing radionuclides to the environment: leakage from the primary containment building and leakage through the Main Steam Isolation Valves. Leakage through the MSIVs is not included in the containment leakage summation, as discussed in Subsection 6.2.6.3. The primary containment leakage pathway is assumed to be no greater than an equivalent release of 0.5% volume per day from the containment per plant Technical Specifications**.** The bulk of the primary containment leakage (98%, or 0.49% volume per day) is released into the Reactor Building. Reactor Building leaks to the environment at a rate specified in Table 15.4-5. The remaining 2% of primary containment leakage is assumed to leak though the PCCS into the airspace directly above the PCCS and IC pools. This leakage is quickly vented directly to the atmosphere. The final leakage pathway is MSIV leakage to the Turbine Building condenser. This pathway is discussed separately below.

15.4.4.5.2.1 Removal of Elemental Iodine from Containment

Natural deposition of elemental iodine is credited in the dose consequence analyses. The elemental iodine coefficient is based on guidance found in SRP 6.5.2. Specifically, the iodine removal rate constant for a particular compartment "n" is based on the following formula:

$$\lambda_n = k_g\left(\frac{A}{V}\right)$$

where,

λ_n = removal rate constant due to surface deposition (0.137 cm/sec based on NUREG/CR-0009 (Reference 15.4-11) page 17),

k_g = average mass transfer coefficient,

A = surface area for deposition, and

V = volume of the contained gas.

The surface area credited for deposition is wall surface area of the building and the floor area for elevation 17500, since that elevation represents the largest cross-section area. The resultant area (803.5 m^2) was then conservatively reduced by 50%. Other surfaces, such as the bioshield wall for the drywell are conservatively neglected. The calculated elemental iodine removal rate constant used is provided in Table 15.4-5.

15.4.4.5.2.2 Aerosol Removal from Containment

There are several natural processes which can remove airborne aerosols from the primary containment atmosphere following a LOCA. The PCCS is used to condense steam and control pressure in the event of a LOCA. The PCCS effectively scrubs the containment atmosphere by removing aerosols from the containment atmosphere. Aerosols are also removed via natural deposition onto containment internal structures. The removal mechanisms for the PCCS and natural deposition of airborne aerosols are similar, therefore one integral model is used. The removal coefficients are based on the results of the ESBWR MELCOR model as discussed in NEDE-33279P, *ESBWR Containment Fission Product Removal Evaluation Model* (Reference 15.4-13).

Aerosols released from the RPV will be airborne in the containment. One path for the aerosols to contribute to the off-site dose is via containment leakage from containment atmosphere to the Reactor Building and subsequently to the environment. Aerosols must be airborne in the containment atmosphere to leave via this pathway. Since aerosols are suspended in the containment atmosphere, they will circulate with the bulk gas movement, which is from the RPV to the PCCS. Figure 6.2-16 shows the PCCS heat exchanger and its associated piping. Steam, nitrogen and any airborne fission products will enter the PCCS heat exchanger inlet line from the drywell, which discharges to a header at the top of the PCCS tube bundles. Steam vapor will condense on the header and inside walls of the PCCS heat exchanger tubes, which are cooled on the outside by the water in the PCC/IC pool. The deposition processes of aerosol are gravity, Brownian diffusion, thermophoresis and diffusiopheresis. Aerosol and fission product vapors can deposit directly on surfaces such as heat structures and water pools. In addition, aerosol can agglomerate and settle. The aerosols deposited on the various surfaces can relocate. If a water film drains from a heat structure to the pool in the associated volumes, fission products deposited on that structure are transported with the water. This relocation is proportional to the fraction of the film that is drained. Aerosols and fission product vapors are transported between control volumes by bulk fluid and gas flows. Aerosols may also settle from a volume to a lower volume in the absence of bulk flow. Diffusiophoresis, the phenomenon of aerosol movement in condensing vapor, will drive the aerosol particles to the condensate film on the PCCS tube inner wall. In addition to condensation, some fraction of the airborne activity will also “plate out” in the PCCS. The PCCS effectiveness in removing aerosols has been demonstrated in third party tests. For example, in “Investigation on Aerosol Deposition in a Heat Exchanger Tube” (Reference 15.4-8), a short length of PCC tube was capable of removing a significant portion of in-flowing aerosols, and removing them with the condensate flow.

The condensate from the PCCS will drain into the GDCS pool, and then back into the reactor pressure vessel. The PCCS heat exchanger vents non-condensables, including noble gases to the suppression pool. Aerosols which do not deposit in the PCCS are transported by non-condensable gases via PCCS vent line into the wetwell. The vent mass flow rate is less than onequarter of the PCCS flow rate, and aerosol and iodine transport to the suppression pool would be a small fraction of the total decontamination factor credited to the PCCS.

Since the net effect is the removal of activity from the drywell atmosphere, one set of removal coefficients are applied to model the effect on the airborne activity. Because non-condensables will leave the suppression pool for the WW airspace, and flow back into the DW during vacuum breaker openings, and because the PCCS removal mechanisms are not effective for organic iodines, no credit for noble gas or organic iodine decontamination is taken in the analysis (the decontamination factor is 1).

The IC/PCCS pools cooling and cleaning subsystem is completely independent and separated from FAPCS cooling and cleaning subsystem that serves other pools (FPC, GDCS, suppression pools, and upper pools). There is no physical (pipe) connection between the FAPCS cooling and cleaning subsystem and the IC/PCCS pools cooling and cleaning subsystem, which is supported by the discussion in Subsection 9.1.3. There is no concern of contamination of the PCC/IC pool when injecting into GDCS, RPV, and suppression pools post-accident.

The various containment pools were evaluated to determine the pH levels following a LOCA with a release of fission products. Credit is taken buffering as a result of the sodium pentaborate injected into the RPV via the SLC system. The formation of CsOH also assists in buffering pool pH levels. Early in the event pool pH increases to 8 to 10 for the various pools, primarily as a result of CsOH. Acids produced as a result radiation, such as HCl produced as result of electrical cables, lower the pH levels in the various pools. Calculations confirm that the GDCS pool could become acidic roughly 12 hours following the event, however at that point the pool is essentially depleted and contains minimal fission products. Therefore, re-evolution of iodine is not of concern do to the extremely low overall activity level in that pool. The remaining pools' pH remains above 7 for at least the first 24 hours.

15.4.4.5.2.3 Main Steamline Modeling

The second potential release pathway is via the main steamline through leakage in the main steamline isolation valves. It is assumed that a pathway exists which permits the containment atmosphere, or in the non-break case, pressure vessel air space direct access to the main steamlines. The main steamline isolation valves are assumed to leak at the Technical Specification limit. Furthermore, it is assumed that the most critical main steamline isolation valve fails in the open position. Therefore, the total leakage through the steamlines is equal to the Technical Specification limit for the plant.

The main steamlines are classified (see Table 3.2-1) as Seismic Category I from the pressure vessel interface to the outboard seismic restraint outboard of the downstream MSIV, thereby providing a qualified safety-related mitigation system for fission product leakage, which, in this case, is limited by the leakage criteria specific in the technical specifications for the MSIVs. The primary purpose of this system is to stop any potential flow through the main steamlines. Downstream of the seismic restraint referred to above, the steamlines pass through the Reactor Building - Turbine Building interface into the Turbine Building steam tunnel. The Turbine Building steam tunnel is a heavily shielded reinforced concrete structure designed to shield workers from main steamline radiation shine. The steamlines and their associated branch lines outboard of the last Reactor Building seismic restraint are Quality Group B structures. In addition, these lines and structures are required to be dynamically analyzed to SSE conditions

(Table 3.2-1) that determine the flexibility and structural capabilities of the lines under hypothetical SSE conditions.

The analysis of leakage from the containment through the main steamlines involves the determination of

- Probable and alternate flow pathways,
- Physical conditions in the pathways, and
- Physical phenomena that affect the flow and concentration of radionuclides in the pathways.

The most probable pathway for radionuclide transport from the main steamlines is found to be from the outboard MSIVs into the drain lines coming off the outboard MSIV and then into the Turbine Building to the main condenser. A secondary path is found along the main steamlines into the turbine though flow through this pathway as described below is a minor fraction of the flow through the drain lines.

Consideration of the main steamlines and drain line complex downstream of the Reactor Building as a mitigative factor in the analysis of LOCA leakage is based upon the following determination:

- The main steamlines and drain lines are high quality lines inspected on a regular schedule.
- The main steamlines and drain lines are designed to meet SSE criteria and analyzed to dynamic loading criteria.
- The main steamlines and drain lines are enclosed in a shielded corridor that protects them from collateral damage in the event of an SSE. For those portions not enclosed in the steam tunnel complex, an as-built inspection is required to verify that no damage could be expected from other components and structures in a SSE.
- The main steamlines and drain lines are required under normal conditions to function to loads at temperature and pressure far exceeding the loads expected from an SSE. This capability inherent in the basic design of these components furnishes a level of toughness and flexibility to ensure their survival under SSE conditions. A large database of experience in the survival of these types of components under actual earthquake conditions proves this contention (Reference 15.4-4). In the case of the ESBWR, further margin for survival can be expected, because the ESBWR lines are designed through dynamic analysis to survive such events, whereas in the case of the actual experience database, the lines shown to survive were designed to lesser standards to meet only normally expected loads.

Based upon the facts above, the main steamlines and drain lines are credited in that they direct potential leakage through the MSIVs to the main condenser. No credit is taken for plateout of fission products in either the main steam lines or main steam line drain lines.

15.4.4.5.2.4 Condenser Modeling

The condenser is modeled as detailed in Reference 15.4-4 with specific values used given in Table 15.4-5. The volume is modeled primarily as a stagnant volume assuming the shutdown of all active components. The condenser is used as a mitigative volume based upon the determination that such components, designed to standard engineering practice, are sufficiently strong to withstand SSE conditions (Reference 15.4-4).

Releases from the condenser/Turbine Building pathway are assumed via diffuse sources in the Turbine Building. The two major points of release in the Turbine Building are expected to be the truck doors at the far end of the Turbine Building and the Turbine Building vent panels located midway on the Turbine Building on the side away from the Reactor Building. Releases are assumed to be ground level releases.

15.4.4.5.3 Control Room

The control room is physically integrated into the Control Building and is located below grade adjacent to the Reactor, Service, and Turbine Buildings. During a LOCA, exposure to the operators consists of contributions from airborne fission products entrained into the control room ventilation system.

Exposure to the operators from airborne contamination consists almost entirely of radionuclides entrained into the control room environment via the HVAC from the atmosphere. The control room is designed to operate under minimal power and HVAC conditions with air flow into the control room and positive pressure maintained by battery powered safety-related charcoal filter trains. The system is initiated as a result of a high radioactivity signal in the normal intake, or as a result of a loss of power. The transit time for the normal air intake is designed such that isolation of the control room will occur prior to radioactivity reaching the isolation dampers. The system can maintain the control room under positive pressure minimal in leakage conditions for 72 hourswith the most limiting single active failure. After 72 hours, recharging of batteries can be performed via either the main plant diesel generators or dedicated diesel generators (via an external connection). The control room habitability system is described in Subsection 6.4.3.

Control room dose is based upon fission product releases modeled as described above and the values presented in Table 15.4-8. Operator exposure is also based upon those conditions given in Table 15.4-8. The occupancy factors presented in Table 15.4-5 are derived from RG 1.183.

15.4.4.5.4 Meteorology and Site Assumptions

Offsite Meteorology - This DCD uses a generic U.S. site that does not specifically identify meteorological parameters adequate to define dispersion conditions for accident evaluation. Therefore, a set of dispersion parameters (χ/Q's) were selected to simulate a U.S. site, which are given in Table 15.4-9 for the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ).

Control Room Meteorology - No specific acceptable method exists to calculate the meteorology for standard plant application for control room dose analysis. The control room assumed dispersion factors (χ/Q) are provided in Table 15.4-9.

15.4.4.5.5 Breathing Rates

The breathing rates assumed in the analysis presented in Table 15.4-5. These values are consistent with RG 1.183, Section 4.1.3.

15.4.4.6 Results

The results of this analysis are presented in Table 15.4-9 for both offsite and control room dose evaluations and are within 10 CFR 50.34 and RG 1.183 regulatory guidelines. The following criteria are met:

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.

(3) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

15.4.4.6.1 Assumptions Requiring Confirmation

Site parameter assumptions in the radiological analysis are confirmed per Subsection 2.0.1.

15.4.4.6.2 (Deleted)

15.4.5 Main Steamline Break Accident Outside Containment

This event involves postulating a large steam line pipe break outside containment. It is assumed that the largest steam line instantaneously and circumferentially breaks at a location downstream of the outermost isolation valve. The plant is designed to immediately detect such an occurrence, initiate isolation of all main steamlines including the broken line and actuate the necessary protective features. This postulated event represents the envelope evaluation of steam line failures outside containment.

The Main Steamline Break Accident (MSLBA) containment response evaluation is provided in Section 6.2.

The MSLB ECCS capability evaluation is provided in Section 6.3.

The MSLB radiological evaluation is as follows:

15.4.5.1 Identification of Causes

A MSLBA is postulated without the cause being identified. These lines are designed to high quality engineering codes and standards, and to 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.

15.4.5.2 Sequence of Events and Systems Operation

15.4.5.2.1 Sequence of Events

Accidents that result in the release of radioactive materials directly outside the containment are the result of postulated breaches in the reactor coolant pressure boundary or the steam power conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting event for breaks outside the containment is a complete severance of one of the main steamlines. The sequence of events and approximate time required to reach the event is given in Table 15.4-10.

Following isolation of the main steam supply system (i.e., MSIV closure), the ADS initiates automatically on low water level (Level 1). Once the reactor system has depressurized, the GDCS automatically begins reflooding the reactor vessel. The core remains covered throughout the accident, and there is no fuel damage.

15.4.5.2.2 Systems Operation

A postulated guillotine break of one of the 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 within the reactor vessel steam outlet nozzle. Flow from the downstream side is initially limited by the flow restrictor within the reactor vessel steam outlet nozzle for the three unbroken lines. Subsequent closure of the MSIVs further limits the flow when the valve area becomes less than the limiter area and finally terminates the mass loss when the full closure is reached.

A discussion of plant and RPS action and ESF action is presented in Sections 6.3, 7.3 and 7.6.

15.4.5.2.3 The Effect of Single Failures and Operator Errors

The steamline break outside the containment is a special case of the general LOCA break spectrum considered in detail in Section 6.3. The general single-failure analysis for LOCAs is presented in Subsection 6.3.3. For the steamline break outside the containment, the worst single failure does not result in core uncovery (see Section 6.3 for analysis details).

15.4.5.3 Core and System Performance

Quantitative results (including mathematical models, input parameters, and consideration of uncertainties) for this event are presented in Section 6.3. The temperature and pressure transient results from this accident are not sufficient to cause fuel damage.

15.4.5.3.1 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the ECCS performance analysis of this event are presented in Table 6.3-1.

15.4.5.3.2 Results

There is no fuel damage as a result of this accident. Refer to Section 6.3 for ECCS analysis.

15.4.5.4 Barrier Performance

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

Initially, only steam issues from the broken end of the steam line. The flow in each line is limited by critical flow at the limiter 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. The total integrated mass leaving the RPV through the steamline break is provided in Table 15.4-11.

15.4.5.5 Radiological Consequences

The radiological analysis for this accident is based on conservative assumptions considered to be acceptable to the NRC for the purposes of determining adequacy of the plant design to meet 10 CFR 50.34(a)(1) guidelines. This analysis is referred to as the "design basis analysis."

15.4.5.5.1 Design Basis Analysis

Specific values of parameters used in the evaluation are presented in Table 15.4-11.

General Compliance or Alternate Approach Statement (RG 1.183): This guide provides assumptions acceptable to the NRC that may be utilized in evaluating the radiological consequences of a MSLBA for a BWR.

Some of the models and conditions that are prescribed are inconsistent with actual physical phenomena. The effect of the conservative bias that is introduced is generally limited to plant design choices not within the scope of the ESBWR Standard Plant design. The resultant dose is within regulatory limits.

Fission Product Release from Fuel: 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 steamlines prior to the break.

Fission Product Transport to the Environment: The transport pathway is a direct unfiltered release to the environment with an air exchange rate in from the Turbine Building of 1.0E+08 volume % per day. Assuming that all of the activity in the steam becomes airborne, the release of activity to the environment is presented in Table 15.4-12.

Assumptions Requiring Confirmation

Site parameter assumptions in the radiological analysis are confirmed per Subsection 2.0.1.

15.4.5.5.2 Results

The calculated exposures for the design basis analysis are presented in Table 15.4-13 and are less than the guidelines of RG 1.183 and 10 CFR 50.34(a)(1).

15.4.6 Control Rod Drop Accident

15.4.6.1 Features of the ESBWR Fine Motion Control Rod Drives

As presented in Subsection 4.6.1, the Fine Motion Control Rod Drive (FMCRD) has several new features that are unique compared with locking piston control rod drives.

In each FMCRD, there are dual safety-related separation-detection devices that detect the separation of the control rod from the FMCRD if the control rod is stuck and separated from the ballnut of the FMCRD. The control rods are normally inserted into the core and withdrawn with the hollow piston, which is connected with the control rod, resting on the ballnut. The separation-detection device is used at all times to ascertain that the hollow piston and control rod are resting on the ballnut of the FMCRD. The separation-detection devices sense motion of a spring loaded support for the ball screw and in turn the hollow piston and the control rod. Separation of either the control rod from the hollow piston or the hollow piston from the ballnut is detected immediately. When separation has been detected, the interlocks preventing rod withdrawal operate to prevent further control rod withdrawal. Also, an alarm signal would be initiated in the control room to warn the operator.

There is also the unique highly reliable bayonet type coupling between the control rod blade and the FMCRD. With this coupling, the connection between the blade and the drive cannot be separated unless they are rotated 45 degrees. This rotation is not possible during reactor operation. There are procedural coupling checks to assure proper coupling. Finally, there is the latch mechanism on the hollow piston part of the drive. If the hollow piston is separated from the ballnut and rest of the drive due to stuck rod, the latch limits any subsequent rod drop to a short distance. More detailed descriptions of the FMCRD system are presented in Subsection 4.6.1. Failure modes of the FMCRD are discussed in Appendix 15A.

15.4.6.2 Identification of Causes

For the rod drop accident with a potentially adverse result to occur, it is necessary for the following highly unlikely events to occur:

(1) The reactor is at < 5% power;

(2) There are failures of both safety-related separation-detection devices or a failure of the rod block interlock;

(3) There is a failure of the latch mechanism to occur;

(4) Simultaneously, there is a additional failure that causes the occurrence of a stuck rod on the same FMCRD;

(5) The control rod drive is withdrawn without the operators noticing that the control rod withdrawal did not result in a neutron flux increase; and

(6) Then the stuck rod has to become unstuck.

Alternatively, separation of the blade from the hollow piston would require either that control rod was installed without coupling and the coupling checks failed, or there is structural failure of this coupling. Under such circumstances of this coupling failure, the rod drop accident can only occur with the simultaneous failure of both separation-detection devices (or the failure of the rod block interlock), together with the occurrence of a stuck rod on the same FMCRD.

In either case, because of the low probability of such simultaneous occurrence of these multiple independent events, there is no basis to postulate this event to occur.

15.4.6.3 Sequence of Events and System Operation

15.4.6.3.1 Sequence of Events

The bayonet coupling and procedural coupling checks preclude the uncoupling of the control rod from the hollow piston of the FMCRD. If the control rod is stuck, the separation-detection devices would detect the separation of the control rod and hollow piston from the ballnut of the FMCRD and rod block interlock would prevent further rod withdrawal. The operator would be alarmed for this separation.

Therefore, there is no technical basis for the control rod drop event to occur.

15.4.6.3.2 Identification of Operator Actions

No operator actions are required to preclude this event. However, the operator would be notified by the separation-detection alarm if separation is detected.

15.4.6.4 Core and System Performance

The performance of the separation-detection devices and the rod block interlocks virtually preclude the cause of a rod drop accident.

15.4.6.5 Barrier Performance

An evaluation of the barrier performance is not made for this accident, because there is no circumstance for which this event could occur.

15.4.6.6 Radiological Consequences

The radiological analysis is not required.

15.4.7 Feedwater Line Break Outside Containment

The feedwater line break containment response evaluation is provided in Section 6.2.

The feedwater line break ECCS capability evaluation is provided in Section 6.3.

The feedwater line break radiological evaluation is as follows:

The postulated break of the feedwater line, representing the largest liquid line outside the containment, provides the envelope evaluation for this type of break. The break is assumed to be instantaneous, circumferential and downstream of the outermost isolation valve.

A more limiting event from a core performance evaluation standpoint (Feedwater Line Break Inside Containment) has been quantitatively analyzed and is presented in Section 6.3. Therefore, the following discussion provides only new information not presented in Section 6.3. All other information is cross-referenced to appropriate Chapter 6 subsections.

15.4.7.1 Identification of Causes

A feedwater line break is assumed without the cause being identified. The subject piping is designed to high quality, to engineering codes and standards, and to seismic environmental requirements.

15.4.7.2 Sequence of Events and System Operation

15.4.7.2.1 Identification of Operator Actions

Because automatic actuation and operation of the ECCS is a system design basis, no operator action is required.

However, the operator should perform the following (shown for informational purposes only):

- Determine that a line break has occurred and evacuate the area of the Turbine Building.
- Ensure that the reactor is shut down and that the ICS and the CRD systems are operating normally or, if failed, that the ADS and GDCS are operating.
- Implement site radiation incident procedures.
- Shut down the feedwater system and de-energize any electrical equipment that may be damaged by water from the feedwater system in the Turbine Building.
- Continue to monitor reactor water level and the performance of the ECCS while the radiation incident procedure is being implemented and begin normal reactor cooldown measures.
- Initiate the FAPCS in the suppression pool cooling mode (if necessary) and RWCU/SDC in the shutdown cooling mode.

These actions occur over an elapsed time of 3-4 hours.

15.4.7.2.2 Systems Operation

It is assumed that the normally operating plant instrument and controls are functioning. Credit is taken for the actuation of the ECCS. The Reactor Protection System, SRVs, ECCS, and Control Rod Drive system are assumed to function properly to ensure a safe shutdown.

The ESF systems, including the ADS and GDCS, are assumed to operate normally.

15.4.7.2.3 The Effect of Single Failures and Operator Errors

The feedwater line outside the containment is a special case of the general LOCA break spectrum presented in detail within Section 6.3. The general single-failure analysis for LOCAs is presented in detail in Subsection 6.3.3. For the feedwater line break outside the containment, the worst single failure does not result in core uncovery, and there would be no fuel damage.

15.4.7.3 Core and System Performance

15.4.7.3.1 Qualitative Summary

The accident evaluation qualitatively considered in this subsection is considered to be a conservative and envelope assessment of the consequences of the postulated failure (i.e., severance) of one the feedwater piping lines external to the containment.

15.4.7.3.2 Qualitative Results

The feedwater line break outside containment is less limiting, from a core performance evaluation standpoint, than the main steamline break outside the containment analysis presented in Subsection 15.4.5 and the LOCA inside the containment analysis presented in and 15.4.4.

The break is isolated by closure of the feedwater check valves. The main steamlines are isolated on water level 2, and the ADS and the GDCS together restore the reactor water level to the normal elevation. The fuel is covered throughout the transient and there is no pressure or temperature transient sufficient to cause fuel damage.

15.4.7.3.3 Consideration of Uncertainties

This event is conservatively analyzed and uncertainties were adequately considered (see Section 6.3).

15.4.7.4 Barrier Performance

Accidents that result in the release of radioactive materials outside the containment are the result of postulated breaches in piping connected to the reactor coolant pressure boundary or the steam power-conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the main steamlines as presented in Subsection 15.4.5. The feedwater system piping break is less severe than the main steamline break. Results of analysis of this event can be found in Section 6.3.

15.4.7.5 Radiological Consequences

15.4.7.5.1 Analysis

The NRC provides no specific regulatory guidelines for the evaluation of this accident; therefore, the analysis presented is based upon conservative assumptions considered acceptable to the NRC. However, for consistency, the RG 1.183 guideline exposure acceptance criteria for the MSLBA are used for the Feedwater Line Break accident.

The analysis is based on a conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are presented in Reference 15.4-1. Specific values of parameters used in the evaluation are presented in Table 15.4-14.

15.4.7.5.2 Fission Product Release

There is no fuel damage as a consequence of this accident. In addition, an insignificant quantity of activity (compared to that existing in the main condenser hotwell prior to the occurrence of the break) is released from the contained piping system prior to isolation closure.

The iodine concentration assumed is that of the maximum equilibrium reactor water concentration used for the MSLBA, subject to a 2% carryover of iodine in the water to steam condensate. Noble gas activity in the condensate is negligible, because the air ejectors remove all noble gases from the condenser.

15.4.7.5.3 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.

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 15.4-15.

15.4.7.5.4 Assumptions Requiring Confirmation

Site parameter assumptions in the radiological analysis are confirmed per Subsection 2.0.1.

15.4.7.5.5 Results

The calculated exposures for the analysis are presented in Table 15.4-16, and are less than the regulatory guideline exposures.

15.4.8 Failure of Small Line Carrying Primary Coolant Outside Containment

This event postulates a small steam or liquid line pipe break inside or outside the containment, but within a controlled release structure. To bound the event, it is assumed that a small instrument line, instantaneously and circumferentially, breaks at a location where it may not be able to be isolated and where detection is not automatic or apparent. This event is less limiting from a core performance evaluation standpoint than the postulated events presented in Subsections 15.4.5 (Main Steamline Break Accident Outside Containment), 15.4.4 (Loss-of-Coolant Accident Inside Containment Radiological Analysis), and 15.4.7 (Feedwater Line Break Outside Containment).

This postulated event represents the envelope evaluation for small line failure inside and outside the containment relative to sensitivity for detection.

15.4.8.1 Identification of Causes

There is no identified specific event or circumstance that results in the failure of an instrument line. These lines are designed to high quality, engineering standards, seismic and environmental requirements. They also are equipped with either excess flow check valves or isolation valves. However, for the purpose of evaluating the consequences of a small line rupture, the rupture of an instrument line is assumed to occur along with a failure to isolate the break.

A circumferential rupture of an instrument line that is connected to the primary coolant system is postulated to occur outside the drywell, but inside the Reactor Building. The associated effects from a rupture in the drywell would not be as significant as those from the failure in the Reactor Building.

15.4.8.2 Sequence of Events and Systems Operations

15.4.8.2.1 Sequence of Events

The leak may result in noticeable increases in radiation, temperature, humidity, or audible noise levels in the Reactor Building or abnormal indications of actuations caused by the affected instrument.

Termination of the analyzed event is dependent on operator action. The action is initiated with the discovery of the unisolatable leak. The action consists of the orderly shutdown and depressurization of the reactor.

15.4.8.2.2 Systems Operation

A presentation of plant, RPS, ESF and other safety-related action are given in Sections 6.3, 7.3, and 7.6.

15.4.8.2.3 The Effect of Single Failures and Operator Errors

There is no single failure or operator error that significantly affects the system response to this event. Single failures in other instrument channels could lead to actuation of ESF actions such as reactor scram or MSIV closure under the assumption that the line break trips one division and the single failure trips another division to produce a 2-out-of-4 trip condition.

15.4.8.3 Core and System Performance

Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint than the events examined in Subsections 15.4.4, 15.4.5 and 15.4.7. Consequently, instrument line breaks are considered to be bounded specifically by the MSLBA (Subsection 15.4.5). Details of this calculation, including those pertinent to core and system performance, are presented in Subsection 15.4.5.3.

15.4.8.3.1 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the analysis of this event are presented in Table 15.4-17.

15.4.8.3.2 Results

No fuel damage or core uncovery occurs as a result of this accident. Instrument line breaks are within the spectrum considered in ECCS performance calculations presented in Section 6.3.

15.4.8.4 Barrier Performance

The following assumptions and conditions are the basis for the mass loss during the release period of this event.

- The instrument line releases coolant into the Reactor Building for 30 minutes at normal operating temperature and pressure. Following this time period, the operator detects the event, scrams the reactor and initiates reactor depressurization.
- Reactor coolant is released to the Reactor Building, until the reactor is depressurized.
- The flow from the instrument line is limited by reactor pressure and a 6-mm (0.25-inch) diameter flow restricting orifice inside the drywell. The Moody critical blowdown model is applicable, and the flow is critical at the orifice (Reference 15.4-6).

15.4.8.5 Radiological Analysis

15.4.8.5.1 General

The radiological analysis is based upon conservative assumptions considered acceptable to the NRC. The assumptions found in Table 15.4-17 assume that all of the iodine available in the flashed water is transported via the Reactor Building to the environment without treatment.

15.4.8.5.2 Fission Product Release

The iodine activity in the coolant is assumed to be at the maximum equilibrium Technical Specification limit (see MSLBA in Subsection 15.4.5.5) for continuous operation. Based on data in Table 15.4-17, the amount of iodine released to the Reactor Building atmosphere and to the environment is presented in Table 15.4-18.

15.4.8.5.3 Assumptions Requiring Confirmation

Site parameter assumptions in the radiological analysis are confirmed per Subsection 2.0.1.

15.4.8.5.4 Results

The calculated exposures for the analysis are presented in Table 15.4-19, and are less than the regulatory guideline exposures.

15.4.9 RWCU/SDC System Line Failure Outside Containment

15.4.9.1 Identification of Causes

To evaluate liquid process line pipe breaks outside containment, the failure of a cleanup water line is assumed to evaluate the response of the plant design to this postulated event. The postulated break of the cleanup water line, representing the most significant liquid reactor coolant line outside containment, provides the envelope evaluation for this type of break. The break is assumed to be instantaneous, circumferential and downstream of the outermost isolation valve.

15.4.9.2 Sequence of Events and Systems Operation

15.4.9.2.1 Sequence of Events

The sequence of events is presented in Table 15.4-20.

15.4.9.2.2 Identification of Operator Actions

Because automatic actuation and operation of the ECCS is a system design basis, no operator actions are required or credited in the radiological consequence analysis. However, the operator should perform the following (shown for informational purposes only):

- Determine that a line break has occurred
- Ensure that if vessel water level is below level 3 that reactor has scrammed,
- Confirm RWCU/SDC System containment isolation valves closed,
- Monitor vessel water level and ensure actuation of ECCS as needed, and

- Implement site radiation incident procedures.

These actions occur over an elapsed time of 3–4 hours.

15.4.9.2.3 Systems Operation

It is assumed that the normally operating plant instrument and controls are functioning. Credit is taken for the actuation of the ECCS. The Reactor Protection System, SRVs, ECCS, and safety-related functions of the Control Rod Drive system are assumed to function properly to ensure a safe shutdown.

The ESF systems, including the ADS and GDCS, are assumed to operate normally.

15.4.9.3 Core and System Performance

The fuel is covered throughout the transient and there are no pressure or temperature transients sufficient to cause fuel damage.

15.4.9.4 Barrier Performance

Accidents that result in the release of radioactive materials outside the containment are the result of postulated breaches in piping connected to the RCPB or the steam power-conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the main steamlines as presented in Subsection 15.4.5. The cleanup water system piping break is less severe than the main steamline break.

15.4.9.5 Radiological Consequences

15.4.9.5.1 General

The NRC provides no specific regulatory guidelines for the evaluation of this accident; therefore, the analysis presented is based upon conservative assumptions considered acceptable to the NRC.

Specific values of parameters used in the evaluation are presented in Table 15.4-21.

15.4.9.5.2 Fission Product Release

There is no fuel damage as a consequence of this accident. The only activity available for release from the break is that which is present in the reactor coolant and RWCU/SDC System downstream components prior to the break.

Isolation of the line is conservatively analyzed based upon actuation of the flow differential pressure instrumentation. A total of 46 seconds is allowed for differential flow detection and time delay prior to initiating containment isolation valve closure. After the initial 46 seconds containment isolation valves close over a period of 20 seconds. The initial break flow rate is limited to 2218 kg/sec assuming two-phase critical flow for limiting diameter piping inside containment. The initial break flow rate is assumed to remain constant for the initial 50-seconds following the pipe break. The flow rate is assumed to linearly decrease to zero over the subsequent 16-second period. The total break release period for sources inside containment is 66 seconds.

In addition to the flow of reactor coolant out of the break, the total non-filtered inventory contained in the RWCU/SDC System regenerative and non-regenerative heat exchanger is released. Check valves prevent back flow of inventory from the upstream demineralizer. A break on the downstream side of the demineralizer is bounded by the assumed break location due to reduced flow, steam flashing, and radionuclide source concentrations downstream of the heat exchangers and demineralizer.

Reactor coolant radionuclide source terms are calculated consistent with Regulatory Guide 1.183 iodine spiking assumptions provided for analyzing consequences of a Main Steam Line Break for a BWR. Noble gas activity in the reactor coolant is negligible and is therefore ignored in this analysis.

A summary of RWCU/SDC System line break accident radiological consequence assumptions are provided in Table 15.4-21.

15.4.9.5.3 Fission Product Transport to the Environment

It is conservatively assumed that the release to the environment is instantaneous, with no iodine plateout. Fission product releases to the environment are presented in Table 15.4-22.

15.4.9.5.4 Assumptions Requiring Confirmation

Site parameter assumptions in the radiological analysis are confirmed per Subsection 2.0.1.

15.4.9.5.5 Results

The calculated exposures for the analysis are presented in Table 15.4-23 and are less than the regulatory guideline exposures.

15.4.10 Spent Fuel Cask Drop Accident

15.4.10.1 Identification of Causes

The fuel building design is such that a spent fuel cask drop height of 9.2 m, as specified in SRP 15.7.5, is not exceeded. This feature, along with administrative procedures limiting the travel range of the Fuel Building crane during cask handling activities, precludes damage of equipment or release of radioactivity due to dropping of a spent fuel shipping cask. Therefore, the radiological consequences of this accident are not evaluated.

15.4.10.2 Radiological Analysis

As stated above, the radiological consequences of this accident are not evaluated.

15.4.11 COL Information

15.4-1-A EAB X/Q Value (Deleted)

15.4-2-A LPZ X/Q Value (Deleted)

15.4-3-A Fuel Handling Accident (Deleted)

15.4-4-A Loss of Coolant Accident (Deleted)

15.4-5-A Main Steam Line Break Accident (Deleted)

15.4-6-A Feedwater Line Break Accident (Deleted)

15.4-7-A Instrument Line Break Accident (Deleted)

15.4-8-A RWCU/SDC Line Break Accident (Deleted)

15.4.12 References

15.4-1 General Electric Co., "Radiological Accident Evaluation - The CONAC04A Code," NEDO-32708, August 1997.

15.4-2 Electric Power Research Institute, "Advanced Light Water Reactor Utility Requirements Document," Volume III.

15.4-3 General Electric Company, "Anticipated Chemical Behavior of Iodine under LOCA Conditions," NEDO-25370, January 1981.

15.4-4 GE Nuclear Energy, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," NEDC-31858P (GE proprietary), Revision 2, September 1993.

15.4-5 General Electric Company, "Alternatives to Current Procedures Used to Estimate Concentrations in Building Wakes," 21st DOE/NRC Nuclear Air Cleaning Conference, pgs 714-729.

15.4-6 General Electric Company, "Maximum Two-Phase Vessel Blowdown from Pipes," ASME Paper Number 65-WA/HT-1, March 15, 1965.

15.4-7 NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," USNRC, April 1998.

15.4-8 VTT Energy, "Investigation on Aerosol Deposition in a Heat Exchanger Tube," Jouni Hoklanen, Ari Auvinen, Tommi Renvall, Wolfgang Ludwig, Joma Jokriniemi, VTT Research Report ENE53/46/2000, August 2001.

15.4-9 NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," USNRC, July 1996.

15.4-10 ABWR Design Control Document, Section 19E.

15.4-11 NUREG/CR-0009, "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels," USNRC, October 1978.

15.4-12 NUREG/CR-6119, "MELCOR Computer Code Manuals," USNRC, September 2005.

15.4-13 NEDE-33279P, "ESBWR Containment Fission Product Removal Evaluation Model," October 2006.

Table 15.4-1

Fuel Handling Accident Sequence of Events

Sequence of Events	Elapsed Time
Channeled fuel bundle is being handled by a crane over reactor core. Crane motion changes from horizontal and the fuel grapple releases, dropping the bundle. The channeled bundle is assumed to strike channeled bundles in the reactor core.	0
Some rods in both the dropped and struck bundles fail, releasing radioactive gases to the pool water.	0
Gases pass from the water to the Reactor Building, fuel handling area.	0
The Reactor Building ventilation system high radiation alarm alerts plant personnel.	0+
Operator actions begin.	0+

Table 15.4-2

FHA Parameters

I. Data and Assumptions Used to Estimate Source Terms		
A.	Power level, MWt	4590
B.	Core Source Term	Table 15B-1
C.	Plenum Activity Radioactivity for I-131, % Radioactivity for Kr-85, % Radioactivity for other noble gases, % Radioactivity for other halogens, % Radioactivity for alkali metals, %	 8 10 5 5 12
D.	Radial peaking factor for damaged rods	1.5
E.	Duration of accident, hr	2
F	Total No. of Bundles in Core	1132
G.	No. bundles damaged	2
H.	Minimum time after shutdown to accident, hr	24
I.	Average fuel exposure, MWd/MT	35,000
II. Data and Assumptions Used to Estimate Activity Released		
A.	Species fraction	
	Released From Fuel	
	Organic iodine, %	0.15
	Elemental iodine, %	4.85
	Particulate iodine, %	95
	Noble gas, %	100
	Reactor Building/Fuel Building Atmosphere	
	Organic iodine, %	43
	Elemental iodine, %	57
	Particulate iodine, %	0
	Noble gas, %	100
B.	Pool Water Level, m (ft)	≥7.01 (23.0)
C.	Pool Retention decontamination factor	
	Iodine (effective)	200

Table 15.4-2

FHA Parameters

Noble gas	1
Alkali metals/particulates	Infinite
C. Reactor Building release rate, %/hr	
0 – 1.95 hours	500
1.95 – 2.0 hours	1.0E+08
III. Control Room Parameters	
A. Control Room Volume, m^3 (ft^3)	2.2E+03 (7.8E+04)
B. Unfiltered intake, l/s (cfm)	200 (424)$^+$
C. Filtered intake, l/s (cfm)	0 (0)
D. Unfiltered inleakage, l/s (cfm)	0 (0)
E. Occupancy Factors	
0 – 1 day	1.0
1 – 4 days	0.6
4 – 30 days	0.4
III. Dispersion and Dose Data	
A. Atmospheric Dispersion Factors**	
Exclusion Area Boundary, sec./m^3	2.00E-03
Low Population Zone, sec/m^3	
0-8 hours	1.90E-04
8 hours - 30 days	N/A*
Control Room	
Reactor Building Release	
0-2 hours	1.50E-03
2 hours - 30 days	N/A*
Fuel Building Release	
0-2 hours	2.80E-03

Table 15.4-2

FHA Parameters

2 hours - 30 days	N/A*
B. Dose conversion assumptions	RG 1.183
C. Activity inventory/releases	Table 15.4-3
D. Dose evaluations	Table 15.4-4

\+ The design flow rate for the control room normal ventilation system is 200 l/s (424 cfm), however a value of 212.4 l/s (450 cfm) was conservatively used in the actual dose consequence analysis.

* Since the release lasts only two hours, dispersion factors > 2 hours do not impact the calculated doses.

** See Table 2.0-1

Table 15.4-3

FHA Activity Released from Fuel

Isotope	(Ci)	(MBq)
I-131	2.71E+04	1.00E+09
I-132	1.94E+01	7.18E+05
I-133	1.71E+04	6.32E+08
I-134	2.41E-04	8.91E+00
I-135	2.88E+03	1.07E+08
Kr-85m	2.48E+02	9.19E+06
Kr-85	4.58E+02	1.70E+07
Kr-87	2.04E-02	7.57E+02
Kr-88	3.95E+01	1.46E+06
Xe-133	3.31E+04	1.22E+09
Xe-135	2.02E+03	7.46E+07
Cs-134	8.84E+03	3.27E+08
Cs-136	2.92E+03	1.08E+08
Cs-137	5.74E+03	2.12E+08
Rb-86	1.01E+02	3.74E+06

Table 15.4-3a FHA Isotopic Release to Environment		
Isotope	**Activity (Ci)**	**Activity (MBq)**
I-131	1.36E+02	5.02E+06
I-132	9.12E-02	3.37E+03
I-133	8.48E+01	3.14E+06
I-134	1.04E-06	3.84E-02
I-135	1.41E+01	5.21E+05
Kr-85m	2.41E+02	8.93E+06
Kr-85	4.58E+02	1.69E+07
Kr-87	1.84E-02	6.81E+02
Kr-88	3.76E+01	1.39E+06
Xe-133	3.31E+04	1.22E+09
Xe-135	2.01E+03	7.42E+07

Table 15.4-4

FHA Analysis Results

Accident Location, Exposure Location and Time Duration	Maximum Calculated TEDE (rem)	10 CFR 50.34(a)(1) Acceptance Criterion TEDE (rem)
Reactor Building Release Results:		
Exclusion Area Boundary (EAB) for the worst 2 hours	4.13	6.3
Outer boundary of Low Population Zone (LPZ) for the duration of the accident (30 days)	0.39	6.3
Control Room dose for the duration of the accident (30 days)	2.58	5.0
Fuel Building Release Results:		
Exclusion Area Boundary (EAB) for the worst 2 hours	4.13	6.3
Outer boundary of Low Population Zone (LPZ) for the duration of the accident (30 days)	0.39	6.3
Control Room dose for the duration of the accident (30 days)	4.82	5.0

Table 15.4-5

Loss-of-Coolant Accident Parameters

I. Data and Assumptions used to estimate source terms.	
A. Power Level, MWt	4590
B. Fraction of Core Inventory Released	RG 1.183, Table 1
C. Iodine Chemical Species	
Elemental, %	4.85
Particulate, %	95
Organic, %	0.15
D. Time to fuel damage, min.	20
E. Core Source Term	Table 15B-1
II. Data and Assumptions used to estimate activity released	
A. Primary Containment	
Leak rate, %/day	0.5
Fraction to Reactor Building (Leak Rate, %/day)	0.98 (0.49)
Fraction to PCCS Airspace (Leak Rate, %/day)	0.02 (0.01)
Volume, m^3	7.206E+03
Elemental iodine removal rate constant, hr^{-1} (0-12 hrs)	0.92
Aerosol removal rate constants, hr^{-1}	
0 – 0.333 hr	0.0
0.333 – 0.833 hr	5.0
0.833 – 2.333 hrs	3.0
2.333 – 3.0 hrs	1.0
3.0 – 4.0 hrs	0.8
4.0 – 5.0 hrs	1.0
5.0 – 6.0 hrs	0.6
6.0 – 7.0 hrs	0.4
7.0 – 9.5 hrs	0.2
9.5 – 12.0 hrs	0.1
>12.0	0.0

Table 15.4-5

Loss-of-Coolant Accident Parameters

B. Reactor Building	
Leak rate, %/day	50
Mixing efficiency, %	40
Volume, m^3	6.05E+04
C. Condenser Data	
Free air volume, m^3	6.23E+03
Fraction of volume involved, %	20
Iodine removal factors	
Particulate, %	99.5
Elemental, %	99.5
Organic, %	0
D. MSIV Data	
MSIV leakage (total all lines), m^3/min	6.23E-02
Plateout factors	0 (Not Credited)
III. Control Room Parameters	
A. Control Room Volume, m^3	2.46E+03
B. EBAS Flow, m^3/min	2.83
C. Unfiltered inleakage, m^3/min	1.13E-02
D. Occupancy Factors	
0 – 1 day	1.0
1 – 4 days	0.6
4 – 30 days	0.4
IV. Dispersion and Dose Data	
A. Meteorology	Table 15.4-9
B. Method of Dose Calculation	RG 1.183, Section 4
C. Dose Conversion Factor Assumptions	RG 1.183, Section 4
D. Breathing Rate Assumptions	
Control Room, m^3/sec.	3.5E-04
Off-Site, m^3/sec.	

Table 15.4-5

Loss-of-Coolant Accident Parameters

0 – 8 hours	3.5E-04
8 – 24 hours	1.8E-04
> 24 hours	2.3E-04
E. Activity / Inventory Releases	Tables 15.4-6, 7, 8
F. Dose Evaluations	Table 15.4-9

Table 15.4-6

LOCA Compartment Inventories (MBq)

Drywell								
	0.5 hours	2 hours	8 hours	12 hours	24 hours	4 days	7 days	30 days
Co-58	0.0E+00	1.3E+07	3.1E+05	2.0E+05	1.9E+05	1.8E+05	1.7E+05	8.8E+04
Co-60	0.0E+00	1.2E+07	3.0E+05	1.9E+05	1.9E+05	1.8E+05	1.7E+05	1.1E+05
Kr-85	2.8E+09	5.7E+10	5.6E+10	5.6E+10	5.6E+10	5.3E+10	5.0E+10	3.3E+10
Kr-85m	5.5E+10	8.7E+11	3.4E+11	1.9E+11	2.9E+10	3.9E+05	5.4E+00	0.0E+00
Kr-87	7.7E+10	6.8E+11	2.6E+10	3.5E+09	4.1E+06	0.0E+00	0.0E+00	0.0E+00
Kr-88	1.4E+11	1.9E+12	4.4E+11	1.8E+11	8.8E+09	2.0E+02	4.3E-06	0.0E+00
Rb-86	2.0E+08	4.7E+08	1.1E+07	7.2E+06	7.0E+06	5.9E+06	5.0E+06	1.4E+06
Sr-89	0.0E+00	2.0E+10	4.7E+08	3.1E+08	3.0E+08	2.7E+08	2.5E+08	1.2E+08
Sr-90	0.0E+00	2.0E+09	4.7E+07	3.0E+07	3.0E+07	2.9E+07	2.7E+07	1.8E+07
Sr-91	0.0E+00	2.1E+10	3.3E+08	1.6E+08	6.5E+07	3.2E+05	1.6E+03	0.0E+00
Sr-92	0.0E+00	1.5E+10	7.6E+07	1.9E+07	8.2E+05	7.8E-03	0.0E+00	0.0E+00
Y-90	0.0E+00	2.9E+07	3.6E+06	3.4E+06	6.7E+06	1.8E+07	2.3E+07	1.8E+07
Y-91	0.0E+00	2.6E+08	6.6E+06	4.4E+06	4.6E+06	4.4E+06	4.0E+06	2.1E+06
Y-92	0.0E+00	1.3E+09	1.2E+08	5.3E+07	7.9E+06	7.7E+00	5.6E-06	0.0E+00
Y-93	0.0E+00	2.7E+08	4.2E+06	2.1E+06	9.0E+05	6.1E+03	4.1E+01	0.0E+00
Zr-95	0.0E+00	3.6E+08	8.6E+06	5.5E+06	5.5E+06	5.0E+06	4.6E+06	2.4E+06
Zr-97	0.0E+00	3.4E+08	6.3E+06	3.5E+06	2.1E+06	1.0E+05	5.1E+03	5.1E-07
Nb-95	0.0E+00	3.6E+08	8.7E+06	5.6E+06	5.5E+06	5.3E+06	5.0E+06	3.1E+06
Mo-99	0.0E+00	4.7E+09	1.0E+08	6.5E+07	5.6E+07	2.5E+07	1.1E+07	2.3E+04
Tc-99m	0.0E+00	4.2E+09	9.8E+07	6.2E+07	5.7E+07	2.6E+07	1.1E+07	2.3E+04
Ru-103	0.0E+00	3.8E+09	8.9E+07	5.8E+07	5.7E+07	5.1E+07	4.6E+07	2.0E+07
Ru-105	0.0E+00	1.8E+09	1.6E+07	6.0E+06	8.6E+05	1.1E+01	1.3E-04	0.0E+00
Ru-106	0.0E+00	1.3E+09	3.1E+07	2.0E+07	2.0E+07	1.9E+07	1.8E+07	1.1E+07
Rh-105	0.0E+00	2.3E+09	5.1E+07	3.2E+07	2.5E+07	5.8E+06	1.3E+06	1.8E+01
Sb-127	0.0E+00	5.1E+09	1.2E+08	7.3E+07	6.6E+07	3.7E+07	2.0E+07	2.2E+05
Sb-129	0.0E+00	1.1E+10	9.9E+07	3.6E+07	4.9E+06	4.5E+01	4.1E-04	0.0E+00
Te-127	0.0E+00	5.2E+09	1.2E+08	7.7E+07	7.2E+07	4.5E+07	2.9E+07	5.8E+06
Te-127m	0.0E+00	6.9E+08	1.6E+07	1.1E+07	1.1E+07	1.0E+07	9.4E+06	5.5E+06
Te-129	0.0E+00	1.3E+10	1.4E+08	6.2E+07	3.7E+07	2.7E+07	2.4E+07	1.0E+07
Te-129m	0.0E+00	2.3E+09	5.5E+07	3.6E+07	3.5E+07	3.1E+07	2.8E+07	1.2E+07
Te-131m	0.0E+00	6.8E+09	1.4E+08	8.4E+07	6.2E+07	1.1E+07	2.0E+06	3.9E+00
Te-132	0.0E+00	7.0E+10	1.6E+09	9.9E+08	8.8E+08	4.4E+08	2.2E+08	1.1E+06
I-131	8.8E+10	2.7E+11	7.6E+09	5.5E+09	5.2E+09	3.8E+09	2.8E+09	2.6E+08

Table 15.4-6

LOCA Compartment Inventories (MBq)

Drywell								
	0.5 hours	2 hours	8 hours	12 hours	24 hours	4 days	7 days	30 days
I-132	1.2E+11	3.6E+11	3.2E+09	1.5E+09	1.0E+09	5.2E+08	2.6E+08	1.3E+06
I-133	1.8E+11	5.2E+11	1.2E+10	7.9E+09	5.2E+09	4.5E+08	3.8E+07	2.6E-01
I-134	1.0E+11	9.8E+10	2.5E+07	9.8E+05	5.7E+01	0.0E+00	0.0E+00	0.0E+00
I-135	1.6E+11	4.1E+11	6.3E+09	3.1E+09	8.5E+08	4.2E+05	2.1E+02	0.0E+00
Xe-133	4.6E+11	9.3E+12	8.9E+12	8.7E+12	8.1E+12	5.2E+12	3.3E+12	1.1E+11
Xe-135	1.6E+11	3.3E+12	2.1E+12	1.6E+12	6.2E+11	2.4E+09	9.5E+06	0.0E+00
Cs-134	1.7E+10	4.0E+10	9.6E+08	6.2E+08	6.1E+08	5.8E+08	5.5E+08	3.6E+08
Cs-136	5.8E+09	1.4E+10	3.3E+08	2.1E+08	2.0E+08	1.6E+08	1.3E+08	2.6E+07
Cs-137	1.1E+10	2.6E+10	6.2E+08	4.0E+08	4.0E+08	3.8E+08	3.6E+08	2.4E+08
Ba-139	0.0E+00	1.1E+10	1.3E+07	1.4E+06	2.7E+03	0.0E+00	0.0E+00	0.0E+00
Ba-140	0.0E+00	3.5E+10	8.3E+08	5.3E+08	5.1E+08	4.1E+08	3.3E+08	6.4E+07
La-140	0.0E+00	5.9E+08	9.5E+07	9.0E+07	1.7E+08	3.6E+08	3.6E+08	7.4E+07
La-141	0.0E+00	2.2E+08	1.9E+06	6.3E+05	7.1E+04	2.0E-01	5.9E-07	0.0E+00
La-142	0.0E+00	1.1E+08	1.8E+05	2.3E+04	8.9E+01	0.0E+00	0.0E+00	0.0E+00
Ce-141	0.0E+00	8.5E+08	2.0E+07	1.3E+07	1.3E+07	1.1E+07	1.0E+07	4.1E+06
Ce-143	0.0E+00	7.5E+08	1.6E+07	9.4E+06	7.2E+06	1.5E+06	3.2E+05	1.9E+00
Ce-144	0.0E+00	6.9E+08	1.6E+07	1.1E+07	1.0E+07	9.8E+06	9.3E+06	5.9E+06
Pr-143	0.0E+00	3.1E+08	7.5E+06	4.9E+06	4.9E+06	4.5E+06	3.8E+06	7.8E+05
Nd-147	0.0E+00	1.3E+08	3.1E+06	2.0E+06	1.9E+06	1.5E+06	1.2E+06	1.9E+05
Np-239	0.0E+00	9.5E+09	2.1E+08	1.3E+08	1.1E+08	4.3E+07	1.7E+07	1.3E+04
Pu-238	0.0E+00	1.7E+06	4.0E+04	2.6E+04	2.6E+04	2.4E+04	2.3E+04	1.6E+04
Pu-239	0.0E+00	2.0E+05	4.9E+03	3.1E+03	3.1E+03	3.0E+03	2.8E+03	1.9E+03
Pu-240	0.0E+00	2.6E+05	6.3E+03	4.1E+03	4.0E+03	3.8E+03	3.6E+03	2.4E+03
Pu-241	0.0E+00	7.6E+07	1.8E+06	1.2E+06	1.2E+06	1.1E+06	1.0E+06	7.0E+05
Am-241	0.0E+00	3.4E+04	8.2E+02	5.3E+02	5.3E+02	5.2E+02	5.0E+02	4.1E+02
Cm-242	0.0E+00	8.1E+06	1.9E+05	1.2E+05	1.2E+05	1.1E+05	1.1E+05	6.5E+04
Cm-244	0.0E+00	3.9E+05	9.3E+03	6.0E+03	6.0E+03	5.7E+03	5.4E+03	3.6E+03

Table 15.4-6
LOCA Compartment Inventories (MBq)

Condenser

	0.5 hours	2 hours	8 hours	12 hours	24 hours	4 days	7 days	30 days
Co-58	0.0E+00	7.8E+03	1.5E+04	1.5E+04	1.5E+04	1.7E+04	1.7E+04	1.2E+04
Co-60	0.0E+00	7.6E+03	1.4E+04	1.5E+04	1.5E+04	1.7E+04	1.8E+04	1.5E+04
Kr-85	3.7E+05	2.3E+07	2.0E+08	3.0E+08	6.3E+08	2.2E+09	3.2E+09	4.2E+09
Kr-85m	7.2E+06	3.6E+08	1.2E+09	1.0E+09	3.2E+08	1.6E+04	3.5E-01	0.0E+00
Kr-87	9.9E+06	2.8E+08	8.9E+07	1.8E+07	4.7E+04	8.1E+00	0.0E+00	0.0E+00
Kr-88	1.8E+07	8.0E+08	1.6E+09	9.7E+08	1.0E+08	0.0E+00	0.0E+00	0.0E+00
Rb-86	3.5E+04	3.5E+05	6.1E+05	6.1E+05	6.1E+05	5.9E+05	5.4E+05	1.9E+05
Sr-89	0.0E+00	1.2E+07	2.3E+07	2.3E+07	2.4E+07	2.5E+07	2.6E+07	1.6E+07
Sr-90	0.0E+00	1.2E+06	2.3E+06	2.3E+06	2.4E+06	2.7E+06	2.8E+06	2.4E+06
Sr-91	0.0E+00	1.3E+07	1.6E+07	1.2E+07	5.2E+06	3.0E+04	1.7E+02	0.0E+00
Sr-92	0.0E+00	9.1E+06	3.7E+06	1.5E+06	6.5E+04	7.2E-04	0.0E+00	0.0E+00
Y-90	0.0E+00	1.2E+04	2.2E+04	2.2E+04	2.0E+04	9.9E+03	4.8E+03	1.0E+01
Y-91	0.0E+00	1.6E+05	3.0E+05	3.0E+05	3.1E+05	3.3E+05	3.4E+05	2.2E+05
Y-92	0.0E+00	1.0E+05	6.2E+04	3.1E+04	2.8E+03	2.4E-03	0.0E+00	0.0E+00
Y-93	0.0E+00	1.6E+05	2.0E+05	1.6E+05	7.1E+04	5.7E+02	4.3E+00	0.0E+00
Zr-95	0.0E+00	2.2E+05	4.2E+05	4.2E+05	4.3E+05	4.7E+05	4.8E+05	3.2E+05
Zr-97	0.0E+00	2.1E+05	3.1E+05	2.7E+05	1.7E+05	9.6E+03	5.3E+02	0.0E+00
Nb-95	0.0E+00	2.2E+05	4.2E+05	4.2E+05	4.3E+05	4.5E+05	4.5E+05	2.4E+05
Mo-99	0.0E+00	2.8E+06	5.1E+06	5.0E+06	4.5E+06	2.3E+06	1.2E+06	3.0E+03
Tc-99m	0.0E+00	2.0E+06	1.9E+06	1.3E+06	3.1E+05	8.7E+01	2.3E-02	0.0E+00
Ru-103	0.0E+00	2.3E+06	4.4E+06	4.4E+06	4.5E+06	4.7E+06	4.7E+06	2.7E+06
Ru-105	0.0E+00	1.1E+06	8.0E+05	4.6E+05	6.9E+04	1.0E+00	1.4E-05	0.0E+00
Ru-106	0.0E+00	8.0E+05	1.5E+06	1.6E+06	1.6E+06	1.8E+06	1.8E+06	1.5E+06
Rh-105	0.0E+00	1.3E+06	2.3E+06	2.1E+06	1.7E+06	4.7E+05	1.2E+05	2.1E+00
Sb-127	0.0E+00	3.1E+06	5.7E+06	5.6E+06	5.3E+06	3.4E+06	2.1E+06	2.8E+04
Sb-129	0.0E+00	6.7E+06	4.9E+06	2.7E+06	3.9E+05	4.1E+00	4.2E-05	0.0E+00
Te-127	0.0E+00	2.7E+06	3.3E+06	2.6E+06	1.1E+06	5.6E+03	2.9E+01	0.0E+00
Te-127m	0.0E+00	4.2E+05	8.0E+05	8.1E+05	8.3E+05	9.1E+05	9.4E+05	6.9E+05
Te-129	0.0E+00	2.4E+06	1.3E+05	1.4E+04	9.2E+00	0.0E+00	0.0E+00	0.0E+00
Te-129m	0.0E+00	1.4E+06	2.7E+06	2.7E+06	2.7E+06	2.9E+06	2.8E+06	1.5E+06
Te-131m	0.0E+00	4.1E+06	6.9E+06	6.4E+06	4.9E+06	1.0E+06	2.1E+05	5.1E-01
Te-132	0.0E+00	4.3E+07	7.7E+07	7.6E+07	7.0E+07	4.1E+07	2.3E+07	1.5E+05
I-131	1.5E+07	1.9E+08	3.3E+08	3.4E+08	3.4E+08	3.1E+08	2.6E+08	3.4E+07

Table 15.4-6
LOCA Compartment Inventories (MBq)

Condenser

I-132	1.8E+07	1.4E+08	4.1E+07	1.4E+07	3.5E+05	1.6E-04	0.0E+00	0.0E+00
I-133	3.1E+07	3.6E+08	5.4E+08	4.8E+08	3.4E+08	3.7E+07	3.6E+06	3.5E-02
I-134	1.8E+07	6.8E+07	1.1E+06	6.1E+04	3.7E+00	0.0E+00	0.0E+00	0.0E+00
I-135	2.7E+07	2.9E+08	2.8E+08	1.9E+08	5.5E+07	3.5E+04	2.0E+01	0.0E+00
Xe-133	6.0E+07	3.8E+09	3.1E+10	4.6E+10	9.1E+10	2.1E+11	2.1E+11	1.3E+10
Xe-135	1.9E+07	1.1E+09	5.7E+09	6.6E+09	5.4E+09	7.7E+07	4.7E+05	0.0E+00
Cs-134	3.0E+06	3.0E+07	5.2E+07	5.2E+07	5.3E+07	5.7E+07	5.9E+07	4.8E+07
Cs-136	1.0E+06	1.0E+07	1.8E+07	1.8E+07	1.8E+07	1.6E+07	1.4E+07	3.5E+06
Cs-137	1.9E+06	1.9E+07	3.4E+07	3.4E+07	3.5E+07	3.7E+07	3.9E+07	3.2E+07
Ba-139	0.0E+00	7.0E+06	6.5E+05	1.0E+05	2.2E+02	0.0E+00	0.0E+00	0.0E+00
Ba-140	0.0E+00	2.2E+07	4.1E+07	4.1E+07	4.1E+07	3.9E+07	3.5E+07	8.4E+06
La-140	0.0E+00	2.1E+05	3.7E+05	3.5E+05	2.9E+05	9.4E+04	2.9E+04	1.8E+00
La-141	0.0E+00	1.4E+05	9.0E+04	4.8E+04	5.6E+03	1.9E-02	0.0E+00	0.0E+00
La-142	0.0E+00	7.0E+04	9.0E+03	1.8E+03	7.0E+00	0.0E+00	0.0E+00	0.0E+00
Ce-141	0.0E+00	5.1E+05	9.8E+05	9.9E+05	1.0E+06	1.0E+06	1.0E+06	5.4E+05
Ce-143	0.0E+00	4.6E+05	7.7E+05	7.2E+05	5.7E+05	1.4E+05	3.3E+04	2.6E-01
Ce-144	0.0E+00	4.2E+05	8.0E+05	8.1E+05	8.3E+05	9.2E+05	9.6E+05	7.7E+05
Pr-143	0.0E+00	1.9E+05	3.5E+05	3.5E+05	3.5E+05	3.4E+05	3.1E+05	8.1E+04
Nd-147	0.0E+00	8.2E+04	1.5E+05	1.5E+05	1.5E+05	1.4E+05	1.2E+05	2.5E+04
Np-239	0.0E+00	5.8E+06	1.0E+07	9.9E+06	8.8E+06	4.0E+06	1.8E+06	1.7E+03
Pu-238	0.0E+00	1.0E+03	2.0E+03	2.0E+03	2.0E+03	2.3E+03	2.4E+03	2.0E+03
Pu-239	0.0E+00	1.2E+02	2.4E+02	2.4E+02	2.5E+02	2.7E+02	2.9E+02	2.5E+02
Pu-240	0.0E+00	1.6E+02	3.1E+02	3.1E+02	3.2E+02	3.5E+02	3.7E+02	3.2E+02
Pu-241	0.0E+00	4.6E+04	8.9E+04	9.0E+04	9.2E+04	1.0E+05	1.1E+05	9.2E+04
Am-241	0.0E+00	2.1E+01	4.0E+01	4.1E+01	4.2E+01	4.6E+01	4.9E+01	4.2E+01
Cm-242	0.0E+00	4.9E+03	9.4E+03	9.5E+03	9.7E+03	1.1E+04	1.1E+04	8.6E+03
Cm-244	0.0E+00	2.4E+02	4.6E+02	4.6E+02	4.7E+02	5.3E+02	5.5E+02	4.7E+02

Table 15.4-6
LOCA Compartment Inventories (MBq)

Reactor Building								
Isotope	0.5 hours	2 hours	8 hours	12 hours	24 hours	4 days	7 days	30 days
Co-58	0.0E+00	3.1E+03	5.4E+03	5.1E+03	4.4E+03	2.3E+03	1.8E+03	9.0E+02
Co-60	0.0E+00	3.0E+03	5.2E+03	5.0E+03	4.3E+03	2.3E+03	1.8E+03	1.1E+03
Kr-85	1.4E+05	9.1E+06	7.3E+07	1.1E+08	2.1E+08	4.6E+08	4.9E+08	3.4E+08
Kr-85m	2.8E+06	1.4E+08	4.5E+08	3.7E+08	1.1E+08	3.4E+03	5.3E-02	1.5E+04
Kr-87	3.9E+06	1.1E+08	3.3E+07	6.7E+06	1.5E+04	0.0E+00	0.0E+00	0.0E+00
Kr-88	7.1E+06	3.1E+08	5.8E+08	3.5E+08	3.3E+07	1.7E+00	0.0E+00	0.0E+00
Rb-86	1.4E+04	1.4E+05	2.2E+05	2.1E+05	1.7E+05	8.1E+04	5.5E+04	0.0E+00
Sr-89	0.0E+00	4.8E+06	8.3E+06	8.0E+06	6.8E+06	3.6E+06	2.7E+06	1.2E+06
Sr-90	0.0E+00	4.7E+05	8.2E+05	7.9E+05	6.8E+05	3.7E+05	2.9E+05	1.8E+05
Sr-91	0.0E+00	5.1E+06	5.7E+06	4.2E+06	1.5E+06	4.2E+03	1.7E+01	0.0E+00
Sr-92	0.0E+00	3.5E+06	1.3E+06	5.0E+05	1.8E+04	1.0E-04	0.0E+00	0.0E+00
Y-90	0.0E+00	9.6E+03	6.5E+04	9.1E+04	1.5E+05	2.4E+05	2.5E+05	1.8E+05
Y-91	0.0E+00	6.2E+04	1.2E+05	1.2E+05	1.0E+05	5.8E+04	4.4E+04	2.1E+04
Y-92	0.0E+00	7.2E+05	2.2E+06	1.4E+06	1.8E+05	1.0E-01	0.0E+00	0.0E+00
Y-93	0.0E+00	6.3E+04	7.4E+04	5.5E+04	2.0E+04	7.9E+01	4.5E-01	2.4E+04
Zr-95	0.0E+00	8.6E+04	1.5E+05	1.4E+05	1.2E+05	6.6E+04	5.0E+04	0.0E+00
Zr-97	0.0E+00	8.1E+04	1.1E+05	9.1E+04	4.7E+04	1.4E+03	5.6E+01	0.0E+00
Nb-95	0.0E+00	8.7E+04	1.5E+05	1.5E+05	1.3E+05	6.9E+04	5.4E+04	3.1E+04
Mo-99	0.0E+00	1.1E+06	1.8E+06	1.7E+06	1.3E+06	3.3E+05	1.2E+05	2.3E+02
Tc-99m	0.0E+00	1.0E+06	1.7E+06	1.6E+06	1.3E+06	3.4E+05	1.2E+05	2.4E+02
Ru-103	0.0E+00	9.0E+05	1.6E+06	1.5E+06	1.3E+06	6.7E+05	5.0E+05	2.1E+05
Ru-105	0.0E+00	4.2E+05	2.9E+05	1.6E+05	1.9E+04	1.4E-01	1.5E-06	0.0E+00
Ru-106	0.0E+00	3.1E+05	5.5E+05	5.3E+05	4.5E+05	2.5E+05	1.9E+05	1.2E+05
Rh-105	0.0E+00	5.5E+05	9.0E+05	8.2E+05	5.7E+05	7.6E+04	1.5E+04	1.8E-01
Sb-127	0.0E+00	1.2E+06	2.0E+06	1.9E+06	1.5E+06	4.8E+05	2.2E+05	2.2E+03
Sb-129	0.0E+00	2.6E+06	1.7E+06	9.3E+05	1.1E+05	5.8E-01	4.4E-06	0.0E+00
Te-127	0.0E+00	1.2E+06	2.1E+06	2.0E+06	1.6E+06	5.9E+05	3.1E+05	5.9E+04
Te-127m	0.0E+00	1.6E+05	2.9E+05	2.8E+05	2.4E+05	1.3E+05	1.0E+05	5.6E+04
Te-129	0.0E+00	3.0E+06	2.5E+06	1.6E+06	8.3E+05	3.5E+05	2.6E+05	1.0E+05
Te-129m	0.0E+00	5.5E+05	9.7E+05	9.2E+05	7.9E+05	4.1E+05	3.0E+05	1.2E+05
Te-131m	0.0E+00	1.6E+06	2.5E+06	2.2E+06	1.4E+06	1.5E+05	2.2E+04	4.0E-02
Te-132	0.0E+00	1.7E+07	2.8E+07	2.6E+07	2.0E+07	5.7E+06	2.4E+06	1.1E+04
I-131	6.0E+06	7.4E+07	1.2E+08	1.2E+08	9.7E+07	4.6E+07	3.0E+07	2.6E+06
I-132	7.9E+06	8.4E+07	4.9E+07	3.3E+07	2.4E+07	6.9E+06	2.9E+06	1.3E+04

Table 15.4-6
LOCA Compartment Inventories (MBq)

Reactor Building								
I-133	1.2E+07	1.4E+08	1.9E+08	1.7E+08	9.6E+07	5.4E+06	4.1E+05	2.7E-03
I-134	7.1E+06	2.7E+07	3.9E+05	2.1E+04	1.1E+00	0.0E+00	0.0E+00	0.0E+00
I-135	1.1E+07	1.1E+08	1.0E+08	6.6E+07	1.6E+07	5.2E+03	2.3E+00	0.0E+00
Xe-133	2.4E+07	1.5E+09	1.2E+10	1.7E+10	3.0E+10	4.5E+10	3.2E+10	1.1E+09
Xe-135	8.3E+06	5.2E+08	2.8E+09	3.1E+09	2.3E+09	2.1E+07	9.3E+04	0.0E+00
Cs-134	1.2E+06	1.2E+07	1.9E+07	1.8E+07	1.5E+07	7.9E+06	6.0E+06	3.6E+06
Cs-136	4.1E+05	4.1E+06	6.4E+06	6.1E+06	5.0E+06	2.2E+06	1.5E+06	2.7E+05
Cs-137	7.6E+05	7.6E+06	1.2E+07	1.2E+07	9.9E+06	5.1E+06	3.9E+06	2.4E+06
Ba-139	0.0E+00	2.7E+06	2.3E+05	3.6E+04	6.2E+01	0.0E+00	0.0E+00	0.0E+00
Ba-140	0.0E+00	8.4E+06	1.5E+07	1.4E+07	1.2E+07	5.4E+06	3.6E+06	6.5E+05
La-140	0.0E+00	2.2E+05	1.7E+06	2.4E+06	3.9E+06	4.8E+06	3.9E+06	7.5E+05
La-141	0.0E+00	5.3E+04	3.3E+04	1.6E+04	1.6E+03	2.7E-03	0.0E+00	0.0E+00
La-142	0.0E+00	2.7E+04	3.2E+03	6.0E+02	2.0E+00	0.0E+00	0.0E+00	0.0E+00
Ce-141	0.0E+00	2.0E+05	3.5E+05	3.4E+05	2.9E+05	1.5E+05	1.1E+05	4.2E+04
Ce-143	0.0E+00	1.8E+05	2.8E+05	2.5E+05	1.6E+05	2.0E+04	3.4E+03	2.0E-02
Ce-144	0.0E+00	1.6E+05	2.9E+05	2.8E+05	2.4E+05	1.3E+05	1.0E+05	6.0E+04
Pr-143	0.0E+00	7.4E+04	1.3E+05	1.3E+05	1.1E+05	5.9E+04	4.1E+04	8.0E+03
Nd-147	0.0E+00	3.2E+04	5.5E+04	5.2E+04	4.4E+04	2.0E+04	1.3E+04	1.9E+03
Np-239	0.0E+00	2.3E+06	3.7E+06	3.4E+06	2.5E+06	5.7E+05	1.8E+05	1.3E+02
Pu-238	0.0E+00	4.0E+02	7.1E+02	6.8E+02	5.8E+02	3.2E+02	2.5E+02	1.6E+02
Pu-239	0.0E+00	4.9E+01	8.5E+01	8.2E+01	7.0E+01	3.9E+01	3.1E+01	1.9E+01
Pu-240	0.0E+00	6.3E+01	1.1E+02	1.1E+02	9.1E+01	5.0E+01	3.9E+01	2.5E+01
Pu-241	0.0E+00	1.8E+04	3.2E+04	3.1E+04	2.6E+04	1.4E+04	1.1E+04	7.1E+03
Am-241	0.0E+00	8.2E+00	1.4E+01	1.4E+01	1.2E+01	6.8E+00	5.5E+00	4.1E+00
Cm-242	0.0E+00	1.9E+03	3.4E+03	3.2E+03	2.8E+03	1.5E+03	1.2E+03	6.6E+02
Cm-244	0.0E+00	9.3E+01	1.6E+02	1.6E+02	1.3E+02	7.4E+01	5.8E+01	3.6E+01

Table 15.4-7
LOCA Integrated Environment Release (MBq)

	0.5 hr	2 hr	8 hr	12 hours	24 hours	4 days	7 days	30 days
Co-58	0.0E+00	1.0E+02	8.0E+02	1.2E+03	2.4E+03	7.2E+03	1.0E+04	2.5E+04
Co-60	0.0E+00	9.9E+01	7.8E+02	1.2E+03	2.4E+03	7.1E+03	1.0E+04	2.7E+04
Kr-85	3.5E+03	3.5E+05	1.0E+07	2.2E+07	9.2E+07	1.1E+09	2.8E+09	1.8E+10
Kr-85m	7.0E+04	5.7E+06	9.2E+07	1.5E+08	2.3E+08	2.6E+08	2.6E+08	2.6E+08
Kr-87	1.0E+05	5.3E+06	2.8E+07	3.0E+07	3.1E+07	3.1E+07	3.1E+07	3.1E+07
Kr-88	1.8E+05	1.3E+07	1.6E+08	2.2E+08	2.7E+08	2.7E+08	2.7E+08	2.7E+08
Rb-86	3.3E+02	5.1E+03	3.4E+04	5.0E+04	1.0E+05	2.8E+05	3.8E+05	7.3E+05
Sr-89	0.0E+00	1.6E+05	1.2E+06	1.9E+06	3.8E+06	1.1E+07	1.6E+07	3.8E+07
Sr-90	0.0E+00	1.6E+04	1.2E+05	1.9E+05	3.8E+05	1.1E+06	1.6E+06	4.3E+06
Sr-91	0.0E+00	1.8E+05	1.1E+06	1.5E+06	2.1E+06	2.5E+06	2.5E+06	2.5E+06
Sr-92	0.0E+00	1.4E+05	5.5E+05	6.1E+05	6.5E+05	6.5E+05	6.5E+05	6.5E+05
Y-90	0.0E+00	2.1E+02	5.3E+03	1.1E+04	4.4E+04	3.7E+05	7.4E+05	3.3E+06
Y-91	0.0E+00	2.1E+03	1.7E+04	2.6E+04	5.5E+04	1.7E+05	2.5E+05	6.0E+05
Y-92	0.0E+00	9.2E+03	2.7E+05	4.1E+05	5.6E+05	5.8E+05	5.8E+05	5.8E+05
Y-93	0.0E+00	2.2E+03	1.4E+04	1.9E+04	2.8E+04	3.3E+04	3.3E+04	3.3E+04
Zr-95	0.0E+00	2.9E+03	2.3E+04	3.4E+04	6.9E+04	2.0E+05	2.9E+05	7.0E+05
Zr-97	0.0E+00	2.8E+03	1.9E+04	2.7E+04	4.4E+04	6.3E+04	6.3E+04	6.3E+04
Nb-95	0.0E+00	2.9E+03	2.3E+04	3.4E+04	6.9E+04	2.1E+05	3.0E+05	7.8E+05
Mo-99	0.0E+00	3.7E+04	2.8E+05	4.2E+05	8.0E+05	1.8E+06	2.1E+06	2.4E+06
Tc-99m	0.0E+00	3.0E+04	2.5E+05	3.8E+05	7.5E+05	1.8E+06	2.1E+06	2.3E+06
Ru-103	0.0E+00	3.0E+04	2.4E+05	3.5E+05	7.1E+05	2.1E+06	3.0E+06	6.8E+06
Ru-105	0.0E+00	1.5E+04	7.7E+04	9.4E+04	1.1E+05	1.1E+05	1.1E+05	1.1E+05
Ru-106	0.0E+00	1.1E+04	8.2E+04	1.2E+05	2.5E+05	7.5E+05	1.1E+06	2.8E+06
Rh-105	0.0E+00	1.8E+04	1.4E+05	2.1E+05	3.8E+05	7.4E+05	7.9E+05	8.1E+05
Sb-127	0.0E+00	4.1E+04	3.1E+05	4.7E+05	9.0E+05	2.2E+06	2.7E+06	3.2E+06
Sb-129	0.0E+00	9.5E+04	4.8E+05	5.8E+05	6.7E+05	6.8E+05	6.8E+05	6.8E+05
Te-127	0.0E+00	3.9E+04	3.1E+05	4.7E+05	9.4E+05	2.4E+06	3.0E+06	4.4E+06
Te-127m	0.0E+00	5.5E+03	4.3E+04	6.5E+04	1.3E+05	3.9E+05	5.7E+05	1.5E+06
Te-129	0.0E+00	6.8E+04	5.1E+05	6.7E+05	9.6E+05	1.7E+06	2.1E+06	4.0E+06
Te-129m	0.0E+00	1.8E+04	1.4E+05	2.2E+05	4.4E+05	1.3E+06	1.8E+06	4.1E+06
Te-131m	0.0E+00	5.5E+04	4.0E+05	5.8E+05	1.0E+06	1.8E+06	1.9E+06	2.0E+06
Te-132	0.0E+00	5.6E+05	4.3E+06	6.3E+06	1.2E+07	2.9E+07	3.5E+07	4.0E+07
I-131	1.4E+05	2.7E+06	1.9E+07	2.8E+07	5.6E+07	1.7E+08	2.5E+08	4.6E+08

Table 15.4-7

LOCA Integrated Environment Release (MBq)

	0.5 hr	2 hr	8 hr	12 hours	24 hours	4 days	7 days	30 days
I-132	1.7E+05	2.8E+06	1.3E+07	1.6E+07	2.3E+07	4.3E+07	4.9E+07	5.5E+07
I-133	2.9E+05	5.3E+06	3.3E+07	4.7E+07	8.1E+07	1.3E+08	1.4E+08	1.4E+08
I-134	1.9E+05	1.7E+06	2.9E+06	2.9E+06	2.9E+06	2.9E+06	2.9E+06	2.9E+06
I-135	2.5E+05	4.4E+06	2.3E+07	2.9E+07	3.8E+07	4.1E+07	4.1E+07	4.1E+07
Xe-133	5.8E+05	5.6E+07	1.6E+09	3.5E+09	1.4E+10	1.3E+11	2.6E+11	5.8E+11
Xe-135	1.8E+05	1.7E+07	4.1E+08	7.6E+08	1.8E+09	3.1E+09	3.1E+09	3.1E+09
Cs-134	2.7E+04	4.3E+05	2.9E+06	4.3E+06	8.6E+06	2.5E+07	3.5E+07	9.0E+07
Cs-136	9.5E+03	1.5E+05	9.9E+05	1.5E+06	2.9E+06	7.9E+06	1.1E+07	1.9E+07
Cs-137	1.8E+04	2.8E+05	1.9E+06	2.8E+06	5.6E+06	1.6E+07	2.3E+07	5.9E+07
Ba-139	0.0E+00	1.2E+05	3.1E+05	3.2E+05	3.2E+05	3.2E+05	3.2E+05	3.2E+05
Ba-140	0.0E+00	2.8E+05	2.2E+06	3.3E+06	6.6E+06	1.8E+07	2.5E+07	4.5E+07
La-140	0.0E+00	4.3E+03	1.4E+05	3.0E+05	1.1E+06	8.5E+06	1.5E+07	3.7E+07
La-141	0.0E+00	2.0E+03	9.5E+03	1.1E+04	1.3E+04	1.3E+04	1.3E+04	1.3E+04
La-142	0.0E+00	1.2E+03	3.3E+03	3.4E+03	3.4E+03	3.4E+03	3.4E+03	3.4E+03
Ce-141	0.0E+00	6.7E+03	5.3E+04	7.9E+04	1.6E+05	4.7E+05	6.6E+05	1.5E+06
Ce-143	0.0E+00	6.1E+03	4.5E+04	6.5E+04	1.2E+05	2.2E+05	2.3E+05	2.3E+05
Ce-144	0.0E+00	5.5E+03	4.3E+04	6.5E+04	1.3E+05	3.9E+05	5.6E+05	1.5E+06
Pr-143	0.0E+00	2.5E+03	1.9E+04	2.9E+04	6.0E+04	1.8E+05	2.6E+05	4.9E+05
Nd-147	0.0E+00	1.1E+03	8.3E+03	1.3E+04	2.5E+04	6.9E+04	9.4E+04	1.6E+05
Np-239	0.0E+00	7.6E+04	5.7E+05	8.5E+05	1.6E+06	3.5E+06	4.0E+06	4.3E+06
Pu-238	0.0E+00	1.3E+01	1.1E+02	1.6E+02	3.2E+02	9.6E+02	1.4E+03	3.7E+03
Pu-239	0.0E+00	1.6E+00	1.3E+01	1.9E+01	3.9E+01	1.2E+02	1.7E+02	4.5E+02
Pu-240	0.0E+00	2.1E+00	1.6E+01	2.5E+01	5.0E+01	1.5E+02	2.2E+02	5.8E+02
Pu-241	0.0E+00	6.1E+02	4.8E+03	7.2E+03	1.5E+04	4.4E+04	6.3E+04	1.7E+05
Am-241	0.0E+00	2.8E-01	2.2E+00	3.3E+00	6.6E+00	2.0E+01	2.9E+01	8.4E+01
Cm-242	0.0E+00	6.4E+01	5.0E+02	7.6E+02	1.5E+03	4.6E+03	6.6E+03	1.7E+04
Cm-244	0.0E+00	3.1E+00	2.4E+01	3.7E+01	7.5E+01	2.2E+02	3.2E+02	8.6E+02

Table 15.4-8

LOCA Control Room Inventories (MBq)

Isotope	0.5 hr	2 hr	8 hr	24 hours	72 hours	4 days	7 days	30 days
Co-58	0.0E+00	1.5E-03	2.8E-03	1.7E-03	1.1E-03	4.9E-04	4.3E-04	2.2E-04
Co-60	0.0E+00	1.5E-03	2.7E-03	1.7E-03	1.1E-03	4.9E-04	4.4E-04	2.8E-04
Kr-85	1.8E+00	1.4E+02	1.3E+03	1.1E+03	2.0E+03	4.8E+03	6.1E+03	6.0E+03
Kr-85m	3.5E+01	2.1E+03	7.9E+03	3.9E+03	1.0E+03	3.6E-02	0.0E+00	0.0E+00
Kr-87	4.9E+01	1.6E+03	5.8E+02	6.9E+01	1.5E-01	8.4E-06	0.0E+00	0.0E+00
Kr-88	8.9E+01	4.6E+03	1.0E+04	3.6E+03	3.1E+02	9.1E-06	0.0E+00	0.0E+00
Rb-86	6.2E-03	7.2E-02	1.2E-01	7.0E-02	4.4E-02	1.7E-02	1.3E-02	3.5E-03
Sr-89	0.0E+00	2.4E+00	4.4E+00	2.7E+00	1.7E+00	7.6E-01	6.5E-01	3.0E-01
Sr-90	0.0E+00	2.4E-01	4.4E-01	2.7E-01	1.7E-01	7.9E-02	7.0E-02	4.5E-02
Sr-91	0.0E+00	2.6E+00	3.0E+00	1.4E+00	3.7E-01	8.9E-04	4.1E-06	0.0E+00
Sr-92	0.0E+00	1.8E+00	7.1E-01	1.7E-01	4.7E-03	0.0E+00	0.0E+00	0.0E+00
Y-90	0.0E+00	3.2E-03	3.0E-02	2.8E-02	3.8E-02	4.9E-02	4.4E-02	3.3E-02
Y-91	0.0E+00	3.1E-02	6.1E-02	3.9E-02	2.6E-02	1.2E-02	9.9E-03	4.8E-03
Y-92	0.0E+00	1.3E-01	1.0E+00	4.4E-01	4.5E-02	0.0E+00	0.0E+00	0.0E+00
Y-93	0.0E+00	3.2E-02	3.9E-02	1.8E-02	5.2E-03	9.8E-05	7.0E-05	4.7E-05
Zr-95	0.0E+00	4.4E-02	8.0E-02	4.9E-02	3.1E-02	1.4E-02	1.2E-02	5.9E-03
Zr-97	0.0E+00	4.1E-02	5.9E-02	3.1E-02	1.2E-02	2.8E-04	1.3E-05	0.0E+00
Nb-95	0.0E+00	4.4E-02	8.0E-02	4.9E-02	3.2E-02	1.4E-02	1.2E-02	6.8E-03
Mo-99	0.0E+00	5.6E-01	9.7E-01	5.7E-01	3.2E-01	6.9E-02	2.9E-02	5.6E-05
Tc-99m	0.0E+00	4.2E-01	8.3E-01	5.1E-01	3.1E-01	6.8E-02	2.2E-02	4.2E-05
Ru-103	0.0E+00	4.5E-01	8.3E-01	5.0E-01	3.2E-01	1.4E-01	1.2E-01	5.0E-02
Ru-105	0.0E+00	2.1E-01	1.5E-01	5.2E-02	4.9E-03	5.0E-02	3.5E-02	2.2E-04
Ru-106	0.0E+00	1.6E-01	2.9E-01	1.8E-01	1.1E-01	1.7E-02	1.4E-02	2.8E-02
Rh-105	0.0E+00	2.7E-01	4.7E-01	2.7E-01	1.4E-01	5.2E-04	7.6E-04	0.0E+00
Sb-127	0.0E+00	6.2E-01	1.1E+00	6.4E-01	3.8E-01	1.0E-01	5.2E-02	5.3E-04
Sb-129	0.0E+00	1.3E+00	9.2E-01	3.1E-01	2.8E-02	9.8E-07	0.0E+00	0.0E+00
Te-127	0.0E+00	5.6E-01	1.1E+00	6.4E-01	4.0E-01	1.2E-01	5.6E-02	1.1E-02
Te-127m	0.0E+00	8.3E-02	1.5E-01	9.3E-02	6.0E-02	2.7E-02	2.4E-02	1.3E-02
Te-129	0.0E+00	7.5E-01	1.2E+00	5.0E-01	2.0E-01	7.1E-02	4.6E-02	1.8E-02
Te-129m	0.0E+00	2.8E-01	5.1E-01	3.1E-01	2.0E-01	8.6E-02	7.2E-02	2.8E-02
Te-131m	0.0E+00	8.2E-01	1.3E+00	7.3E-01	3.5E-01	3.1E-02	5.2E-03	0.0E+00
Te-132	0.0E+00	8.4E+00	1.5E+01	8.6E+00	5.0E+00	1.2E+00	5.7E-01	2.7E-03
I-131	2.7E+00	3.8E+01	6.5E+01	3.9E+01	2.5E+01	1.2E+01	8.9E+00	9.7E-01
I-132	3.1E+00	3.2E+01	2.3E+01	1.0E+01	5.8E+00	1.4E+00	5.1E-01	2.4E-03

Table 15.4-8

LOCA Control Room Inventories (MBq)

Isotope	0.5 hr	2 hr	8 hr	24 hours	72 hours	4 days	7 days	30 days
I-133	5.4E+00	7.4E+01	1.0E+02	5.6E+01	2.5E+01	1.4E+00	1.2E-01	0.0E+00
I-134	3.2E+00	1.4E+01	2.1E-01	7.0E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00
I-135	4.8E+00	5.8E+01	5.4E+01	2.2E+01	4.1E+00	1.3E-03	4.0E-07	0.0E+00
Xe-133	3.0E+02	2.2E+04	2.0E+05	1.7E+05	2.8E+05	4.7E+05	4.0E+05	1.9E+04
Xe-135	9.4E+01	6.5E+03	4.3E+04	2.9E+04	2.0E+04	2.0E+02	1.0E+00	0.0E+00
Cs-134	5.2E-01	6.1E+00	1.0E+01	6.0E+00	3.8E+00	1.7E+00	1.4E+00	8.8E-01
Cs-136	1.8E-01	2.1E+00	3.4E+00	2.0E+00	1.3E+00	4.7E-01	3.5E-01	6.5E-02
Cs-137	3.4E-01	4.0E+00	6.5E+00	3.9E+00	2.5E+00	1.1E+00	9.3E-01	5.9E-01
Ba-139	0.0E+00	1.4E+00	1.2E-01	1.2E-02	1.5E-05	1.1E+00	6.5E-01	1.2E-01
Ba-140	0.0E+00	4.3E+00	7.7E+00	4.7E+00	2.9E+00	4.3E-02	2.2E-01	4.2E-02
La-140	0.0E+00	6.4E-02	8.0E-01	7.5E-01	9.7E-01	9.7E-01	6.9E-01	1.3E-01
La-141	0.0E+00	2.7E-02	1.7E-02	5.5E-03	4.0E-04	0.0E+00	0.0E+00	0.0E+00
La-142	0.0E+00	1.4E-02	1.7E-03	2.0E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Ce-141	0.0E+00	1.0E-01	1.9E-01	1.1E-01	7.2E-02	3.1E-02	2.6E-02	1.0E-02
Ce-143	0.0E+00	9.0E-02	1.5E-01	8.3E-02	4.1E-02	4.2E-03	8.2E-04	0.0E+00
Ce-144	0.0E+00	8.3E-02	1.5E-01	9.2E-02	6.0E-02	2.7E-02	2.4E-02	1.4E-02
Pr-143	0.0E+00	3.7E-02	6.9E-02	4.2E-02	2.8E-02	1.2E-02	9.2E-03	1.8E-03
Nd-147	0.0E+00	1.6E-02	2.9E-02	1.8E-02	1.1E-02	4.2E-03	3.1E-03	4.6E-04
Np-239	0.0E+00	1.1E+00	1.9E+00	1.1E+00	6.3E-01	1.2E-01	4.4E-02	3.2E-05
Pu-238	0.0E+00	2.0E-04	3.7E-04	2.3E-04	1.5E-04	6.7E-05	6.0E-05	3.8E-05
Pu-239	0.0E+00	2.4E-05	4.5E-05	2.7E-05	1.8E-05	8.2E-06	7.3E-06	4.7E-06
Pu-240	0.0E+00	3.2E-05	5.8E-05	3.5E-05	2.3E-05	1.1E-05	9.4E-06	6.0E-06
Pu-241	0.0E+00	9.2E-03	1.7E-02	1.0E-02	6.6E-03	3.0E-03	2.7E-03	1.7E-03
Am-241	0.0E+00	4.1E-06	7.6E-06	4.7E-06	3.0E-06	1.4E-06	1.3E-06	9.5E-07
Cm-242	0.0E+00	9.7E-04	1.8E-03	1.1E-03	7.0E-04	3.2E-04	2.8E-04	1.6E-04
Cm-244	0.0E+00	4.7E-05	8.7E-05	5.3E-05	3.4E-05	1.6E-05	1.4E-05	8.8E-06

Table 15.4-9

LOCA Inside Containment Analysis Results

Exposure Location	**Meteorology (s/m^3)**	**Maximum Calculated TEDE** (Sv)	**10 CFR 50.34(a)(1) Acceptance Criterion TEDE** (Sv)
Exclusion Area Boundary (EAB)	2.00E-03	0.130	0.25
Outer boundary of Low Population Zone (LPZ)	1.9E-04 (0-8 h*) 1.4E-04 (8-24 h*) 7.5E-05 (1-4 d*) 3.0E-05 (4-30 d*)	0.032 0.059 0.111 0.177	0.25
Control Room	Table 2.0-1	0.041	0.05

* The values listed do not account for the additional 20 minutes of decay, therefore the time listed correspond to time after fuel damage (not from the onset of the event).

Table 15.4-10

Sequence of Events for Main Steamline Break Accident (MSLBA) Outside Containment

Time (s)	**Event**
0	Guillotine break of one main steam line outside containment.
0.5	High steamline flow signal initiates closure of MSIVs
< 1.0	Reactor begins scram.
< 2.0	Partial closure (15%) of MSIV initiates isolation condensers.
< 5.5	MSIVs fully closed.
10	Reactor low water Level 2 is reached. Isolation condensers receive second initiation signal.
32	Isolation condensers in full operation. Water level stabilized.
435	SRVs open on high vessel pressure (if isolation condensers are not available). The SRVs open and close to maintain vessel absolute pressure.
3540	Reactor low water Level 1 is reached (if isolation condensers are not available). ADS timer initiated.
3550	ADS timer timed out. ADS actuation sequence initiated. GDCS timer initiated.
3700	GDCS timer timed out. GDCS injection valves open.
3880	Vessel pressure decreases below shutoff head of GDCS. GDCS reflooding flow into the vessel begins.

* The core remains covered throughout the transient and no core heatup occurs.

Table 15.4-11

MSLBA Parameters

1.	Data and assumptions used to estimate source terms	
	A. Fuel Damage	none
	B. Reactor Coolant Activity: Pre-incident Spike Equilibrium Iodine Activity	 4.0 μCi/g DE I-131 0.2 μCi/g DE I-131
	C. Steam Mass Released, kg	4,705
	D. Water Mass Released, kg	82,328
2.	Data and assumptions used to estimate activity released	
	A. Isolation valve closure time, sec	5
	B. MSIV Response time, sec	0.5
3.	Dispersion Data	
	A. Off-site Meteorology	2.00E-03 s/m^3
	B. Method of Dose Calculation	RG 1.183
	C. Dose Conversion Assumptions	RG 1.183
	D. Activity Inventory and Releases	Tables 15.4-12
	E. Dose Evaluations	Table 15.4-13

Table 15.4-12
MSLBA Environment Releases

Isotope	Equilibrium Activity MBq	Iodine Spike Activity MBq	Isotope	Equilibrium Activity MBq	Iodine Spike Activity MBq
Co-58	1.4E+03	1.4E+03	Te-131m	1.3E+03	1.3E+03
Co-60	2.7E+03	2.7E+03	Te-132	1.4E+02	1.4E+02
Kr-85	1.7E+00	1.7E+00	I-131	2.4E+05	4.9E+06
Kr-85m	4.4E+02	4.4E+02	I-132	2.3E+06	4.6E+07
Kr-87	1.4E+03	1.4E+03	I-133	1.7E+06	3.4E+07
Kr-88	1.4E+03	1.4E+03	I-134	4.2E+06	8.5E+07
Rb-86	0.0E+00	0.0E+00	I-135	2.4E+06	4.7E+07
Sr-89	1.4E+03	1.4E+03	Xe-133	5.9E+02	5.9E+02
Sr-90	9.4E+01	9.4E+01	Xe-135	1.6E+03	1.6E+03
Sr-91	5.2E+04	5.2E+04	Cs-134	3.7E+02	3.7E+02
Sr-92	1.2E+05	1.2E+05	Cs-136	2.4E+02	2.4E+02
Y-90	9.4E+01	9.4E+01	Cs-137	9.7E+02	9.7E+02
Y-91	5.5E+02	5.5E+02	Ba-139	0.0E+00	0.0E+00
Y-92	7.6E+04	7.6E+04	Ba-140	5.5E+03	5.5E+03
Y-93	5.2E+04	5.2E+04	La-140	5.5E+03	5.5E+03
Zr-95	1.1E+02	1.1E+02	La-141	0.0E+00	0.0E+00
Zr-97	0.0E+00	0.0E+00	La-142	0.0E+00	0.0E+00
Nb-95	1.1E+02	1.1E+02	Ce-141	4.0E+02	4.0E+02
Mo-99	2.7E+04	2.7E+04	Ce-143	0.0E+00	0.0E+00
Tc-99m	2.7E+04	2.7E+04	Ce-144	4.0E+01	4.0E+01
Ru-103	2.7E+02	2.7E+02	Pr-143	0.0E+00	0.0E+00
Ru-105	0.0E+00	0.0E+00	Nd-147	0.0E+00	0.0E+00
Ru-106	4.0E+01	4.0E+01	Np-239	1.1E+05	1.1E+05
Rh-105	0.0E+00	0.0E+00	Pu-238	0.0E+00	0.0E+00
Sb-127	0.0E+00	0.0E+00	Pu-239	0.0E+00	0.0E+00
Sb-129	0.0E+00	0.0E+00	Pu-240	0.0E+00	0.0E+00
Te-127	0.0E+00	0.0E+00	Pu-241	0.0E+00	0.0E+00
Te-127m	0.0E+00	0.0E+00	Am-241	0.0E+00	0.0E+00
Te-129	0.0E+00	0.0E+00	Cm-242	0.0E+00	0.0E+00
Te-129m	5.5E+02	5.5E+02	Cm-244	0.0E+00	0.0E+00

Table 15.4-13

MSLBA Analysis Results

Exposure Location and Time Period/Duration	Maximum Calculated **TEDE** (rem)	Acceptance Criterion **TEDE** (rem)
Exclusion Area Boundary (EAB) for the Entire Period of the Radioactive Cloud Passage		
Pre-incident Spike	12.6	25
Equilibrium Iodine Activity	0.7	2.5
Outer Boundary of Low Population Zone (LPZ) for the Entire Period of the Radioactive Cloud Passage		
Pre-incident Spike	12.6	25
Equilibrium Iodine Activity	0.7	2.5
Control Room Dose for the Duration of the Accident	4.5	5

Table 15.4-14

Feedwater Line Break Accident Parameters

I. Data and Assumptions Used to Estimate Source Terms		
A.	Total mass of coolant released, kg (lb)	259,654 (571,925)
B.	% of released coolant flashed to steam	22
C.	Demineralizer efficiency, %	99
II. Data and Assumptions Used to Estimate Activity Released		
A.	Iodine water concentration	0.2 μCi/g DE I-131
B.	Iodine plateout fraction	0
C.	Building release rate	Instantaneous
III. Dispersion and Dose Data		
A.	Meteorology	1.00E-03 s/m^3
B.	Method of dose calculation	Reference 15.4-1
C.	Dose conversion assumptions	Reference 15.4-1, FGR-11
D.	Activity inventory/releases	Table 15.4-15
E.	Dose evaluations	Table 15.4-16

Table 15.4-15
Feedwater Line Break Accident
Environment Releases

Isotope	Activity (MBq)
I-131	1.3E+02
I-132	1.2E+03
I-133	8.7E+02
I-134	2.2E+03
I-135	1.2E+03

Table 15.4-16

Feedwater Line Break Analysis Results

Exposure Location and Time Period/Duration	**Maximum Calculated TEDE** (rem)	**Acceptance Criterion TEDE** (rem)
Exclusion Area Boundary (EAB) for the Entire Period of the Radioactive Cloud Passage	1.7E-04	2.5
Outer Boundary of Low Population Zone (LPZ) for the Entire Period of the Radioactive Cloud Passage	1.7E-04	2.5

Table 15.4-17

Instrument Line Break Accident Parameters

I. Data and assumptions used to estimate source terms	
A. Power level, MWt	4590
B. Mass of fluid released, kg (lbm)	14,785 (32,565)
C. Mass of fluid flashed to steam, kg (lbm)	4,007 (8,825)
D. Duration of accident, hr	6
E. Number of bundles in core	1132
II. Data and assumptions used to estimate activity released	
A. Iodine water concentration	4.0 μCi/g DE I-131
B. Iodine plateout fraction, %	0
C. Reactor Building Flow rate, %/hour	500
III Dispersion and Dose Data	
A. Meteorology	Table 15.4-9 (Reactor Building)
B. Method of Dose Calculation	Ref 15.4-1
C. Dose conversion Assumptions	RG 1.183 and Ref. 15.4-1
D. Activity Inventory/releases	Table 15.4-18
E. Dose evaluations	Table 15.4-19

Table 15.4-18

Instrument Line Break Accident Isotopic Inventory (MBq)

Time (hr)	0.02	0.17	0.5	1	2	4	8	12
Co-58	1.2E-03	8.7E-02	7.4E-01	2.6E+00	8.2E+00	2.4E+01	4.8E+01	5.9E+01
Co-60	2.4E-03	1.7E-01	1.5E+00	5.1E+00	1.6E+01	4.8E+01	9.6E+01	1.2E+02
Rb-86	5.2E-01	3.7E+01	3.1E+02	1.1E+03	3.5E+03	1.0E+04	2.0E+04	2.5E+04
Sr-89	1.2E-03	8.7E-02	7.4E-01	2.6E+00	8.2E+00	2.4E+01	4.8E+01	5.9E+01
Sr-90	8.4E-05	6.0E-03	5.1E-02	1.8E-01	5.6E-01	1.7E+00	3.3E+00	4.0E+00
Sr-91	4.6E-02	3.3E+00	2.8E+01	9.7E+01	3.1E+02	9.2E+02	1.8E+03	2.2E+03
Sr-92	1.1E-01	7.9E+00	6.7E+01	2.3E+02	7.5E+02	2.2E+03	4.4E+03	5.3E+03
Y-90	8.4E-05	6.0E-03	5.1E-02	1.8E-01	5.6E-01	1.7E+00	3.3E+00	4.0E+00
Y-91	4.9E-04	3.5E-02	3.0E-01	1.0E+00	3.3E+00	9.7E+00	1.9E+01	2.3E+01
Y-92	6.8E-02	4.8E+00	4.1E+01	1.4E+02	4.6E+02	1.4E+03	2.7E+03	3.3E+03
Y-93	4.6E-02	3.3E+00	2.8E+01	9.7E+01	3.1E+02	9.2E+02	1.8E+03	2.2E+03
Zr-95	9.8E-05	7.0E-03	5.9E-02	2.0E-01	6.6E-01	1.9E+00	3.9E+00	4.7E+00
Nb-95	9.8E-05	7.0E-03	5.9E-02	2.0E-01	6.6E-01	1.9E+00	3.9E+00	4.7E+00
Mo-99	2.4E-02	1.7E+00	1.5E+01	5.1E+01	1.6E+02	4.8E+02	9.6E+02	1.2E+03
Tc-99m	2.4E-02	1.7E+00	1.5E+01	5.1E+01	1.6E+02	4.8E+02	9.6E+02	1.2E+03
Ru-103	2.4E-04	1.7E-02	1.5E-01	5.1E-01	1.6E+00	4.8E+00	9.6E+00	1.2E+01
Ru-106	3.5E-05	2.5E-03	2.1E-02	7.4E-02	2.4E-01	7.0E-01	1.4E+00	1.7E+00
Te-129m	4.9E-04	3.5E-02	3.0E-01	1.0E+00	3.3E+00	9.7E+00	1.9E+01	2.3E+01
Te-131m	1.2E-03	8.5E-02	7.2E-01	2.5E+00	8.0E+00	2.4E+01	4.7E+01	5.7E+01
Te-132	1.2E-04	8.7E-03	7.4E-02	2.6E-01	8.2E-01	2.4E+00	4.8E+00	5.9E+00
I-131	4.3E+00	3.1E+02	2.6E+03	9.1E+03	2.9E+04	8.6E+04	1.7E+05	2.1E+05
I-132	4.1E+01	2.9E+03	2.5E+04	8.6E+04	2.8E+05	8.2E+05	1.6E+06	2.0E+06
I-133	3.0E+01	2.1E+03	1.8E+04	6.3E+04	2.0E+05	6.0E+05	1.2E+06	1.4E+06
I-134	7.5E+01	5.4E+03	4.5E+04	1.6E+05	5.1E+05	1.5E+06	3.0E+06	3.6E+06
I-135	4.2E+01	3.0E+03	2.6E+04	8.8E+04	2.8E+05	8.4E+05	1.7E+06	2.0E+06
Cs-134	3.3E-04	2.3E-02	2.0E-01	6.8E-01	2.2E+00	6.5E+00	1.3E+01	1.6E+01
Cs-136	2.2E-04	1.6E-02	1.3E-01	4.6E-01	1.5E+00	4.3E+00	8.6E+00	1.0E+01
Cs-137	8.7E-04	6.2E-02	5.2E-01	1.8E+00	5.8E+00	1.7E+01	3.4E+01	4.2E+01
Ba-140	4.9E-03	3.5E-01	3.0E+00	1.0E+01	3.3E+01	9.7E+01	1.9E+02	2.3E+02
La-140	4.9E-03	3.5E-01	3.0E+00	1.0E+01	3.3E+01	9.7E+01	1.9E+02	2.3E+02
Ce-141	3.5E-04	2.5E-02	2.1E-01	7.4E-01	2.4E+00	7.0E+00	1.4E+01	1.7E+01
Ce-144	3.5E-05	2.5E-03	2.1E-02	7.4E-02	2.4E-01	7.0E-01	1.4E+00	1.7E+00
Np-239	9.8E-02	7.0E+00	5.9E+01	2.0E+02	6.6E+02	1.9E+03	3.9E+03	4.7E+03

Table 15.4-19

Instrument Line Break Accident Results

Exposure Location and Time Period/Duration	**Maximum Calculated TEDE** (rem)	**Acceptance Criterion TEDE** (rem)
Exclusion Area Boundary (EAB) for the Entire Period of the Radioactive Cloud Passage	0.15	2.5
Outer Boundary of Low Population Zone (LPZ) for the Entire Period of the Radioactive Cloud Passage		2.5
8 hours (0 – 8 hr X/Q = 1.90E-04 sec/m^3)	0.04	
1 day (8 – 24 hr X/Q = 1.40E-04 sec/m^3)	0.05	
4 days (24 – 96 hr X/Q = 7.50E-05 sec/m^3)	0.05	
30 days (96 – 720 hr X/Q = 3.00E-05 sec/m^3)	0.05	
Main Control Room	0.2	5.0

Table 15.4-20

RWCU/SDC System Line Failure Outside Containment Sequence of Events

Sequence of Events	Time (s)
Clean up water line break occurs	0
Check valves on clean up water line to feedwater line isolate. Differential pressure instrumentation initiates delay sequence	0+
Differential pressure instrumentation actuates isolation valves	46
Isolation valves complete closure and isolation	66
Normal reactor shutdown and cooldown procedure	1-2 hour

* Core remains covered throughout the transient and no core heatup occurs.

Table 15.4-21

RWCU/SDC Line Break Accident Parameters

I. Data and assumptions used to estimate source terms	
A. Fuel Damage	none
B. Reactor Coolant Activity: Pre-incident Spike Equilibrium Iodine Activity	 4.0 µCi/g DE I-131 0.2 µCi/g DE I-131
C. Water Mass Released, kg (lbm) RPV Coolant Blow-down RWCU/SDC System RHX RWCU/SDC System NRHX	 128,650 (283,620) 975 (2150) 3651 (8050)
II. Data and assumptions used to estimate activity released	
A. Water-to-Steam Flashing Fractions RPV Coolant Blow-down RWCU/SDC System RHX RWCU/SDC System NRHX	 0.38 0.28 0.074
B. Iodine Plateout Fraction, %	0
C. Reactor Building Flow rate, %/hour	Instantaneous
III Dispersion and Dose Data	
A. Meteorology EAB LPZ Control Room Reactor Building Release	 2.00E-03 s/m^3 Table 2.0-1 Table 2.0-1
B. Method of Dose Calculation	RG 1.183
C. Dose conversion Assumptions	RG 1.183
D. Activity Inventory/Releases	Table 15.4-22
E. Dose Evaluations	Table 15.4-23

Table 15.4-22
RWCU/SDS Line Break Accident Isotopic Release to Environment

Isotope	Coincident Spike (MBq)	Pre-incident Spike (MBq)
I-131	1.46E+05	2.92E+06
I-132	1.38E+06	2.77E+07
I-133	1.01E+06	2.02E+07
I-134	2.54E+06	5.09E+07
I-135	1.42E+06	2.84E+07
Cs-134	1.68E+03	3.37E+04
Cs-136	1.12E+03	2.24E+04
Cs-137	4.49E+03	8.97E+04
Co-58	8.40E+02	8.40E+02
Co-60	1.63E+03	1.63E+03
Sr-89	6.36E+03	1.27E+05
Sr-90	4.49E+02	8.97E+03
Y-90	4.49E+02	8.97E+03
Sr-91	2.39E+05	4.79E+06
Sr-92	5.61E+05	1.12E+07
Y-91	2.47E+03	4.94E+04
Y-92	3.48E+05	6.95E+06
Y-93	2.39E+05	4.79E+06
Zr-95	4.86E+02	9.72E+03
Nb-95	4.86E+02	9.72E+03
Mo-99	1.23E+05	2.47E+06
Tc-99m	1.23E+05	2.47E+06
Ru-103	1.23E+03	2.47E+04
Ru-106	1.87E+02	3.74E+03
Te-129m	2.47E+03	4.94E+04
Te-131m	5.98E+03	1.20E+05
Te-132	5.98E+02	1.20E+04
Ba-140	2.47E+04	4.94E+05
La-140	2.47E+04	4.94E+05
Ce141	1.87E+03	3.74E+04
Ce-144	1.87E+02	3.74E+03
Np-239	4.86E+05	9.72E+06

Table 15.4-23

RWCU/SDC Line Break Accident Results

Exposure Location and Time Period/Duration	**Maximum Calculated TEDE** (rem)	**Acceptance Criterion TEDE** (rem)
Exclusion Area Boundary (EAB) for the Entire Period of the Radioactive Cloud Passage		
Coincident Iodine Spike Case	0.49	2.5
Pre-incident Iodine Spike Case	9.8	25
Outer Boundary of Low Population Zone (LPZ) for the Entire Period of the Radioactive Cloud Passage		
Coincident Iodine Spike Case	0.047	2.5
Pre-incident Iodine Spike Case	0.93	25
Control Room Dose for the Duration of the Accident		
Coincident Iodine Spike Case	0.24	5
Pre-incident Iodine Spike Case	4.7	5

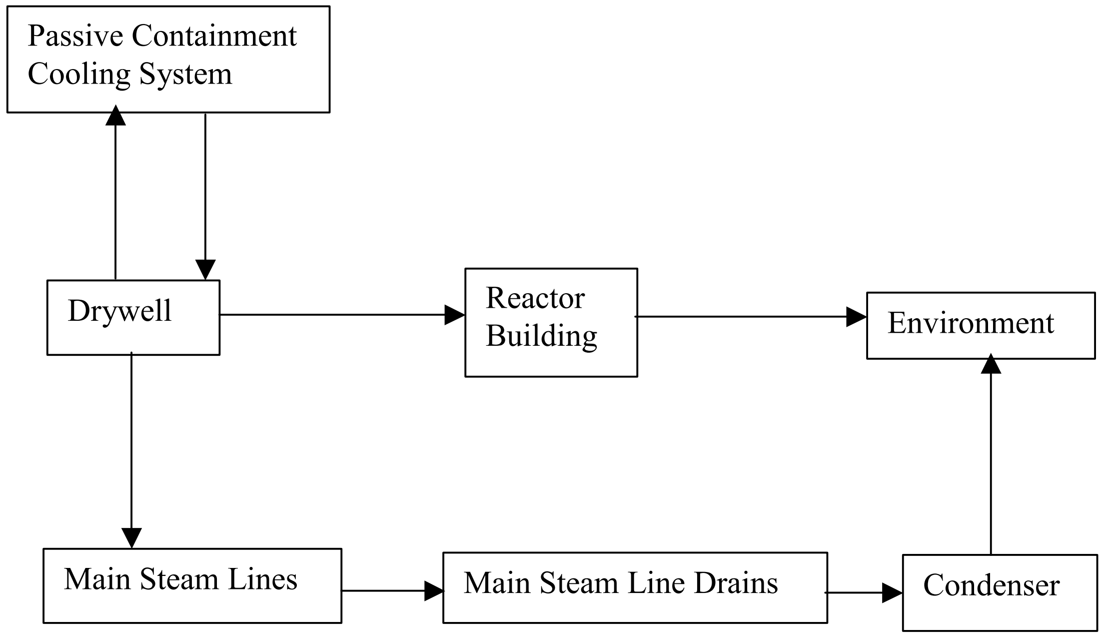


Figure 15.4-1. LOCA Radiological Paths

15.5 SPECIAL EVENT EVALUATIONS

15.5.1 Overpressure Protection

Results of the overpressure protection evaluation are provided in Subsection 5.2.2.

15.5.2 Shutdown Without Control Rods (Standby Liquid Control System Capability)

Reactor shutdown without control rods is an event requiring an alternate method of reactivity control via the Standby Liquid Control (SLC) system. The safety evaluation of SLC system capability is described in Subsection 9.3.5.3.

15.5.3 Shutdown from Outside Main Control Room

Reactor shutdown from outside the main control room is an event investigated to evaluate the capability of the plant to be safely shutdown and cooled to the cold shutdown state from outside the main control room. The evaluation is described in Subsection 7.4.2.

15.5.4 Anticipated Transients Without Scram

15.5.4.1 Requirements

NUREG-0800 Standard Review Plan (SRP) 15.8 requires the BWR to have an automatic recirculation pump trip (RPT) and emergency procedures for ATWS. This SRP has been superseded by the issuance of 10 CFR 50.62, which requires the BWR to have automatic RPT (not applicable to an ESBWR), an Alternate Rod Insertion (ARI) system, and an automatic SLC system. The SLC system is required to have a minimum flow capacity and boron content equivalent to 5.42x10-3 m^3/s (86 gpm) of 13 weight-percent sodium pentaborate solution.

15.5.4.2 Plant Capabilities

For ATWS prevention/mitigation, the ESBWR provides the following:

- An ARI system that utilizes sensors and logic that are diverse and independent of the RPS;
- Electrical insertion of Fine Motion Control Rod Drives (FMCRDs) that also utilize sensors and logic that are diverse and independent of the RPS;
- Automatic feedwater runback under conditions indicative of an ATWS; and
- Automatic initiation of SLC under conditions indicative of an ATWS.

The ATWS rule of 10 CFR 50.62 was written as hardware-specific, rather than functionally, because it clearly reflected the BWR use of forced core flow circulation. Because the ESBWR uses natural circulation, there are no recirculation pumps to be tripped. Hence, no RPT logic can be implemented in the ESBWR. An ATWS automatic feedwater runback feature is implemented, to provide a reduction in water level, core flow and reactor power, similar to RPT in a forced circulation plant. This feature prevents reactor vessel overpressure and possible short-term fuel damage for the most limiting ATWS events.

The ATWS rule of 10 CFR 50.62 is also specific to the use of locking-piston control rod drives. The ESBWR, however, uses the FMCRD design with both hydraulic and electrical means to achieve shutdown. This drive system is described in detail in Section 4.6. The use of this design eliminates the common mode failure potentials of the existing locking-piston CRD by eliminating the scram discharge volume (potential mechanical common mode failure) and by having an electric motor run-in diverse from the hydraulic scram feature. This latter feature allows rod run-in, if scram air header pressure is not exhausted because of a postulated common mode electrical failure and simultaneous failure of the ARI system, and thus satisfies the intent of 10 CFR 50.62. Therefore, the ESBWR design can respond to an ATWS threatening event independent of the SLC system.

The SLC system is required by 10 CFR 50.62(c)(4), and is described within Section 9.3. Because the new drive design eliminates the previous common-mode failure potential and because of the very low probability of simultaneous common mode failure of a large number of FMCRDs, a failure to achieve shutdown is deemed incredible. However, automatic initiation of the SLC system under conditions indicative of an ATWS is also incorporated in order to meet the rule specified in 10 CFR 50.62.

15.5.4.3 Performance Evaluation

15.5.4.3.1 Introduction

Typical ATWS events are analyzed to confirm the design for ESBWR.

The procedure and assumptions used in this analysis are documented in Reference 15.5-2.

All transient analyses, unless otherwise specified, were performed with the TRACG code.

15.5.4.3.2 Performance Requirements

As identified in Reference 15.5-1, the design meets the following requirements:

Fuel Integrity - The long-term core cooling capability is assured by meeting the cladding temperature and oxidation criteria of 10 CFR 50.46 (i.e., peak cladding temperature (PCT) not exceeding 1204°C (2200°F), and the local oxidation of the cladding not exceeding 17% of the total cladding thickness).

Containment Integrity - The long-term containment capability is maintained. The maximum containment pressure does not exceed the design pressure of the containment structure, 414 kPa (60 psia). The suppression pool temperature is limited to the wetwell design temperature of 121°C (250°F).

Primary System - The system transient pressure is limited to 10.34 MPag (1500 psig) such that the maximum primary stress within the reactor coolant pressure boundary (RCPB) does not exceed the emergency limits as defined in the ASME Code, Section III.

Long-Term Shutdown Cooling - Subsequent to an ATWS event, the reactor is brought to a safe shutdown condition, and be cooled down and maintained in a cold shutdown condition.

These performance requirements are summarized in Table 15.5-1.

15.5.4.3.3 Analysis Conditions

The probability of the occurrence of an ATWS is low. Thus, historically nominal parameters and initial conditions have been used in these analyses as specified in Reference 15.5-1.

As the processes for definition of allowable operational flexibility and margin improvement options expanded the analysis process transitioned to a basis that required use of bounding initial conditions. This was done because the frequency of operation within the allowable optional configurations could not be defined. In other words, “nominal” could not be defined. Some initial conditions, the most important being reactor power, are still analyzed without consideration of instrument uncertainties. Those that are applied conservatively include core exposure, core axial power shape, and Safety Relief Valve operability. All events analyzed assume reduced IC heat removal capacity to add a further measure of conservatism. The peak containment pressure presented is estimated in a conservative manner assuming that all the non-condensable gas from the drywell is in the wetwell airspace at the time of the peak pool temperature.

Selected inputs that affect the critical safety parameters are set to bounding values. The most important parameters for peak vessel pressure are Safety Relief Valve capacities and setpoints. These inputs are set to analytical limits. The most important parameters for clad and suppression pool temperature are initial Critical Power Ratio and boron flow rate respectively. These inputs are set to analytical limits.

Tables 15.5-2 and 15.5-3 list the initial conditions and equipment performance characteristics, which are used in the analysis.

15.5.4.3.4 ATWS Logic and Setpoints

The mitigation of ATWS events is accomplished by a multitude of equipment and procedures. These include ARI, FMCRD run-in, feedwater runback, ADS inhibit, and SLC. The ATWS mitigation logic is presented in Section 7.8, Figure 7.8-2 and 7.8-3 and Subsections 7.7.2 and 7.3.4. The following are the initiation signals and setpoints for the above response:

- ARI and FMCRD run-in
  - High pressure, or
  - Level 2, or
  - Either RPS scram command, or SCRRI/SRI command and elevated power levels exist after time delay
  - Manual
- SLC system initiation
  - High pressure and Startup Range Neutron Monitor (SRNM) ATWS permissive for three minutes, or
  - Level 2 and SRNM ATWS permissive three minutes, or
  - Manual ARI/FMCRD run-in signals and SRNM ATWS permissive for three minutes
- Feedwater runback

- High pressure and SRNM ATWS permissive, or
- Either RPS scram command, or SCRRI/SRI command and elevated power levels persist after time delay
- Manual ARI/FMCRD run-in

- ADS inhibit
  - High pressure and Average Power Range Monitoring (APRM) not downscale for one minute, or
  - Level 2 and APRM not downscale
- HP_CRD
  - Level 2 with maximum 10 second delay
  - Level 2 with maximum delay of 145 seconds during loss of off-site power
- IC
  - Closure of MSIV
  - High pressure for 10 seconds
  - Level 2 with 30 second delay or Level 1

15.5.4.3.5 Selection of Events

Based on conclusions from the evaluations for operating BWR plants as documented in Reference 15.5-1, events were selected to demonstrate the performance of the ATWS capabilities. The events are grouped into three categories. The first category includes events that demonstrate ATWS mitigation on the most severe and limiting cases. The second category has events that are generally less severe for ATWS analysis but are analyzed to show the sensitivity of key ATWS parameters to these events. In each of the above cases, ATWS mitigation actions are assumed to occur on the appropriate signals. No operator action is assumed, unless specifically mentioned. The third category covers the cases that have only minor effect on the reactor vessel containment. They are discussed briefly to support the assumption that they do not significantly influence the design of ATWS mitigation. No analysis was performed for events in the third category.

Category 1: Limiting Events

- Main Steamline Isolation Valve (MSIV) Closure - Generic studies have shown that this transient produces high neutron flux, vessel pressure, and suppression pool temperature. The maximum values from this event are, in most cases, bounding of all events considered.
- Loss of Condenser Vacuum - The turbine trips on low condenser vacuum. The bypass valves are available for a short period, and then close on loss of condenser vacuum. Depending on detailed BOP performance the pressurization rate and the energy addition to the pool may be as severe as MSIV closure. This event is included in category I to assure the short-term peak vessel pressure and clad temperature remain within limits.

- Loss of Feedwater Heating - In ESBWR this event is mitigated with Select Control Rod Run-In/ Select Rod Insert (SCRRI/SRI). Consistent with ATWS failure to scram, this event is evaluated with no SCRRI/SRI. This event is included in category I, to determine whether it is limiting for peak clad temperature. Because the turbine bypass valves are available, it is not limiting for vessel pressure or suppression pool temperature.

Category 2: Moderate Events

- Loss of Normal AC Power to Station Auxiliaries - This transient is less severe than the MSIV closure in terms of vessel pressure, neutron flux, and suppression pool temperature. However, because of the loss of AC power, the availability of equipment is different. Therefore, the plant capability of mitigating this event is evaluated.
- Loss of Feedwater Flow - This transient is less severe than the above events. However, it is the only event where the ATWS trip is initiated from the low level signals. Thus, this event is analyzed to show that the low level trips are capable to mitigate the event.
- Generator Load Rejection with a Single Failure in the Turbine Bypass System – In this transient, because half of the bypass valves are available, the pressurization rate is less severe than MSIV closure, the FW temperature change is similar to MSIV closure and the energy addition to the pool is less severe than MSIV closure.

Category 3: Minimum Effect Events

- Inadvertent Isolation Condenser Initiation - Spurious initiation of the isolation condensers would cause a moderator temperature decrease and a slow insertion of positive reactivity into the core. During power operation the system settles at a new steady state.
- Turbine Trip with Full Bypass – In this transient, because full bypass capacity is available, the pressurization rate is less severe than MSIV closure, the FW temperature change is similar to MSIV closure and the energy addition to the pool is less severe than MSIV closure.
- Opening of One Control or Turbine Bypass Valve – This event assumes a hydraulic system failure that causes a mild decrease in pressure, which is compensated for by the control system closing other valves. The ATWS response is not limiting.

15.5.4.3.6 Transient Responses

Main Steam Isolation Valve Closure

Three cases are analyzed for the limiting transient-MSIV closure. The first two are without SLC system injection. The third one is a bounding case with SLC system injection. The bounding case is analyzed to show the in-depth ATWS mitigation capability of the ESBWR.

(1) MSIV closure with scram failure and ARI. This case is intended to show the effectiveness of the ARI design.

(2) MSIV closure with scram failure and FMCRD run-in, assuming a total failure of hydraulic rod insertion (scram and ARI), was performed to show the backup capability of FMCRD run-in.

(3) MSIV closure with scram, ARI and FMCRD run-in failure, bounding case. Shows the ATWS performance with input parameters set per Reference 15.5-2 to produce a conservatively high reactor pressure, peak clad temperature, and a conservatively high suppression pool temperature. This case is intended to show that the peak RPV pressure, peak clad temperature, peak suppression pool temperature and the wetwell pressures are below the acceptance criteria. In this case, both ARI and FMCRD run-in are assumed to fail. Automatic boron injection with a total delay of 191-seconds (180 second timer + 10 second boron transportation delay in the SLC system + 1 second sensor and logic delay in the DCIS system) is relied upon to mitigate the transient event.

    The bounding case is composed of five major elements that are intended to conservatively bound the key ATWS safety parameters for ESBWR. First, the reactor power used in the bounding analysis is 102% of the normal operating value. Second, the feedwater enthalpy is conservatively chosen to be 105% of the nominal value. Third, the SRV capacity input, shown in Table 15.5-3, chosen for the analysis is set to be conservatively bounding for the vessel bottom pressure response. Fourth, the analysis value of Feedwater Runback (FWRB) coastdown time of 15 seconds with an additional delay in the analysis of 10 seconds for the FWRB activation, is chosen to conservatively bound the peak suppression pool temperature. Fifth, the initial Minimum Critical Power Ratio (MCPR) of the hot bundle is set to a value of 1.16 to conservatively bound the Peak Cladding Temperature (PCT), and which is conservatively lower than the nominal value of 1.3.

If the ARI and FMCRD run-in fail at the same time, which has extremely low probability of occurrence, the peak reactor pressure would still be controlled by the SRVs. However, the nuclear shutdown then relies on the automatic SLC system injection. The boron would reach the core in about 11 seconds after the initiation. The operation of accumulator driven SLC system produces the initial volumetric flow rate of sodium pentaborate shown in Table 15.5-2. The nuclear shutdown would begin when boron reaches the core.

Stability performance during an ATWS event is examined for the MSIV closure case and the results are discussed at the end of this section.

For the bounding MSIV closure case, a short time after the MSIVs have closed completely, the ATWS high pressure setpoint is reached, which triggers the initiation of the feedwater runback. In the case that control rods fail to insert, the reactor is brought to hot shutdown by automatic SLC boron injection. Operator actions during this event include reestablishing high-pressure makeup to control the water level at 1.5 m (5 ft) above the top of active fuel (TAF). If the Heat Capacity Temperature Limit (HCTL) is reached, the operator depressurizes the reactor via the SRVs to maintain margin to suppression pool limits.

The results for the ARI and the FMCRD cases are less severe than the bounding MSIV closure case. The reactor system responses are presented in Figures 15.5-1a-d for the ARI case, Figures 15.5-2a-d for the FMCRD run-in case, and Figures 15.5-3a-d for the bounding case, respectively. The transient behavior for the ARI and FMCRD cases are listed in Table 15.5-4a, and Table 15.5-4b, respectively. The transient behavior of the SLCS bounding case is listed in Table 15.5-4c. A sequence of the main events that occur during these transients is presented in Table 15.5-4e.

Loss of Condenser Vacuum

This transient starts with a turbine trip because of the low condenser vacuum; therefore, the initial part of the transient is the same as the turbine trip event. However, the MSIVs and turbine bypass valves also close after the condenser vacuum has further dropped to their closure setpoints. Hence, this event is similar to the MSIV closure event for all the key parameters. Similar to the bounding case for the MSIV closure with SLC system described earlier, a bounding case is also analyzed for the Loss of Condenser Vacuum event with input parameters set per Reference 15.5-2. The bounding Loss of Condenser Vacuum case is composed of five major elements that are intended to conservatively bound the key ATWS safety parameters for ESBWR. First, The reactor power used in the bounding analysis is 102% of the normal operating value. Second, the feedwater enthalpy is conservatively chosen to be 105% of the nominal value. Third, the SRV capacity input, shown in Table 15.5-3, chosen for the analysis is set to be conservatively bounding for the vessel bottom pressure response. Fourth, the analysis value of Feedwater Runback (FWRB) coastdown time of 15 seconds with an additional delay in the analysis of 10 seconds for the FWRB activation, is chosen to conservatively bound the peak suppression pool temperature. Fifth, the initial Minimum Critical Power Ratio (MCPR) of the hot bundle is set to a value of 1.16 to conservatively bound the Peak Cladding Temperature (PCT), which is lower than the nominal value of 1.3.

Table 15.5-5a shows the summary of peak values of key parameters for the bounding case and Table 15.5-5b presents a sequence of main events that occur during this transient. Transient behavior is shown in Figures 15.5-4a-d for the bounding case. The high pressure ATWS setpoint is reached shortly after the closure of MSIVs. The high pressure initiates ARI, FMCRD run-in and the SLC timer. The SLC system trip is activated upon APRM not downscale and high-pressure signals and boron flow starts 3 minutes following the trip with a transportation delay time of 10 seconds, and sensor and logic delay of 1 second. As the poison reaches sufficient concentration in the core, the reactor achieves hot shutdown

Loss of Feedwater Heater

This transient does not trip any automatic ATWS logic. A 10-minute operator delay is assumed at the beginning of this event before the ARI is initiated. FMCRD run-in, and SLC timer are activated with the ARI initiation. At this time, the reactor has settled in a new steady state at a higher power level. However, the feedwater runback initiated by manual ARI signal and SRNM ATWS permissive signal causes the water level to drop below Level 2. Low water level results in a closure of all MSIVs, and subsequent reactor pressure increase. SRV opening mitigates the pressure increase. Upon failure of rod insertion, boron injection via the SLC system can bring the reactor to hot shutdown at approximately 15 minutes after the event starts. The transient behavior for the case is shown in Figure 15.5-5a-d. The peak values of the key parameters are shown in Table 15.5-6a. Table 15.5-6b presents a sequence of main events that occur during this transient.

Loss of Non-Emergency AC Power to Station Auxiliaries

In this event, all scram signal paths, including valve position, high flux, high pressure, low level, and all manual attempts have been assumed to fail.

The loss of AC power has the following effects:

- An immediate load rejection occurs. This causes fast closure of the turbine control valves.
- Due to the loss of power to the condensate and feedwater pumps, feedwater is lost.
- The reactor is isolated after loss of main condenser vacuum.

Figures 15.5-6a-d show the transient behavior for the case with automatic SLC system initiation.

The fast closure of the turbine control valves causes a rapid increase of pressure, and the ATWS high-pressure setpoint is reached shortly after the control valves have closed. The ATWS high-pressure signal initiates ARI and FMCRD run-in. If both modes of rod insertion fail, the ATWS high-pressure signal also initiates the timer for SLC. After confirming the rod insertion failure by monitoring the high pressure and SRNM ATWS permissive signal for 3 minutes, the SLC system would be initiated. The reactor is brought to hot shutdown when enough boron concentration is built up in the reactor core.

Table 15.5-7a shows the summary of peak values of key parameters for the event. Table 15.5-7b presents a sequence of main events that occur during this transient.

Loss of Feedwater Flow

This event does not have rapid excursions, as in some of the other events, but is a long-term power reduction with depressurization. Because the pressure begins to fall at the onset of the transient, SRVs are not required until isolation occurs very late in the event and only single group valve cycling is expected to handle decay heat. The containment limits are not approached.

In this event all feedwater flow is assumed to be lost in about five seconds. The mitigation of this event by the SLC system is illustrated in Figures 15.5-7a-d.

After the loss of feedwater has taken place, the pressure, water level and neutron flux begin to fall. The reaching of low water level, Level 2 (L2), activates ARI and FMCRD run-in and starts the SLC system timer, closes MSIVs, initiates the CRD high pressure make-up and the isolation condensers. Failure of rod insertion initiates SLC boron injection when the timer times out while the SRNM ATWS permissive signal exists. The reactor reaches the hot shutdown condition as the boron concentration builds up the core. Table 15.5-8a shows the summary of peak values of key parameters for the case. Table 15.5-8b presents a sequence of main events that occur during this transient.

Load Rejection with a Single Failure in the Turbine Bypass System

The initial characteristics of this transient are much like the MSIV closure described above with a rapid steam shutoff. Pressure and power increases are limited by the action of the SRVs and feedwater runback. As this event progresses, however, the availability of the main condenser makes it possible for the SRVs to be closed sooner and terminates the steam discharge to the suppression pool. The mitigation of this event with the SLCS is illustrated in Figures 15.5-8a-d.

The closure of the turbine control valves causes a rapid increase of pressure. The ATWS high-pressure setpoint is reached shortly after the closure. The high pressure initiates ARI, FMCRD run-in and the SLC timer. If the rods fail to insert into the core, the SLC system is initiated by the SRNM ATWS permissive signal and the high-pressure signal when the timer times out. Table 15.5-9a shows the summary of peak values of key parameters for these events. Table 15.5-9b presents a sequence of main events that occur during this transient. Later initiation of feedwater runback in this event would not cause it to be a limiting event.

Stability during ATWS

Studies are performed to examine core stability during ATWS. Perturbations are introduced in the core power at different times during the transient when power-flow ratios are steady and high, and the transient response to these perturbations is evaluated. The limiting case, where a perturbation was introduced at 25 seconds, is illustrated in Figure 15.5-9. The perturbations in the power are quickly damped out, indicating that the ESBWR operation remains stable during these events.

15.5.4.4 Conclusion

Based upon the results of this analysis, the proposed ATWS design for the ESBWR is satisfactory in mitigating the consequences of an ATWS. All performance requirements specified in Subsection 15.5.4.3.2 are met. It is also demonstrated that the plant operation remains stable during an ATWS event. The results of the system response analyses for the initial core loading documented in Reference 15.5-3 are provided in Reference 15.5-4. System response analyses bounding operation in the feedwater temperature operating domain are documented in Reference 15.5-5.

It is also concluded from results of the above analysis that automatic boron injection could mitigate the most limiting ATWS event with margin. Therefore, an automatic SLCS injection as a backup for ATWS mitigation is acceptable.

15.5.5 Station Blackout

The performance evaluation for Station Blackout (SBO) show conformance to the requirements of 10 CFR 50.63 and is presented in this subsection.

15.5.5.1 Acceptance Criteria

The design meets the following acceptance criteria:

- **Reactor Vessel Coolant Integrity** - Adequate reactor coolant inventory is maintained such that reactor water level is maintained above the core (i.e., top of active fuel).
- **Hot Shutdown Condition** - Achieve and maintain the plant to those shutdown conditions specified in plant Technical Specifications as Hot Shutdown.
- **Containment Integrity** - If containment isolation is involved, the maximum containment and suppression pool pressures and temperatures are maintained below their design limits.

15.5.5.2 Analysis Assumptions

The analysis assumptions and inputs are summarized below.

- Reactor is operating initially at 102% of rated power/100% rated nominal core flow, nominal dome pressure and normal water level at L4. The reactor has been operating at 102% of rated power for at least 100 days.
- The nominal ANSI/ANS 5.1-1994 decay heat model is assumed with an initial core power of 102%.
- SBO starts with loss of all alternating current (AC) power, which occurs at time zero. Auto bus transfer is assumed to fail.
- Loss of AC power trips reactor, feedwater, condensate and circulating water pumps, and initiates a turbine load rejection.
- The reactor scrams occurs at 2.0 seconds due to loss of power supply to feedwater pumps. When feedwater flow is lost, there is a scram signal with a delay time of 2.0 seconds.
- BPV open on load rejection signal.
- Closure of all Main Steam Isolation Valves (MSIVs) is automatically initiated when the reactor water level reaches Level 2 after 30 second time delay. The valves are fully closed at 5.0 seconds after signal.
- CRD pumps are not available due to loss of all AC power. The systems available for initial vessel inventory and pressure control, containment pressure/temperature control and suppression pool temperature control are:
  - Three Isolation Condensers (ICs)
  - The rest of the safety systems are not credited or they do not actuate during the calculated sequence of events.
- The passive IC system is automatically initiated upon the loss of feedwater pump power buses at 3 seconds, to remove decay heat following scram and isolation and IC drain flow provides initial reactor coolant inventory makeup to the reactor pressure vessel.
- The vessel depressurizes, the vessel and other components inventory remains constant; however, the measured level changes because reactor pressure and liquid temperature changes.
- Other assumptions in Tables 15.2-1, 15.2-2 and 15.2-3, are applied to the TRACG calculation

15.5.5.3 Analysis Results

The results of the system response analyses for the initial core loading documented in Reference 15.5-3 are provided in Reference 15.5-4. System response analyses bounding operation in the feedwater temperature operating domain are documented in Reference 15.5-5. As shown in Figures 15.5-10a through 15.5-10e and Table 15.5-10a, during the first 20,000 seconds of depressurization, level is maintained above Level 1. Vessel inventory analysis demonstrates that level remains above Level 1 during the first 72 hours of the transient. Therefore, the requirement for reactor vessel coolant integrity is satisfied. As shown in Table 15.5-10b, considering a constant mass balance, and increased liquid density, the wide range measured level is above 13.5

m (44.3 ft) above vessel zero, which provides adequate margin to L1 ADS analytical limit [11.5 m (37.7 ft) above vessel zero]. The collapsed water level remains well above TAF.

Subsequent to a SBO event, hot shutdown condition can be achieved and maintained by operation of IC systems. Therefore, the requirement for achieving and maintaining hot shutdown condition is met.

With operation of IC system, the containment and suppression pool pressures and temperatures are maintained within their design limits, since there is no release into the wetwell or the drywell. Therefore, the integrity for containment is maintained.

RPV leakage is expected to be minimal for three reasons: 1) there are no recirculation pumps in the design; 2) isolation occurs on L2; 3) the pressure is reduced significantly by ICS. However, if leakage is significant and power has not been restored, the level could drop below the L1 setpoint. In this case ADS, GDCS and PCCS are available to provide core cooling, inventory control and containment heat removal. Because significant depressurization is provided by ICS the impact of depressurization due to ADS initiation would not be as significant as initiation from rated pressure.

As demonstrated above, each acceptance criterion in Subsection 15.5.5.1 is met. Therefore ESBWR can successfully mitigate a SBO event to meet the requirements of 10 CFR 50.63.

This event bounds AOOs with respect to maintaining water level above the top of active fuel. Reanalysis of this event is performed for each fuel cycle.

15.5.6 Safe Shutdown Fire

The fire hazard analysis is provided in Appendix 9A. The performance evaluation is based on TRACG SBO analysis presented in Subsection 15.5.5.

15.5.6.1 Acceptance Criteria

The design meets the following acceptance criteria:

- **Core Subcriticality** - Core subcriticality is achieved and maintained with adequate core shutdown margin, as specified in the plant Technical Specifications.
- **Reactor Vessel Coolant Integrity** - Adequate reactor coolant inventory is maintained such that reactor water level is maintained above the core (i.e., top of active fuel).
- **Hot Shutdown Condition** - Hot shutdown conditions are achieved and maintained.
- **Cold Shutdown Condition** - The cold shutdown condition is achieved within 72 hours and maintained thereafter.
- **Containment Integrity** - If containment isolation is involved, the maximum containment and suppression pool pressures and temperatures are maintained below their design limits.

15.5.6.2 Analysis Assumptions

The worst fire scenario is a fire occurring in the main control room (MCR), and requires operator evacuation. The analysis assumptions and inputs are summarized below.

- Reactor is operating initially at 100% of rated power/100% rated nominal core flow, nominal dome pressure and normal water level.
- The nominal ANSI/ANS 5.1-1994 decay heat model is assumed.
- Order to evacuate the MCR due to a MCR fire and loss of offsite power (LOOP) occurs at time zero.
- The reactor operator manually scrams the reactor before leaving the MCR.
- Closure of all Main Steamline Isolation Valves (MSIVs) is automatically initiated when the reactor water level reaches Level 2, and the valves are fully closed at 5 seconds.
- Feedwater flow is ramped down linearly to zero in 5 seconds after event initiation due to LOOP.
- A single failure is not assumed because safe shutdown fire protection does not require considering a single failure. The systems available for vessel inventory and pressure control, containment pressure/temperature control and suppression pool temperature control are:
  - Isolation Condensers (ICs)
  - Control Rod Drive (CRD) pumps
  - Fuel and Auxiliary Pools Cooling System (FAPCS) in any mode
  - Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) in any mode
  - Safety Relief Valves (SRVs)
  - Depressurization Valves (DPVs)
  - Gravity-Driven Cooling System (GDCS) squib valves
  - GDCS loops
  - Passive Containment Cooling System (PCCS) loops.
- No Spurious operation of SRV or DPV is assumed.
- It is conservatively assumed that it takes operators 10 minutes to evacuate from the MCR to the remote shutdown panel (RSP).
- Four ICs are automatically initiated when the reactor water level reaches Level 2, to stabilize the plant (three ICs are credited in the SBO analysis). Operators can monitor from RSP and manually control ICs to assure the maximum cooldown rate not exceeding 100^{o}F/hr, if necessary.
- When the reactor water level reaches Level 2, CRD pumps are automatically initiated to provide vessel inventory makeup (not credited in SBO analysis). The maximum delay time is 145 seconds upon restoring alternating current (AC) power because off-site power is not available. CRD pumps shall keep the water level above Level 1 to avoid any Automatic Depressurization System (ADS) initiation to blow down the reactor pressure vessel.

- After the operator regains the control in RSP, monitoring and manual control are necessary. RWCU/SDC shall be initiated following the normal shutdown procedure to ensure the reactor pressure vessel temperature is below 100°C (212°F) within 72 hours to meet the cold shutdown requirement.
- ICs and CRD pump flow stabilize the plant. SRVs, DPVs, PCCS and GDCS can be utilized if IC does not stabilize the plant, which is very unlikely.

15.5.6.3 Analysis Results

At event initiation, reactor scram occurs. Therefore, core subcriticality is achieved and maintained.

The analysis results (station blackout event) in Subsection 15.5.5 can be conservatively applied for this fire protection analysis, because more ICs are available for fire protection. As shown in Figures 15.5-10a through 15.5-10e, with operation of three ICs, the reactor water level is well above the top of active fuel. With HP_CRD the water level would recover above Level 2 within approximately 20 minutes. Therefore, the requirement for reactor vessel coolant integrity is satisfied. Additionally, this minimum water level is well above Level 1 and thus, ADS initiation can be avoided.

Subsequent to a fire event, hot shutdown condition can be achieved and maintained by operation of ICs and CRD pumps.

After reactor control room operators regain the control of the reactor in RSP, cold shutdown conditions can be achieved within 72 hours and maintained thereafter following the normal shutdown procedure because the control panel in RSP is identical to the one in MCR and the systems can be fully functioned as designed and can be controlled from RSP as in MCR.

ICs and CRD pumps stabilize the plant without SRV actuation or ADS blowdown, consequently there is no heat-up in the suppression pool and containment. Therefore, the integrity for containment is maintained.

As demonstrated above, each acceptance criterion in Subsection 15.5.6.1 is met. Therefore ESBWR can successfully mitigate a fire event.

15.5.7 Waste Gas System Leak or Failure

The safety analysis of waste gas system leak or failure is provided in Subsection 11.3.7.

15.5.8 COL Information

15.5-1-A SBO Event for Specific Initial core Design – (Deleted)

15.5-2-H SBO Event for Reload Core Design – (Deleted)

15.5.9 References

15.5-1 General Electric Company, "Assessment of BWR Mitigation of ATWS," NEDE-24222, September 1979.

15.5-2 GE Energy Nuclear, "TRACG Application for ESBWR," NEDE-33083P Supplement 2, Class III, (Proprietary), January 2006.

15.5-3 Global Nuclear Fuel, “ESBWR Initial Core Nuclear Design Report”, NEDC-33326-P, Class III (Proprietary), Revision 0, July 2007, NEDO-33326, Class I (Non-proprietary), Revision 0, July 2007.

15.5-4 GE-Hitachi Nuclear Energy, “ESBWR Initial Core Transient Analyses”, NEDO-33337 Class I, Revision 0, Scheduled September 2007.

15.5-5 GE-Hitachi Nuclear Energy, “ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis”, NEDO-33338 Class I, Revision 0, Scheduled September 2007.

15.5-6 Global Nuclear Fuel, “GE14 for ESBWR Nuclear Design Report”, NEDC-33239-P, Class III (Proprietary), Revision 2, April 2007, NEDO-33239, Class I (Non-proprietary), Revision 2, April 2007.

Table 15.5-1

ATWS Performance Requirements

RPV Peak Pressure MPag (psig)	**Maximum Pool Temperature** °C (°F)	**Fuel Integrity**	**Peak Cladding Temperature** °C (°F)	**Local Oxidation of Cladding**	**Maximum Containment Pressure** kPa (psia)
10.34 (1500)	121 (250)	Coolable Geometry	Less than 1204.4 (2200)	Not to exceed 17% of total cladding thickness	414 (60)

Table 15.5-2

ATWS Initial Operating Conditions

Parameters	Nominal Value	Bounding Value
Power, MWt/% NBR	4500/100	4590/102
Vessel Diameter, m (ft)	7.1 (23.3)	7.1 (23.3)
Numbers of Fuel Bundles	1132	1132
Nominal and Bounding Initial Conditions Used in ATWS Analysis		
Parameters	**Nominal Value**	**Bounding Value**
Dome Pressure, MPaG (psig)	7.07 (1025)	6.98 (1013)
Natural Circulation Core Flow, Mkg/hr (Mlb/hr)*	37.55 (82.78)	37.03 (81.64)
Steam/Feed Flow, kg/s (Mlbm/hr)	2423 (19.23)	2531 (20.09)
Feedwater Temperature,, °C (°F)	215.05 (419.1)	224.9 (436.8)
Nuclear Characteristics Used in TRACG Simulations Condition Exposure	Reference 15.5-6 EOC	Reference 15.5-6 EOC
Suppression Pool Volume, m^3 (ft^3)	4354 (153,760)	4354 (153,760)
3 Isolation Condensers volume, 3 Units, from steam box to discharge at vessel, m^3 (ft^3)	42.1 (1485.8)	42.1 (1485.8)
Initial Suppression Pool Temperature, °C (°F)	43.3 (109.9)	43.3 (109.9)
SLCS accumulator driven initial flow, m^3/s (gpm)	0.03 (475)	0.03 (475)

*The required measurement accuracy is less than or equal to 7.5% of rated core flow for one standard deviation (1σ).

Table 15.5-3

ATWS Equipment Performance Characteristics

Parameters	Value
MSIV Closure Time, s	≥3.0
Delay before start of Electro-Hydraulic Rod Insertion, s	≤1
Electro-Hydraulic Control Rod Insertion Time, s	≤130
Maximum time for start of motion of ARI rods, s	15
Maximum time for all ARI rods to be fully inserted, s	25
SLC system transportation and DCIS logic delay time, s	≤11
Safety Relief Valve (SRV) System Capacity, % NBR Steam Flow/No. of Valves – Nominal Cases[1], and Bounding Cases[1]	≥102/18
High Reactor Pressure Vessel (RPV) Dome pressure setpoint, MPaG (psig)	7.76 (1125)
SRV Setpoint Range, MPaG (psig)	8.62 to 8.76 (1250-1270)
SRV Opening Time, s	<0.5
Pressure Drop Below Setpoint for SRV Closure, % nameplate	≤96
Low Water Level (Level 2) Trip setpoint (from vessel bottom reference zero), cm (in)	1605.0 (631.9)
CRD (High Pressure Make-Up Function) Low Water Level Initiation Setpoint, cm (in)	1605.0 (631.9)
CRD (High Pressure Make-Up Function) Flow Rate, m^3/s (gal/min)	0.07 (1035)
ATWS Dome Pressure Sensor Time Constant, s	≤0.5
ATWS Logic Time Delay, s	≤1
Pool Cooling Capacity, kW/C	430.6
Low Water Level For Closure of MSIVs, cm (in)	1605.0 (631.9)
Low Steamline Pressure For Closure of MSIVs, MPaG (psig)	5.41 (785)
Temperature For Automatic Pool Cooling, ºC (ºF)	48.9 (120)

(1) The SRV capacity used in the analysis is 102% of the ASME rated capacity noted in Table 5.2-2.

Table 15.5-4a

ATWS MSIV Closure Summary - ARI Case

Parameter	Value	Time (s)
Sensed Maximum Neutron Flux, %	212	3
Maximum Vessel Bottom Pressure, MPaG (psig)	9.22 (1337.0)	7
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	52.2 (125.9)	37
Associated Containment Pressure, kPaG (psig)	168.0 (24.37)	37
Peak Cladding Temperature, ºC (ºF)	589.8 (1093.7)	14

Table 15.5-4b

ATWS MSIV Closure Summary - FMCRD Case

Parameter	Value	Time (s)
Sensed Maximum Neutron Flux, %	212	3
Maximum Vessel Bottom Pressure, MPaG (psig)	9.22 (1337.0)	7
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	63.3 (145.9)	103
Associated Containment Pressure, kPaG (psig)	186.1 (27.0)	103
Peak Cladding Temperature, ºC (ºF)	611.1 (1132.0)	16

Table 15.5-4c

ATWS MSIV Closure Summary – SLC System Bounding Case

Parameter	Value	Time (s)
Sensed Maximum Neutron Flux, %	213	3
Maximum Vessel Bottom Pressure, MPaG (psig)	9.41 (1364)	21
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	73.1 (163)	370
Associated Containment Pressure, kPaG (psig)	206.2 (29.91)	370
Peak Cladding Temperature, ºC (ºF)	850.3 (1562.5)	28

Table 15.5-4d

ATWS MSIV Closure Summary – SLC System Bounding Pool Temperature Case

(Deleted)

Table 15.5-4e

ATWS MSIV Closure Sequence of Events

Time (s)			Event
ARI	**FMCRD**	**MSIV Closure SLC System Bounding Case**	
0	0	0	MSIV Closure starts
2	2	2	IC initiates
4	4	4	ATWS trip set at high pressure
5	5	5	SRVs open
31	41	42	Level drops below L2 set point
41	52	52	HP_CRD flow starts
-	-	195	SLCS injection starts*
-	-	715	High pressure design volume of borated solution injected into bypass
20	5		Start of Rod Motion
30	-		Alternate Rod Insertion complete**
-	135		FMCRD Run-in complete***

Table 15.5-5a

ATWS Loss of Condenser Vacuum Summary – SLC System Bounding Case

Parameter	Value	Time (s)
Sensed Maximum Neutron Flux, %	229	9
Maximum Vessel Bottom Pressure, MPaG (psig)	9.40 (1364)	26
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	73.0 (164)	364
Associated Containment Pressure, kPaG (psig)	206.1 (29.89)	364
Peak Cladding Temperature, ºC (ºF)	850.0 (1562)	34

Table 15.5-5b

ATWS Loss of Condenser Vacuum Sequence of Events Bounding Case

Time (s)	Event
0	Loss of Condenser Vacuum
0	Turbine Trip initiated
6	MSIV closure trip set
8	IC initiates
10	ATWS trip set at high pressure
11	SRVs open
48	Level drops below L2 set point
58	HP_CRD flow starts
201	SLCS injection starts
721	High pressure design volume of borated solution injected into bypass

Table 15.5-5c

ATWS Loss of Condenser Vacuum Summary - SLC System Bounding Pool Temperature Case (Deleted)

Table 15.5-5d

ATWS Loss of Condenser Vacuum Sequence of Events Bounding Pool Temperature Case (Deleted)

Table 15.5-6a

ATWS Loss of Feedwater Heating Summary - SLC System Case

Parameter	Value	Time (s)
Sensed Maximum Neutron Flux, %	119	472
Maximum Vessel Bottom Pressure, MPaG (psig)	8.72 (1264.2)	693
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	48.8 (119.9)	903
Associated Containment Pressure, kPaG (psig)	163.3 (23.69)	903
Peak Cladding Temperature, ºC (ºF)	313.2 (595.8)	622

Table 15.5-6b

ATWS Loss of Feedwater Heating Sequence of Events

Time (s)	Event
0	Loss of Feedwater heating
600	Feedwater runback initiated by operator
637	L2 setpoint reached
637	ATWS trip set at L2
648	HP_CRD flow starts
667	MSIV closure starts
671	IC initiates
692	SRVs open
791	SLCS flow starts
1312	High pressure design volume of borated solution injected into bypass

Table 15.5-7a

ATWS Loss of Non-Emergency AC Power to Station Auxiliaries Summary - SLC System Case

Parameter	Value	Time (s)
Sensed Maximum Neutron Flux, %	209	12
Maximum Vessel Bottom Pressure, MPaG (psig)	9.18 (1331.2)	15
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	68.3 (155.0)	349
Associated Containment Pressure, kPaG (psig)	196.0 (28.42)	349
Peak Cladding Temperature, ºC (ºF)	438.2 (820.7)	15

Table 15.5-7b

ATWS Loss of Non-Emergency AC Power to Station Auxiliaries Sequence of Events

Time (s)	Event
0	Loss of AC Power
8	MSIV Closure starts
9	IC initiates
12	ATWS trip set at high pressure
13	SRVs open
43	Level drops below L2 set point
71	Level drops below L1 set point
163	HP_CRD flow starts
203	SLCS injection starts
724	High pressure design volume of borated solution injected into bypass

Table 15.5-8a

ATWS Loss of Feedwater Flow Summary - SLC System Case

Parameter	Value	Time (s)
Sensed Maximum Neutron Flux, %	100	0
Maximum Vessel Bottom Pressure, MPaG (psig)	8.72 (1264.1)	99
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	50.3 (122.5)	358
Associated Containment Pressure, kPaG (psig)	165.3 (23.97)	358
Peak Cladding Temperature, ºC (ºF)	311.4 (592.59)	0.2

Table 15.5-8b

ATWS Loss of Feedwater Flow Sequence of Events

Time (s)	Event
0	Feedwater Pump coastdown starts
34	Level drops below L2 set point, ATWS trip is set
46	HP_CRD flow starts
66	MSIV Closure starts
66	Level drops below L1 set point, IC flow starts
98	SRVs open
227	SLCS injection starts
748	High pressure design volume of borated solution injected into bypass

Table 15.5-9a

ATWS Load Rejection with a Single Failure in the Turbine Bypass System Summary - SLC System Case

Parameter	Value	Time (s)
Sensed Maximum Neutron Flux, %	194	0.8
Maximum Vessel Bottom Pressure, MPaG (psig)	8.88 (1288.0)	14
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	53.7 (128.7)	346
Associated Containment Pressure, kPaG (psig)	170.3 (24.71)	346
Peak Cladding Temperature, ºC (ºF)	327.3 (621.1)	8

Table 15.5-9b

ATWS Load Rejection with a Single Failure in the Turbine Bypass System Sequence of Events

Time (s)	Event
0	Generator Load Rejection
3	ATWS trip set at high pressure
7	SRVs open
9	IC initiates
42	Level drops below L2 set point
53	HP_CRD flow starts
73	MSIV Closure starts
73	Level drops below L1 set point
194	SLCS injection starts
714	High pressure design volume of borated solution injected into bypass

Table 15.5-10a

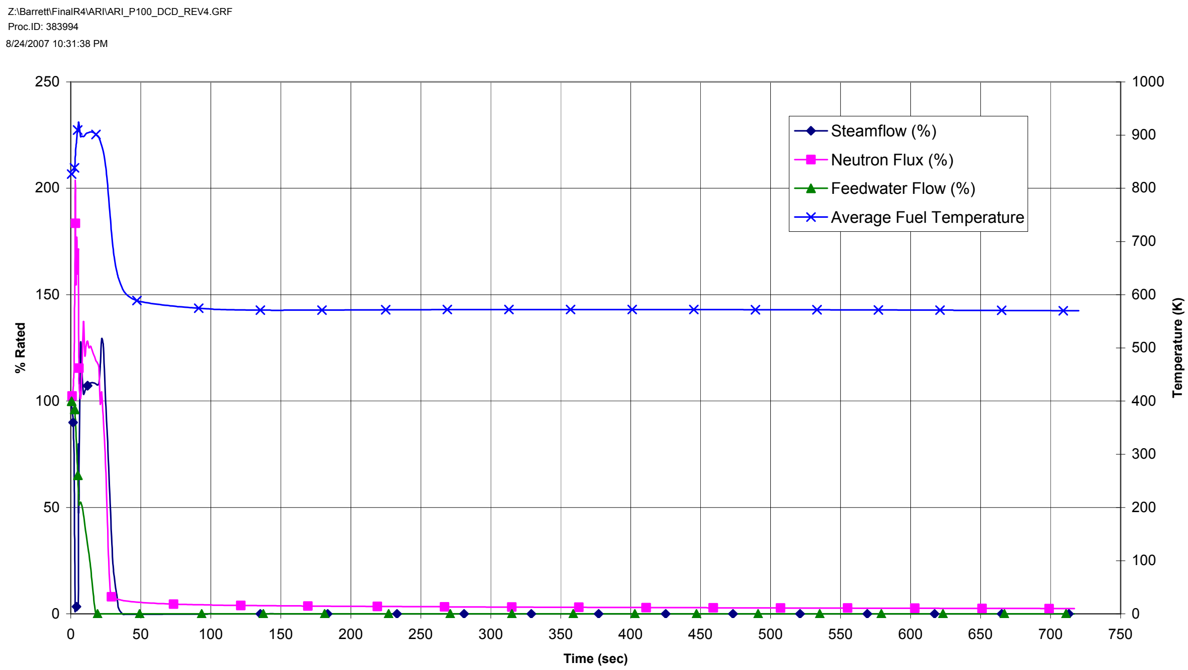
Sequence of Events for Station Blackout

Time (s)	**Event**
0.0	Loss of AC power to station auxiliaries, which initiates a generator trip.
0.0	Additional Failure assumed in transfer to "Island mode", Feedwater, condensate and circulating water pumps are tripped.
0.0	Turbine control valve fast closure is initiated.
0.0	Turbine control valve fast closure initiates main turbine bypass system operation.
0.0	Feedwater and condenser pumps are tripped.
0.02	Turbine bypass valves start to open.
0.08	Turbine control valves closed.
2.0	Loss of power on the four power generation busses is detected and initiates a reactor scram and activation of ICs with one second delay.
5.0	Feedwater flow decay to 0.
6.9	Vessel water level reaches Level 3
9.6	Vessel water level reaches Level 2.
18.0	ICs begins to drop cold water inside the vessel
33.0	ICs drainage valve is fully open
39.6	MSIV valve begins to close
44.6	MSIV is totally closed
72 hours	The system reached the conditions described in Table 15.5-10b

Table 15.5-10b

Theoretical Vessel Conditions at 72 hours after SBO

Parameter	Value
Dome pressure, PaG (psig)	0 (0)
Vessel Bottom Pressure, PaG (psig)	123000 (17.8)
Decay heat, MW	19.5
Wide range measured level over TAF, m (ft)	6.3 (20.7)
Collapsed Level over TAF, m (ft)	5.6 (18.4)
IC flow, kg/s (lb/hr)	8.6 (68,573)



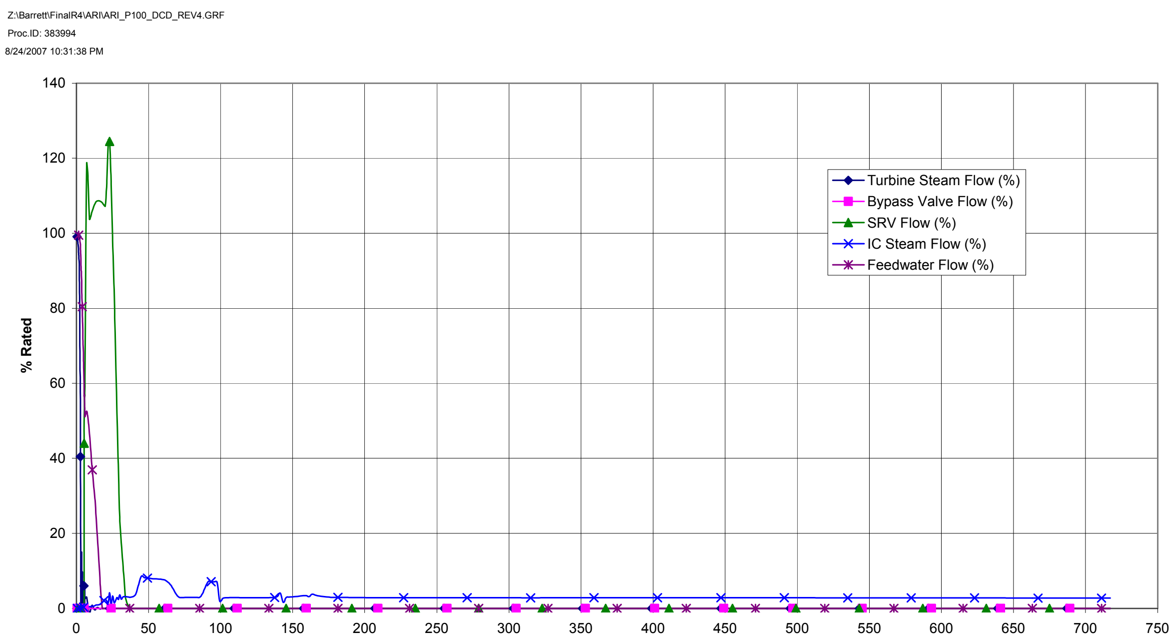
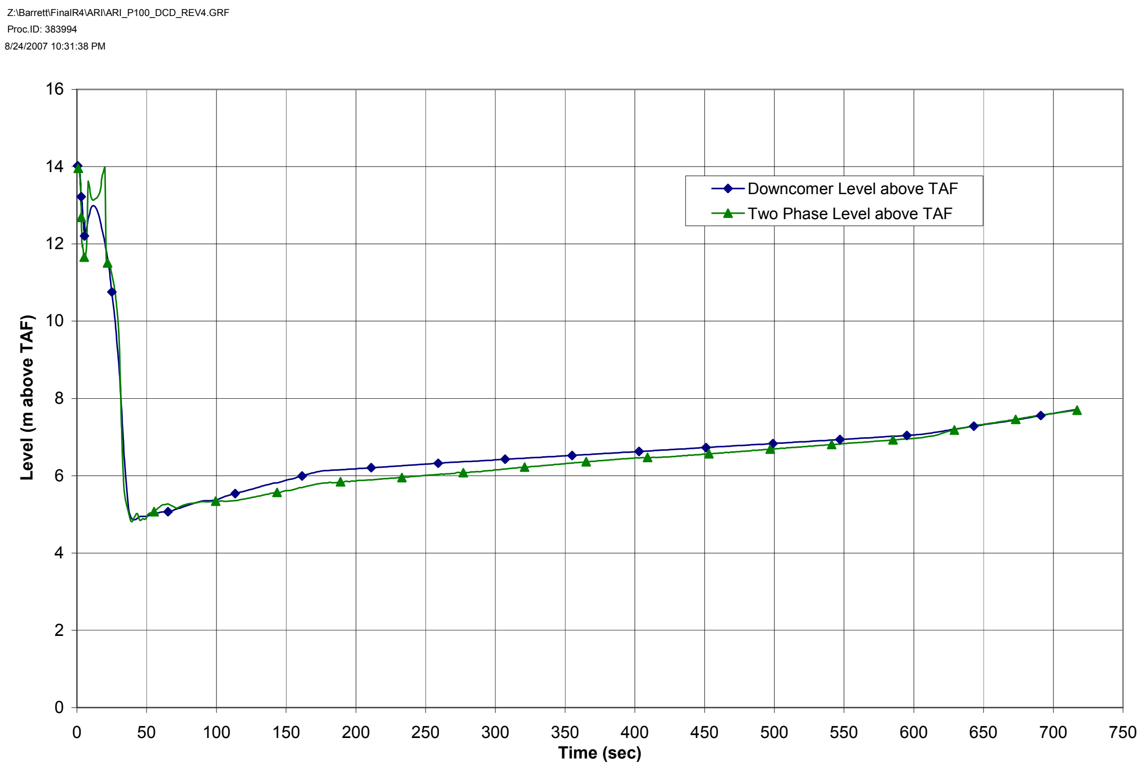


Figure 15.5-1a. MSIV Closure with ARI



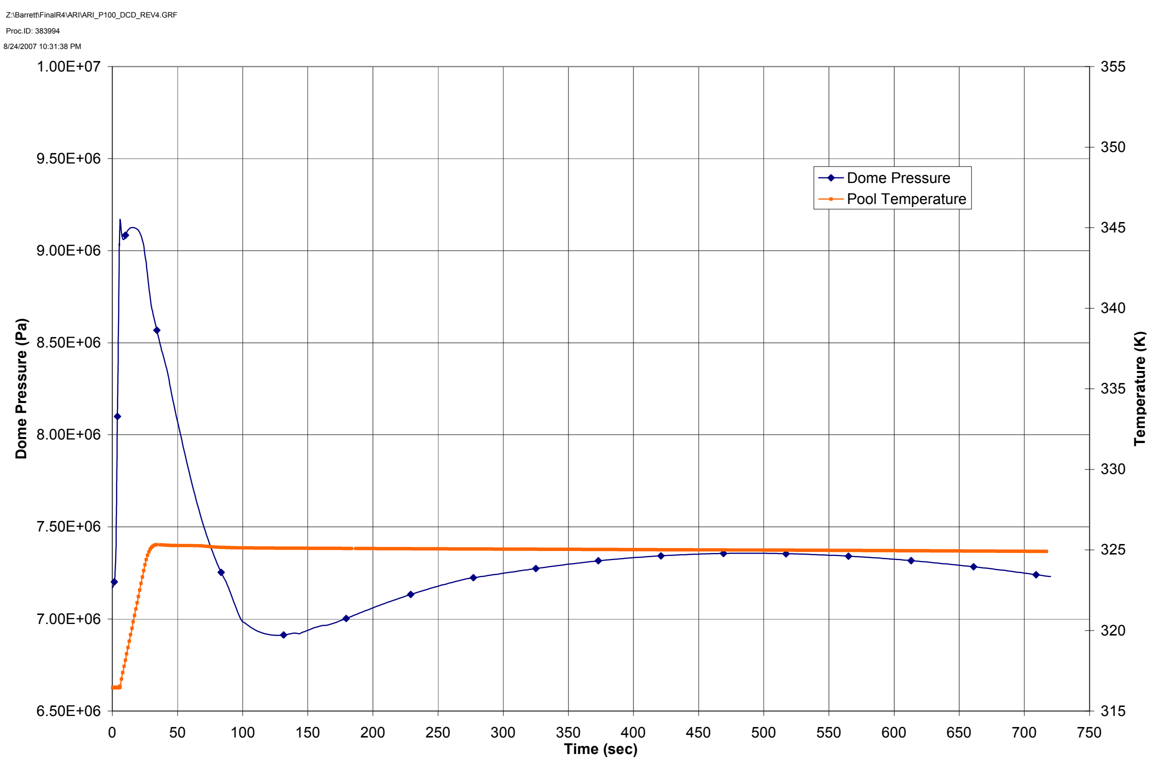
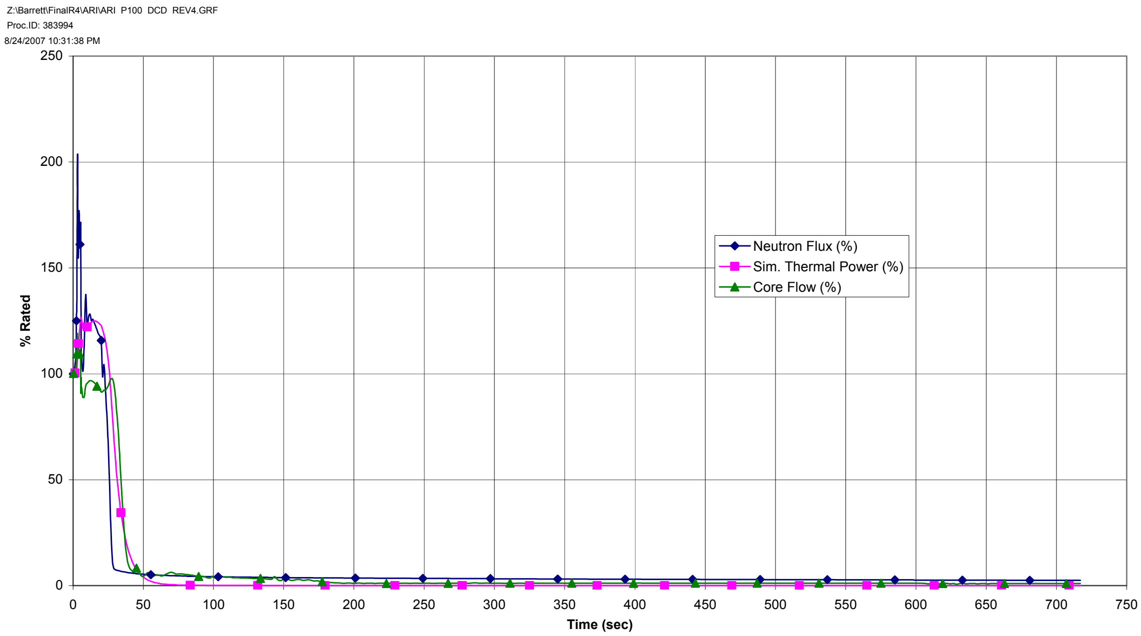


Figure 15.5-1b. MSIV Closure with ARI



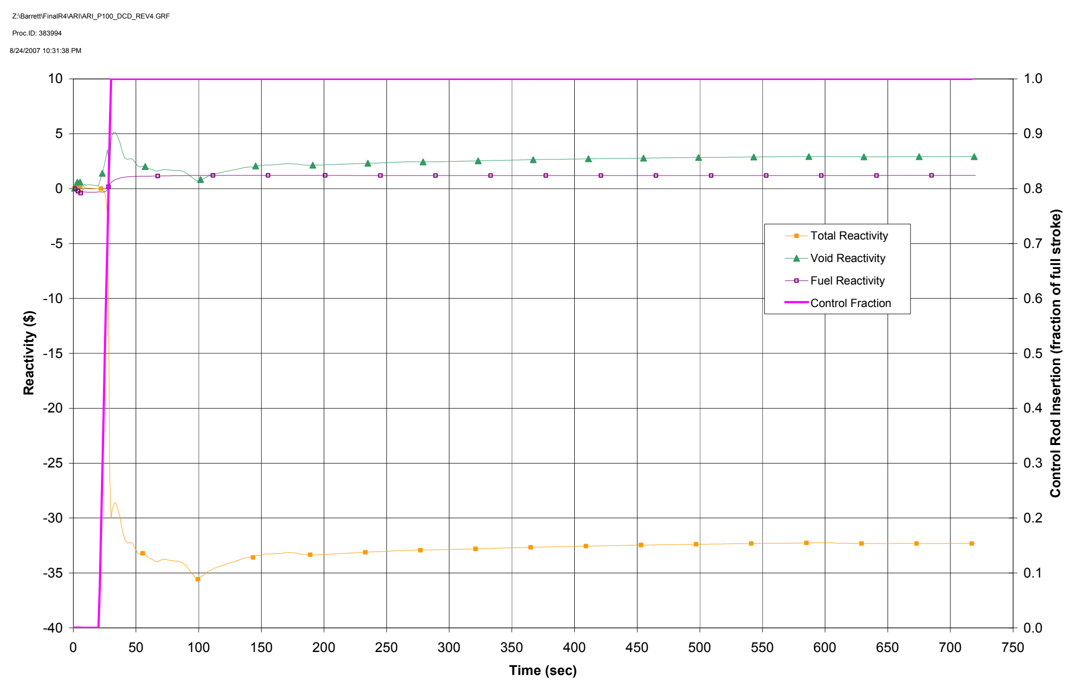


Figure 15.5-1c. MSIV Closure with ARI

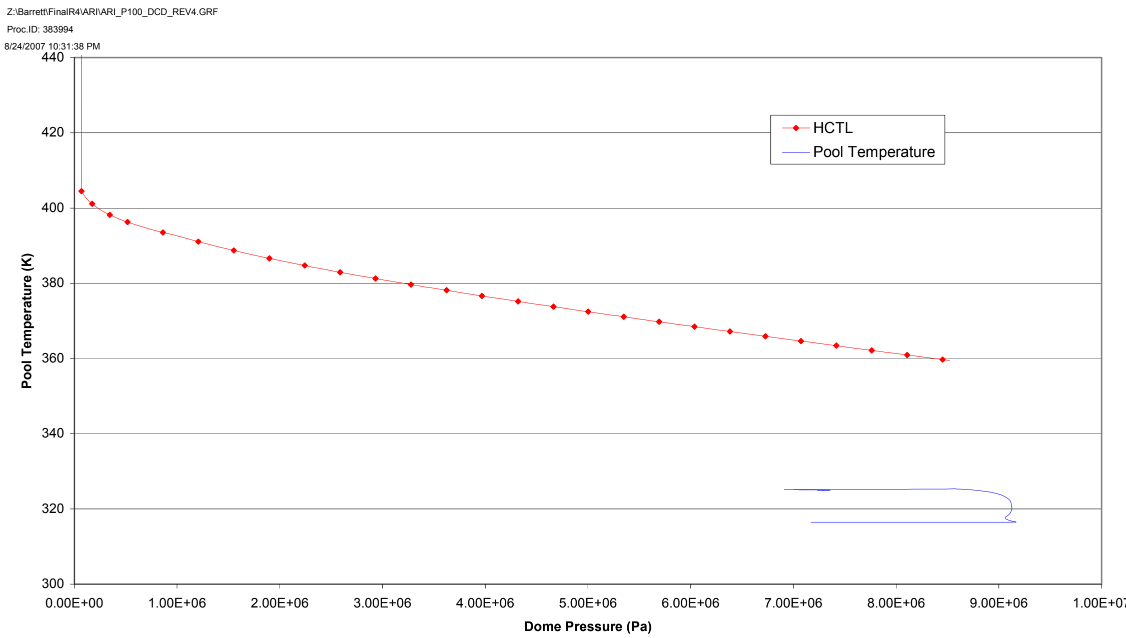


Figure 15.5-1d. MSIV Closure with ARI

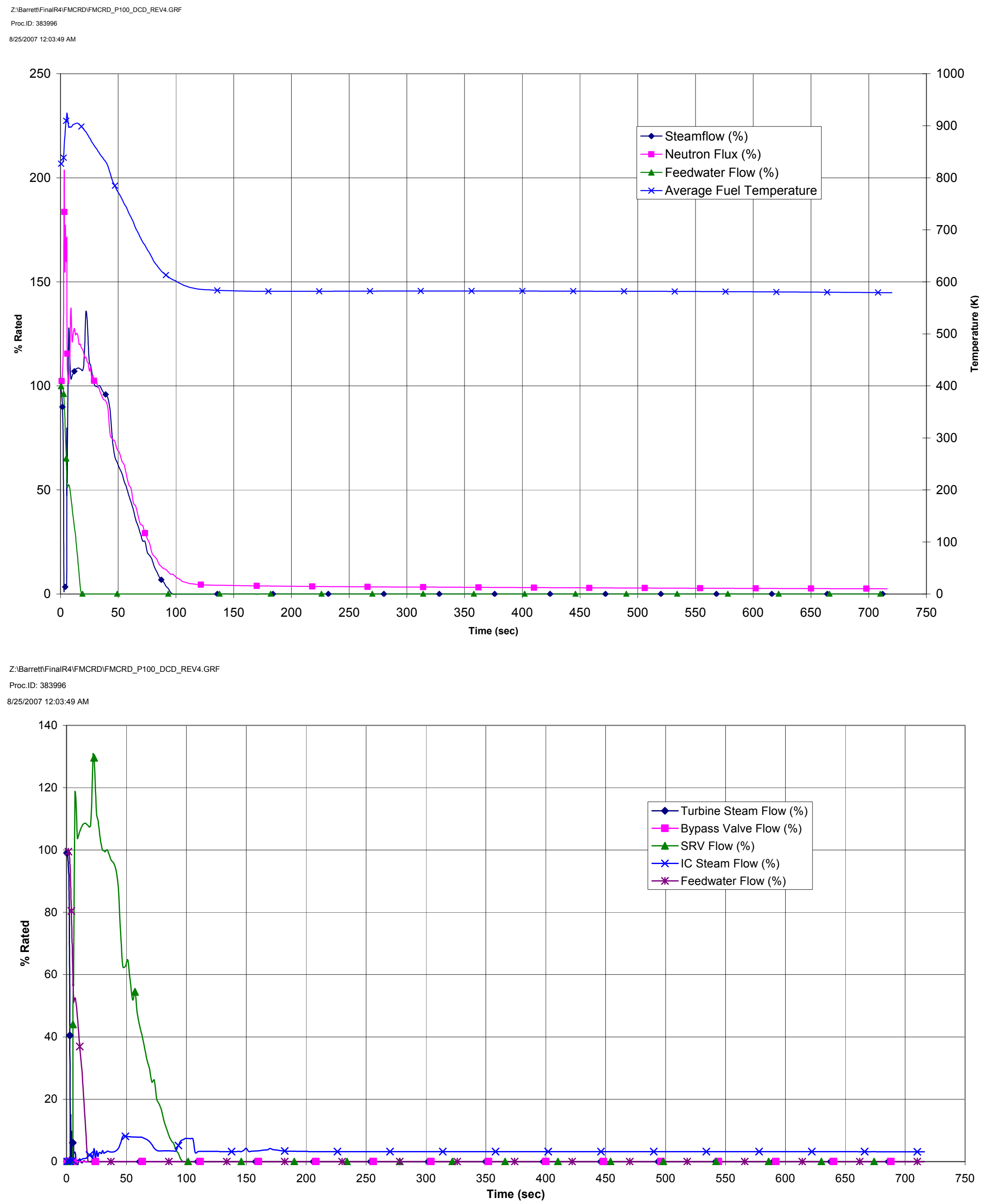
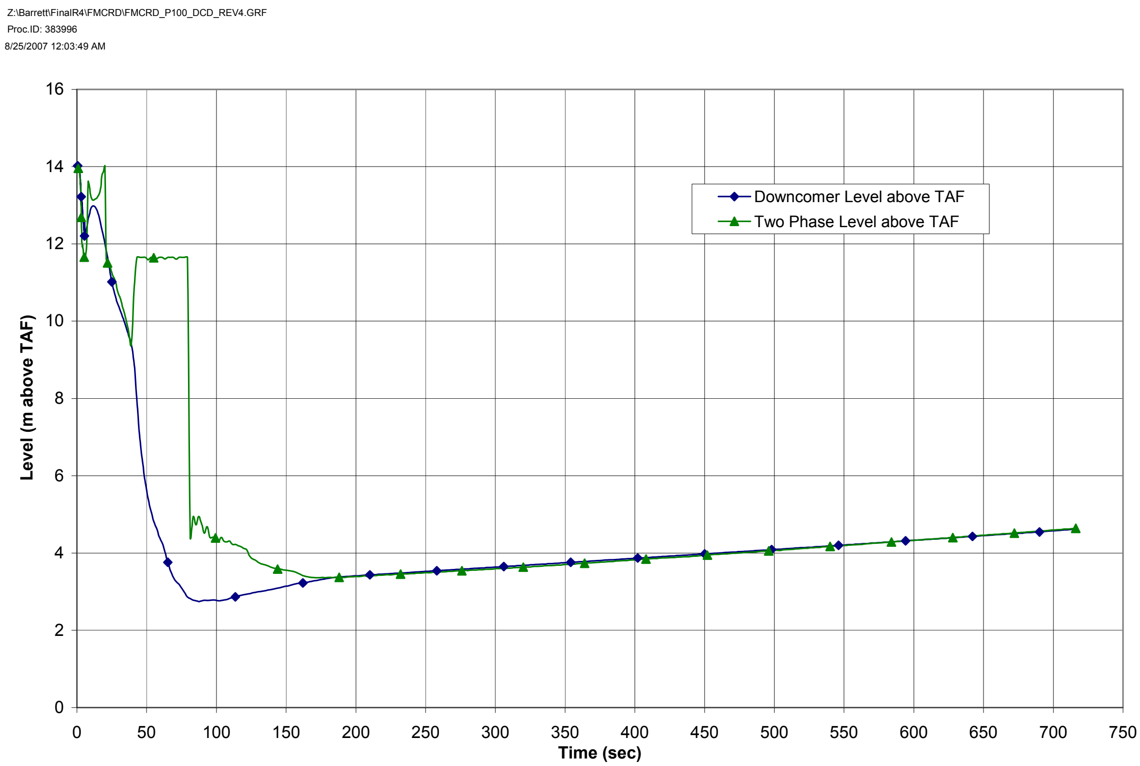


Figure 15.5-2a. MSIV Closure with FMCRD Run-in



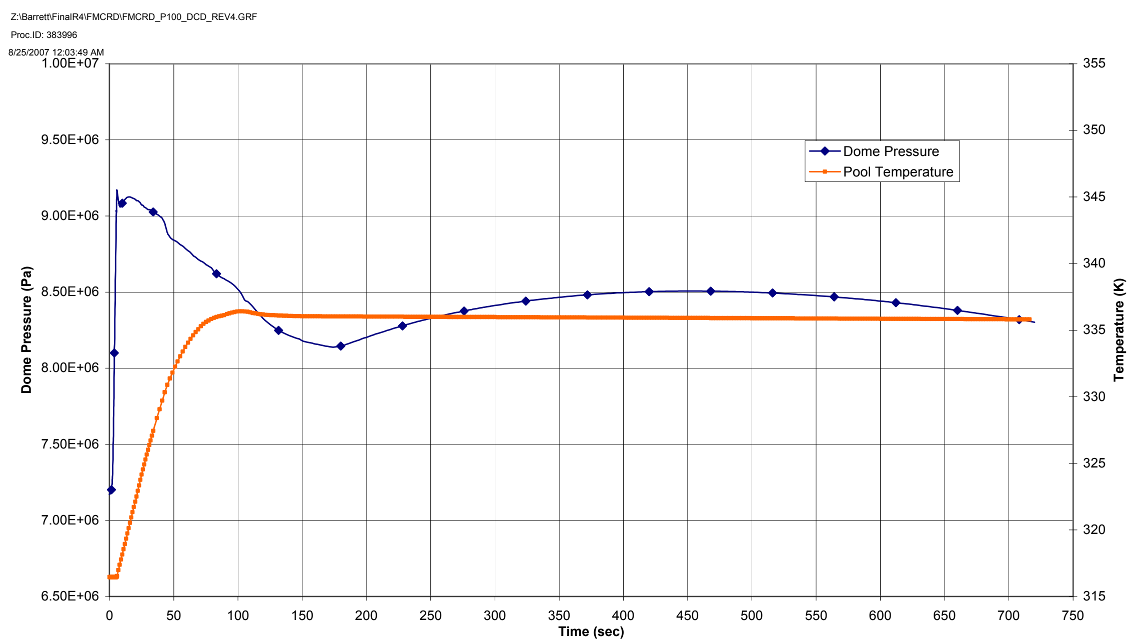


Figure 15.5-2b. MSIV Closure with FMCRD Run-in

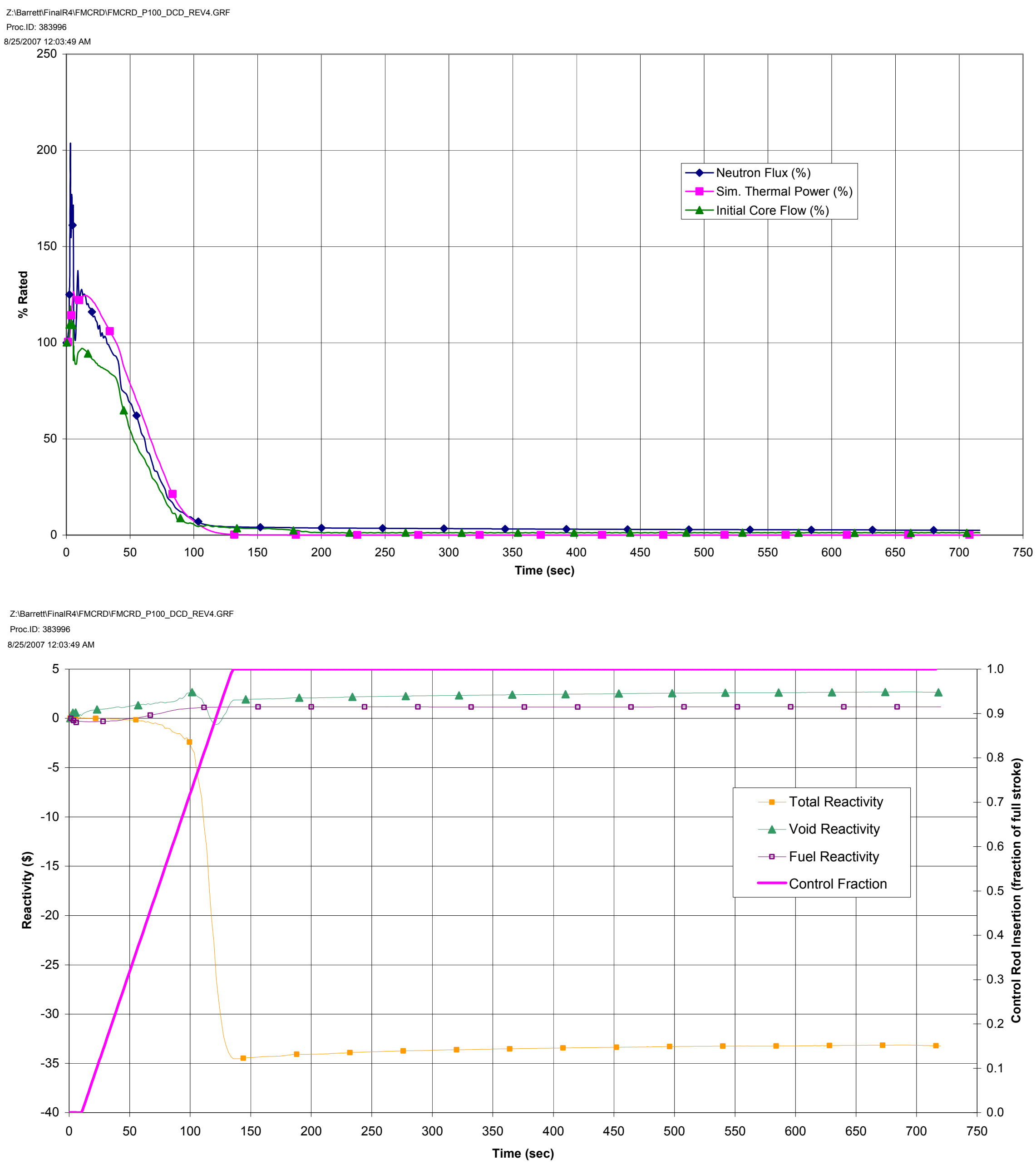


Figure 15.5-2c. MSIV Closure with FMCRD Run-in

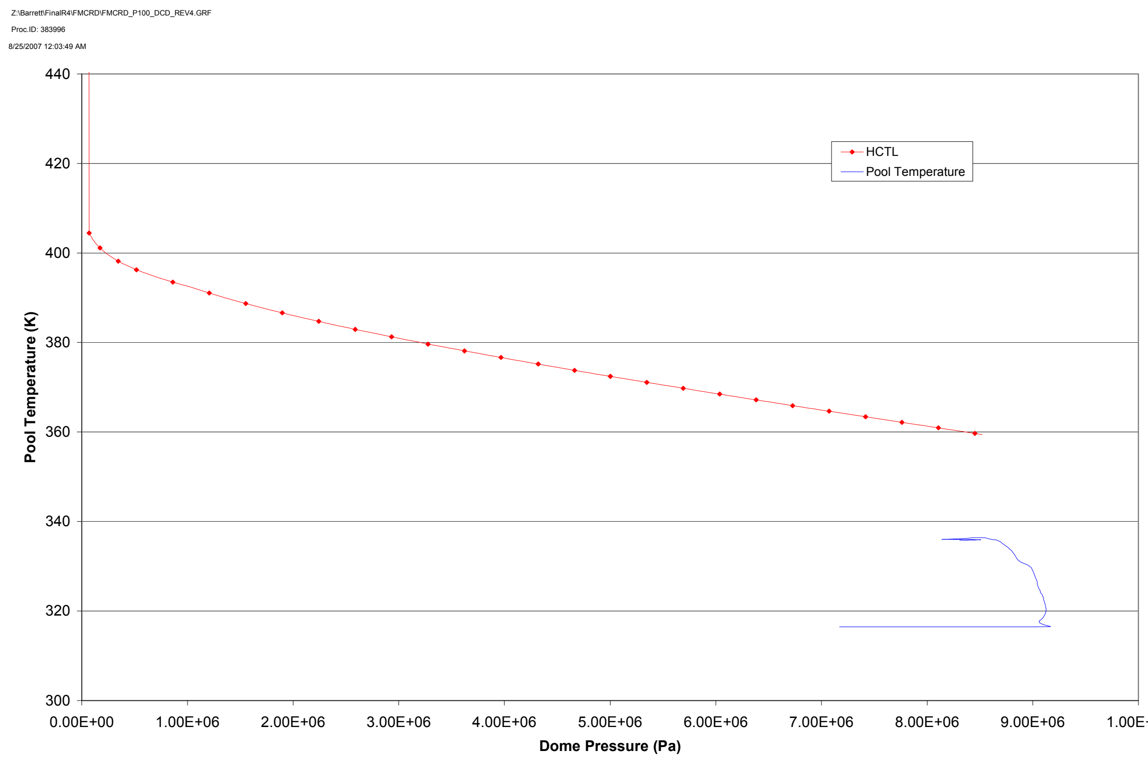
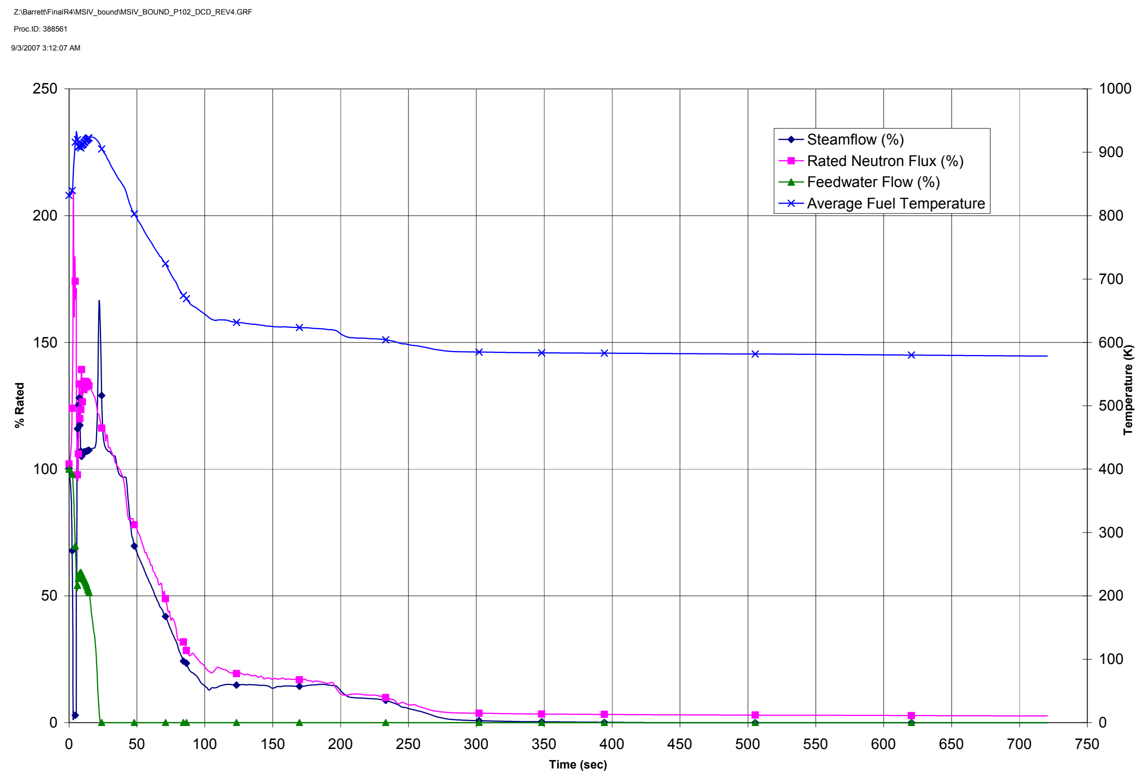


Figure 15.5-2d. MSIV Closure with FMCRD Run-in



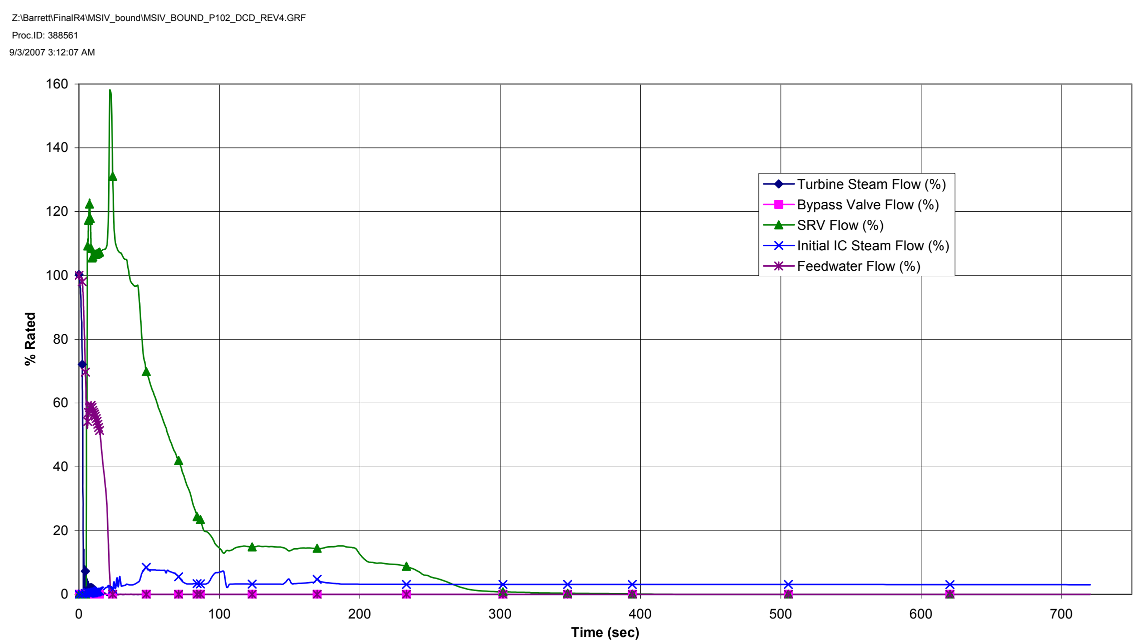


Figure 15.5-3a. MSIV Closure - SLC System Bounding Case

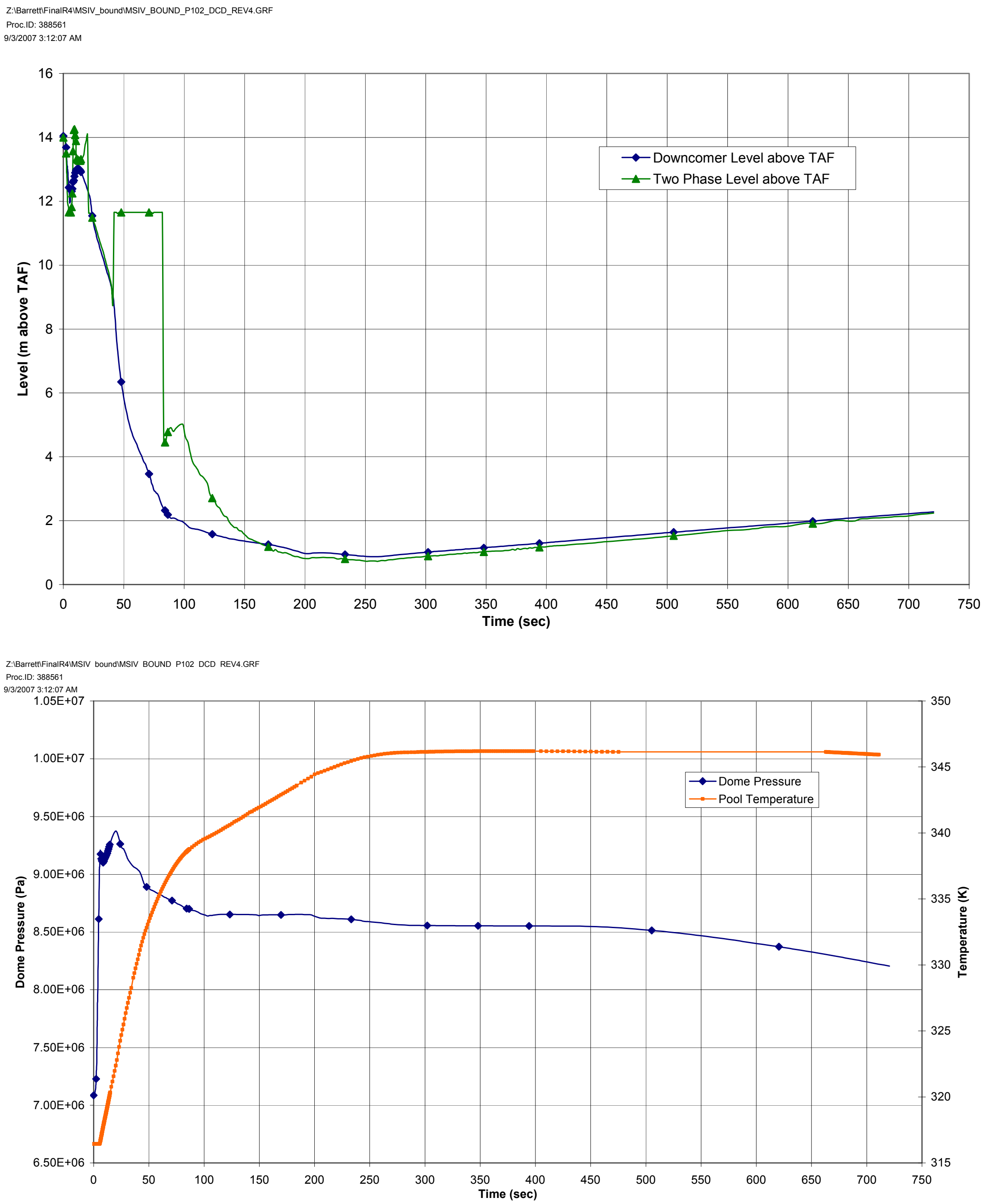
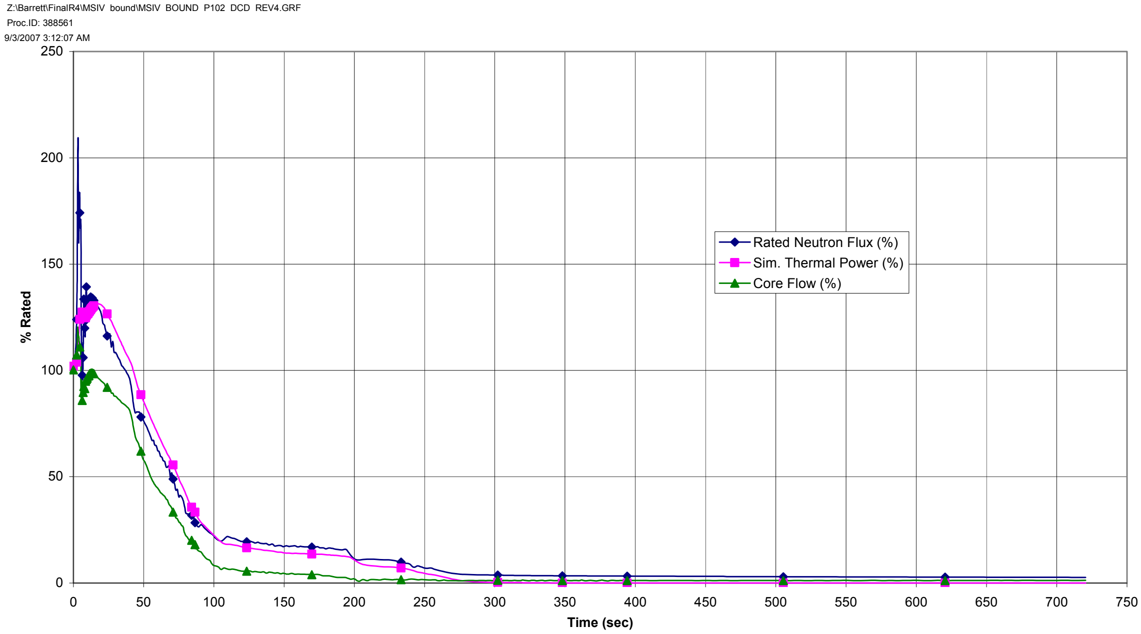


Figure 15.5-3b. MSIV Closure - SLC System Bounding Reactor Vessel Pressure Case



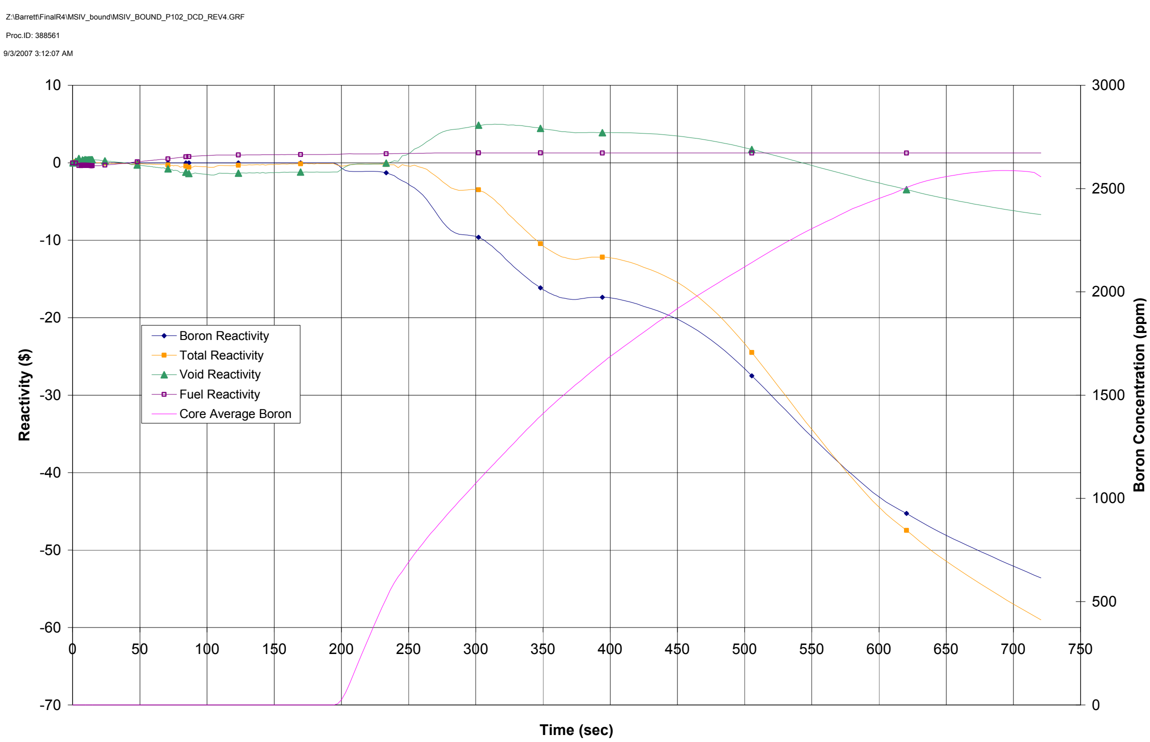


Figure 15.5-3c. MSIV Closure - SLC System Bounding Case

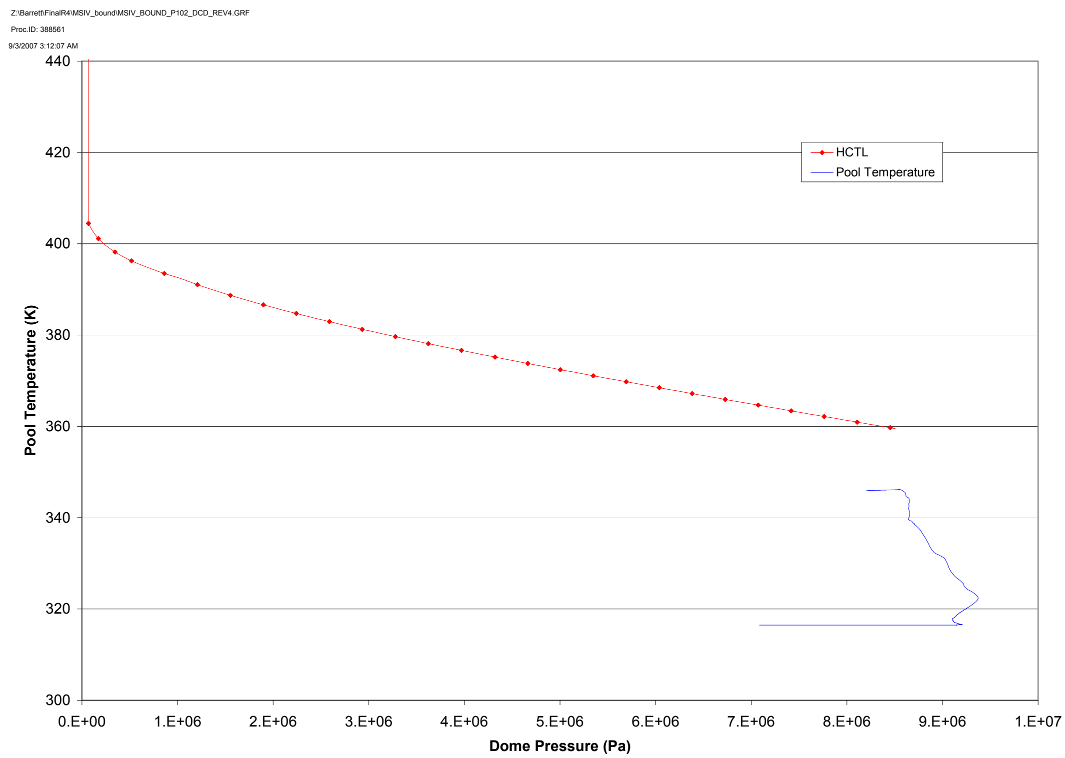


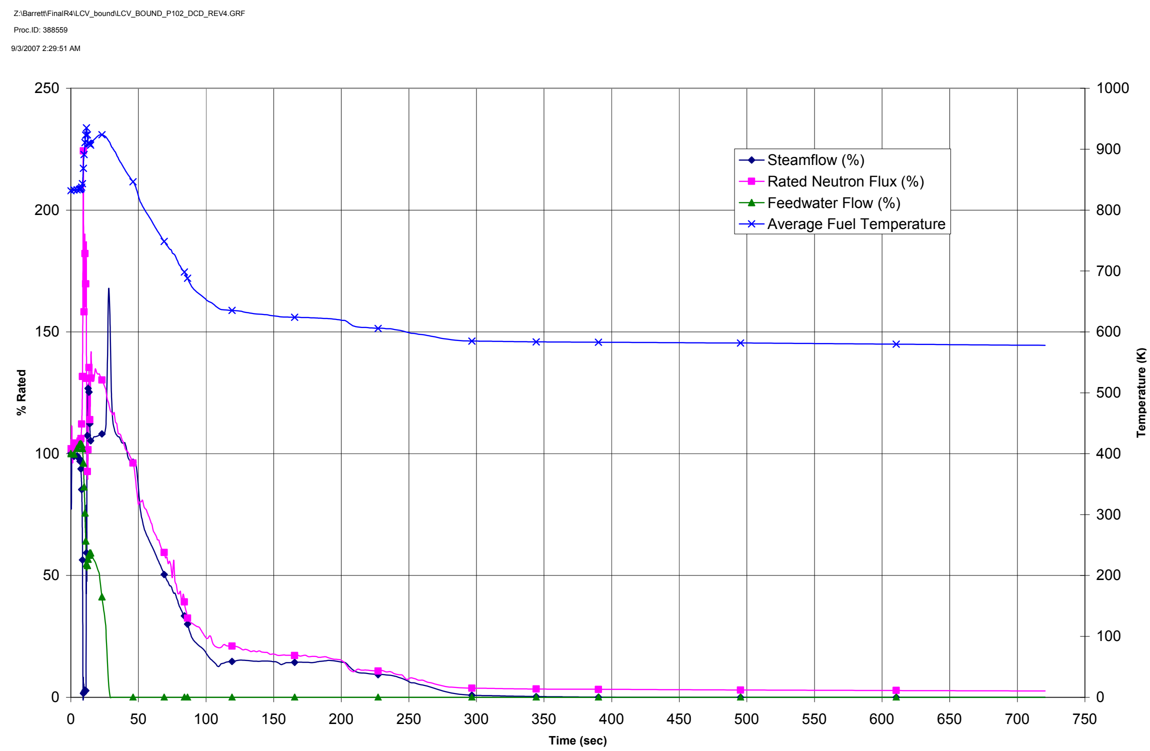
Figure 15.5-3d. MSIV Closure - SLC System Bounding Case

Figure 15.5-3e. MSIV Closure - SLC System Bounding Pool Temperature Case (Deleted)

Figure 15.5-3f. MSIV Closure - SLC System Bounding Pool Temperature Case (Deleted)

Figure 15.5-3g. MSIV Closure - SLC System Bounding Pool Temperature Case (Deleted)

Figure 15.5-3h. MSIV Closure - SLC System Bounding Pool Temperature Case (Deleted)



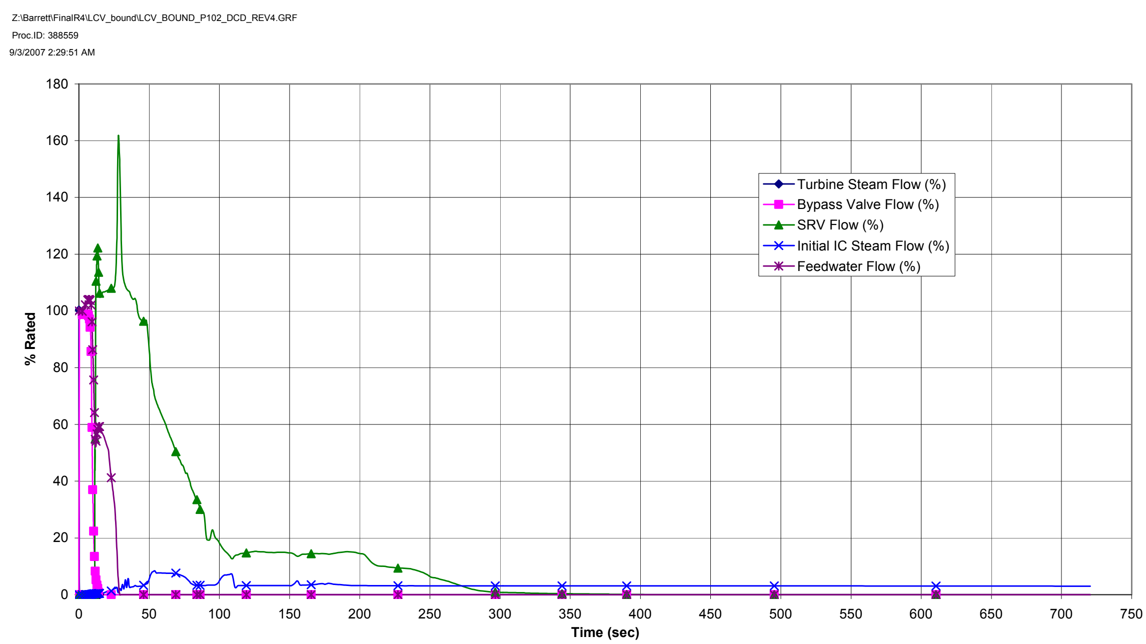
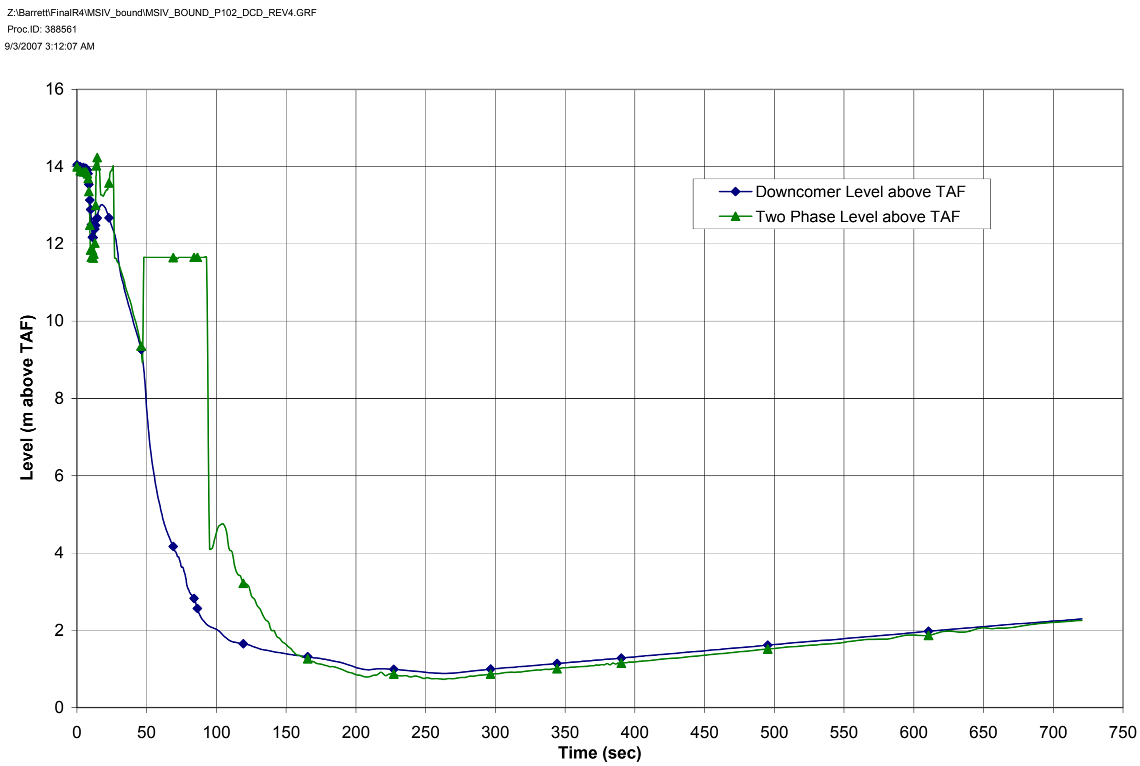


Figure 15.5-4a. Loss of Condenser Vacuum SLC System Bounding Case



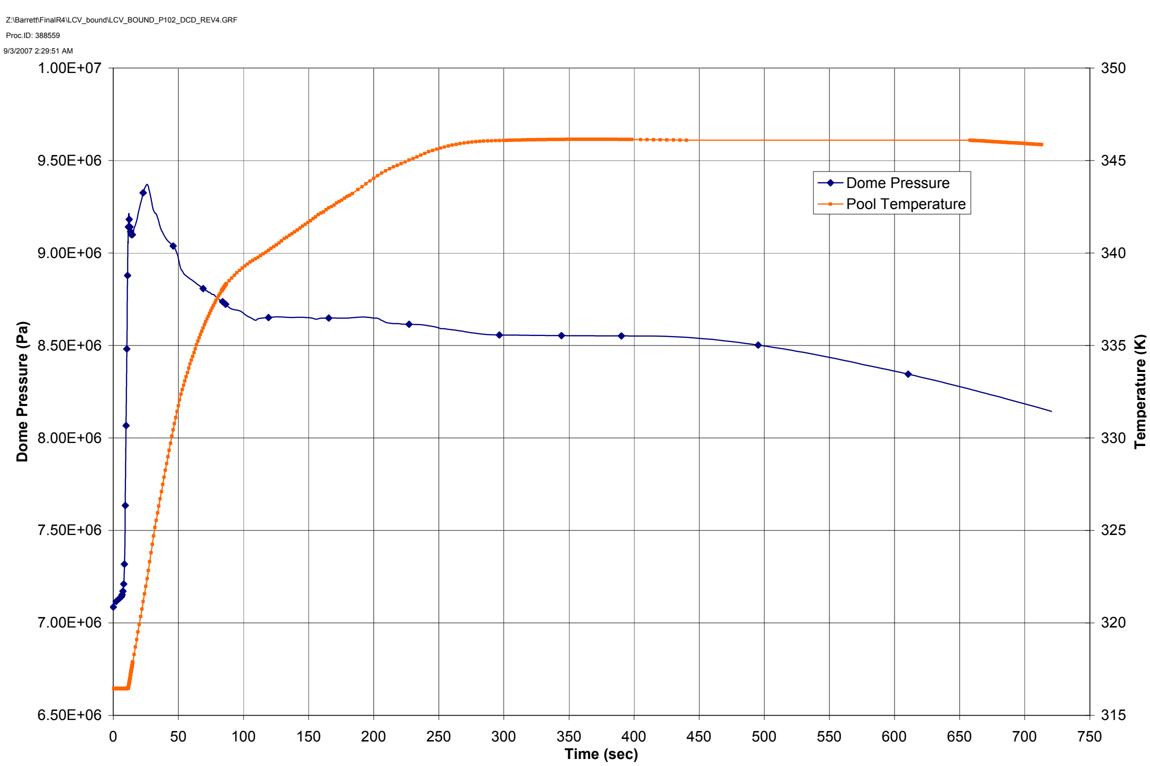
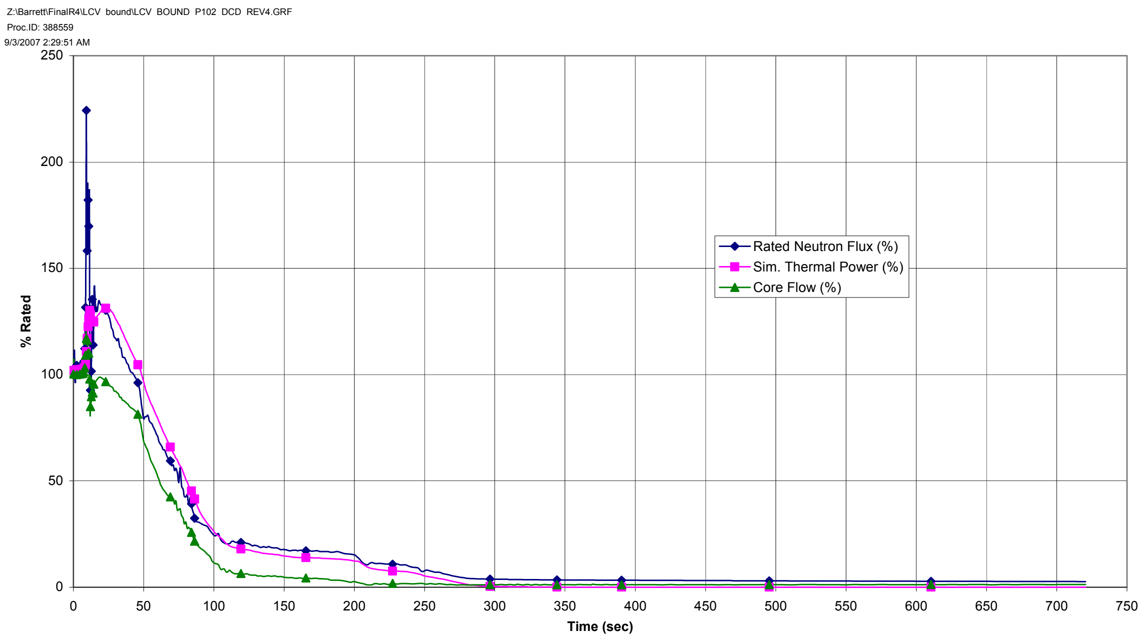


Figure 15.5-4b. Loss of Condenser Vacuum SLC System Bounding Case



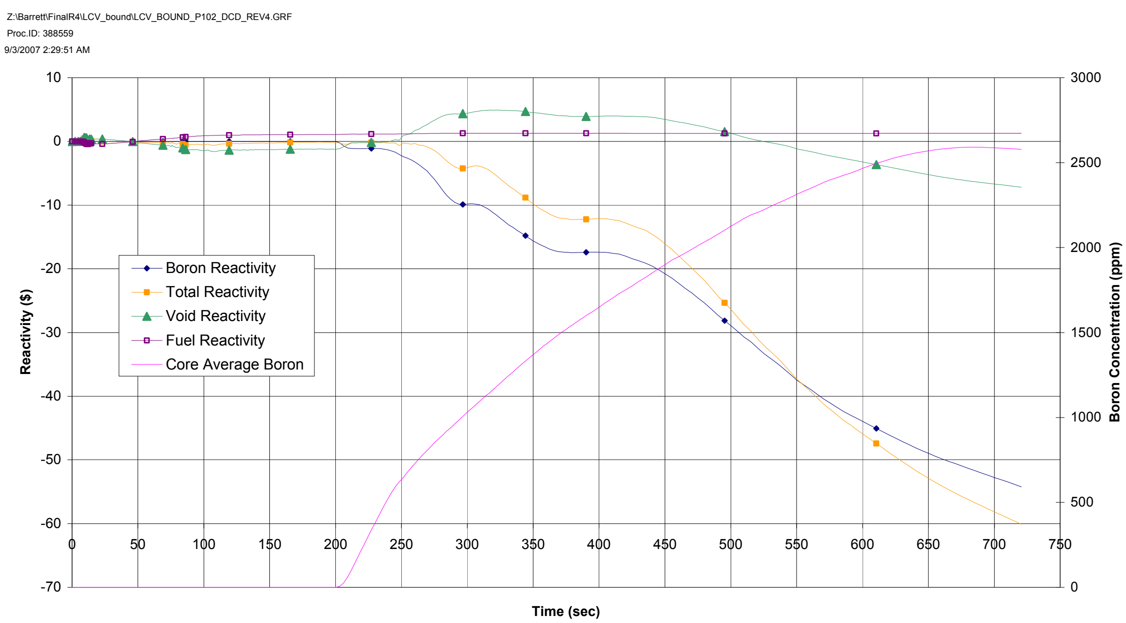


Figure 15.5-4c. Loss of Condenser Vacuum SLC System Bounding Case

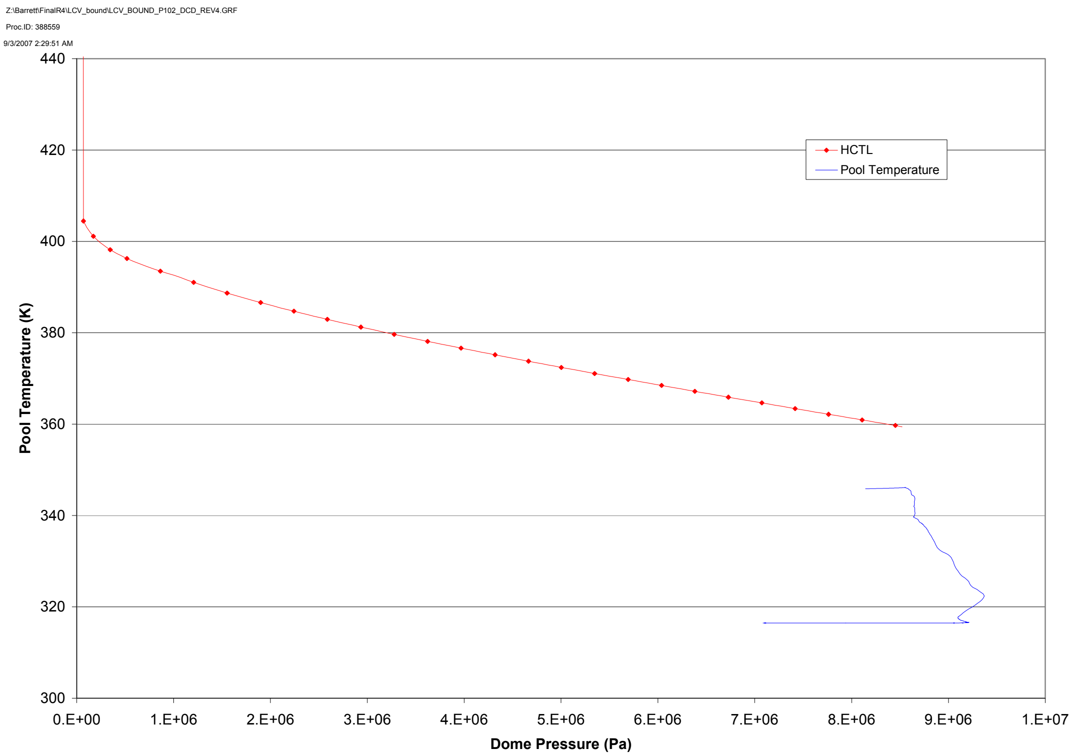


Figure 15.5-4d. Loss of Condenser Vacuum SLC System Bounding Case

Figure 15.5-4e. Loss of Condenser Vacuum SLC System Bounding Pool Temperature Case (Deleted)

Figure 15.5-4f. Loss of Condenser Vacuum SLC System Bounding Pool Temperature Case (Deleted)

Figure 15.5-4g. Loss of Condenser Vacuum SLC System Bounding Pool Temperature Case (Deleted)

Figure 15.5-4h. Loss of Condenser Vacuum SLC System Bounding Pool Temperature Case (Deleted)

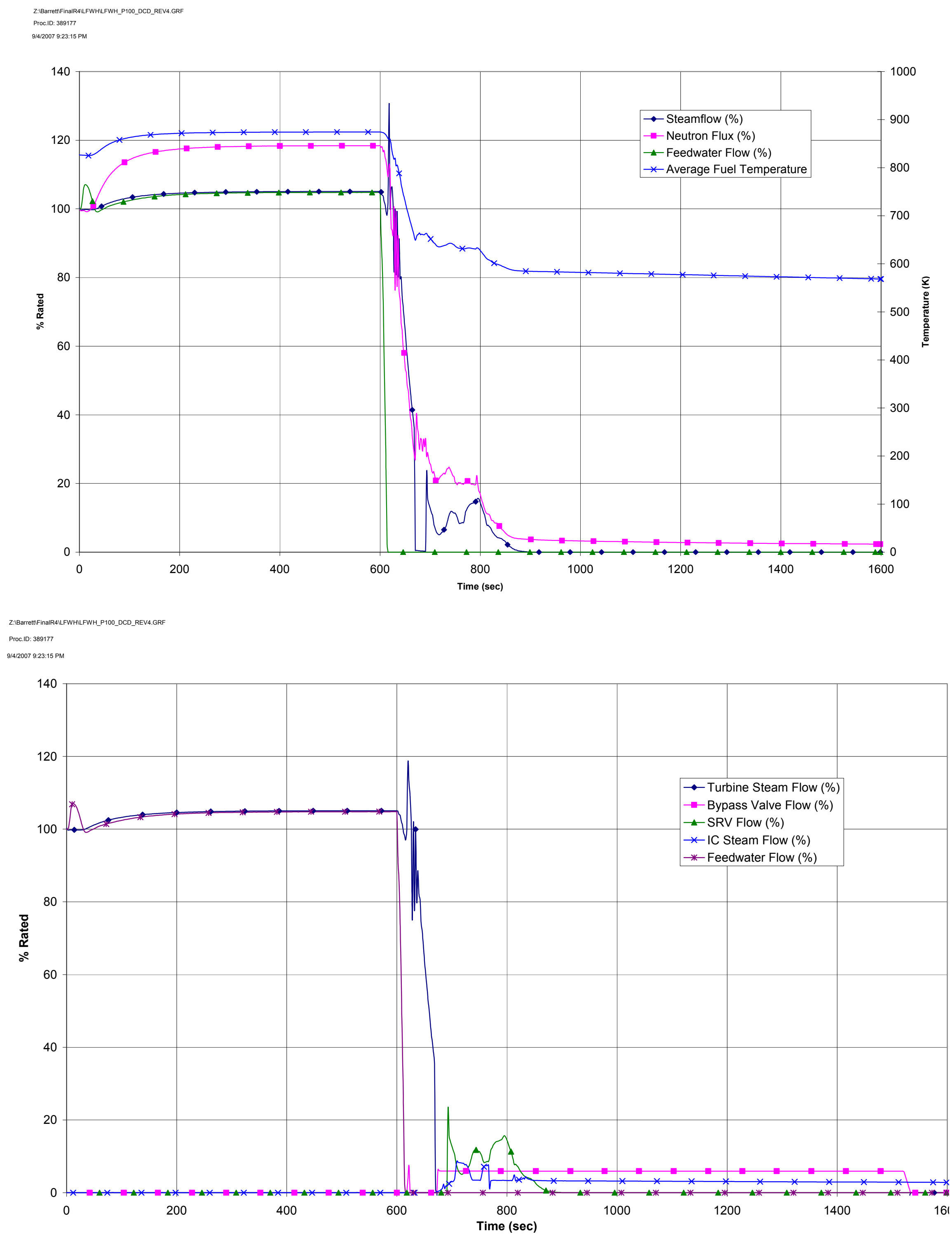
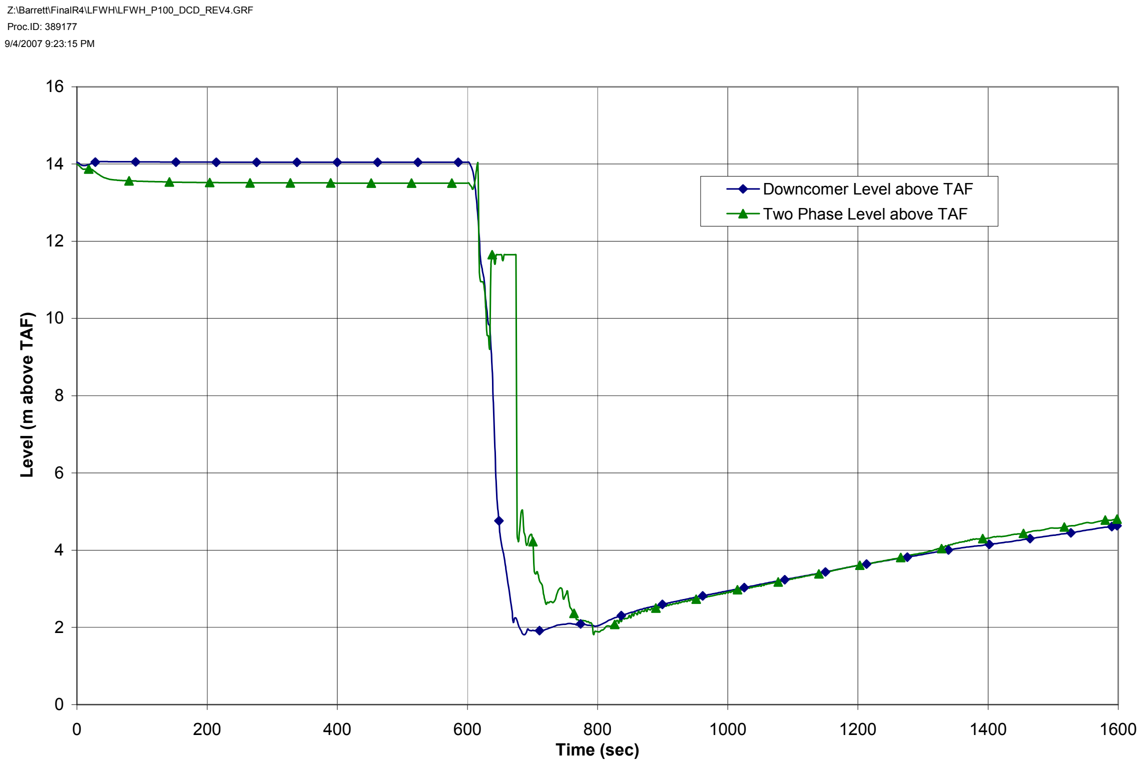


Figure 15.5-5a. Loss of Feedwater Heating with Boron Injection



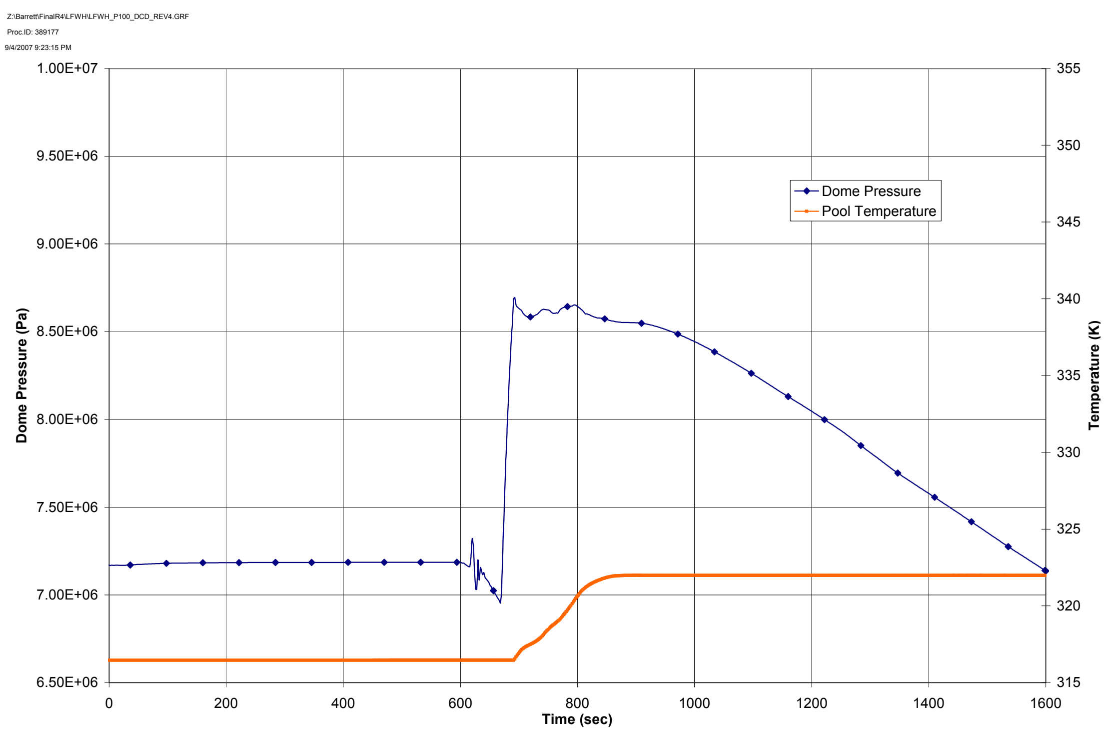
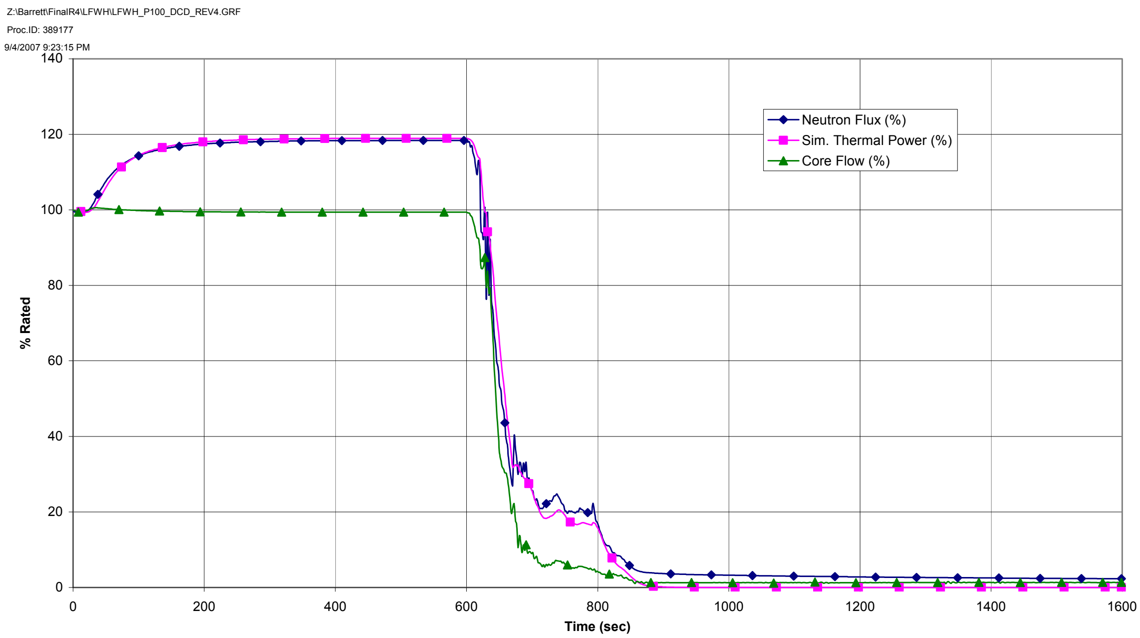


Figure 15.5-5b. Loss of Feedwater Heating with Boron Injection



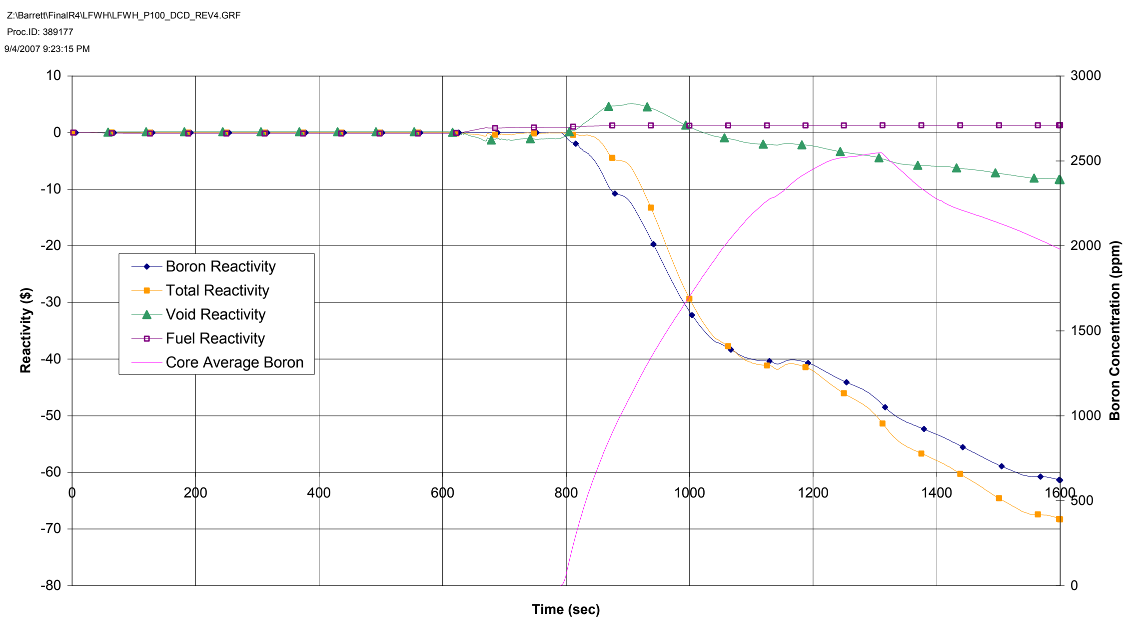


Figure 15.5-5c. Loss of Feedwater Heating with Boron Injection

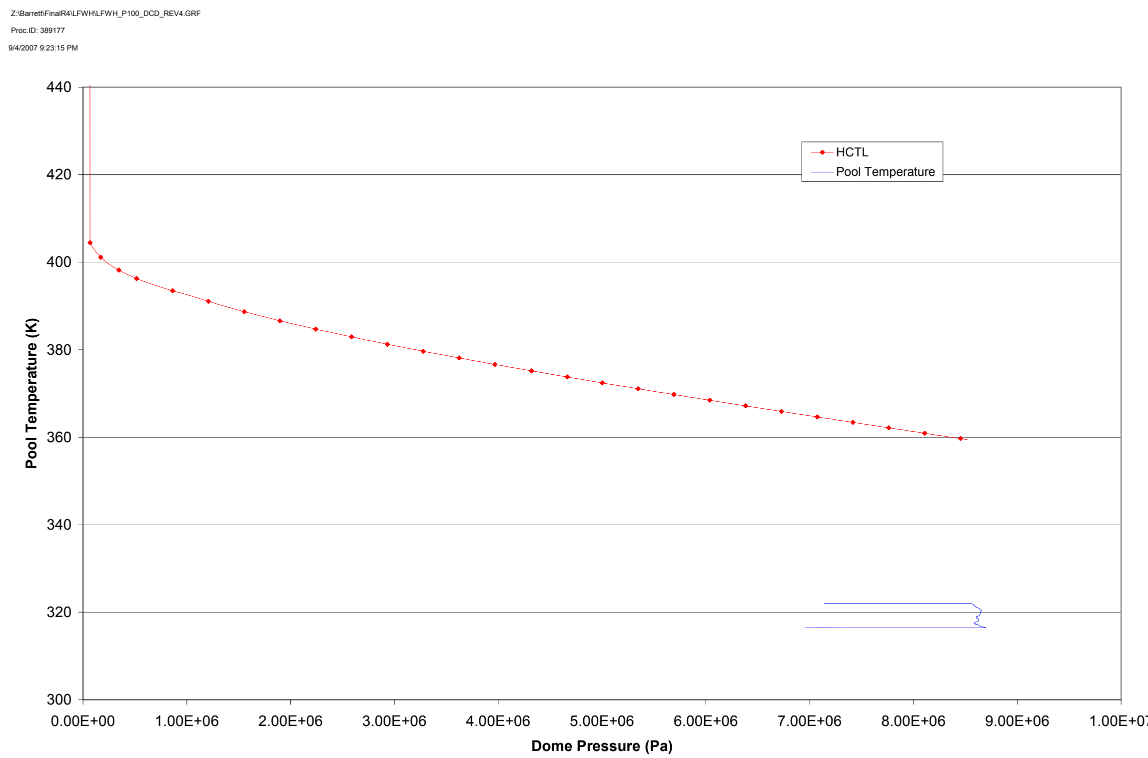
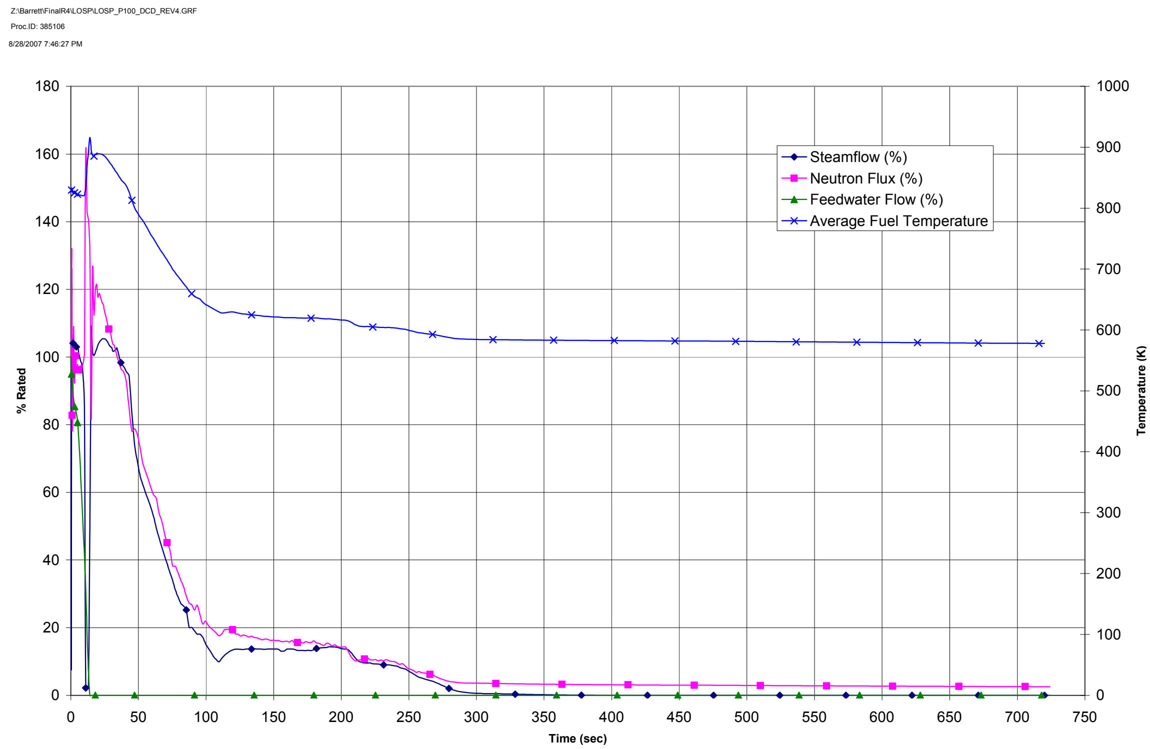


Figure 15.5-5d. Loss of Feedwater Heating with Boron Injection



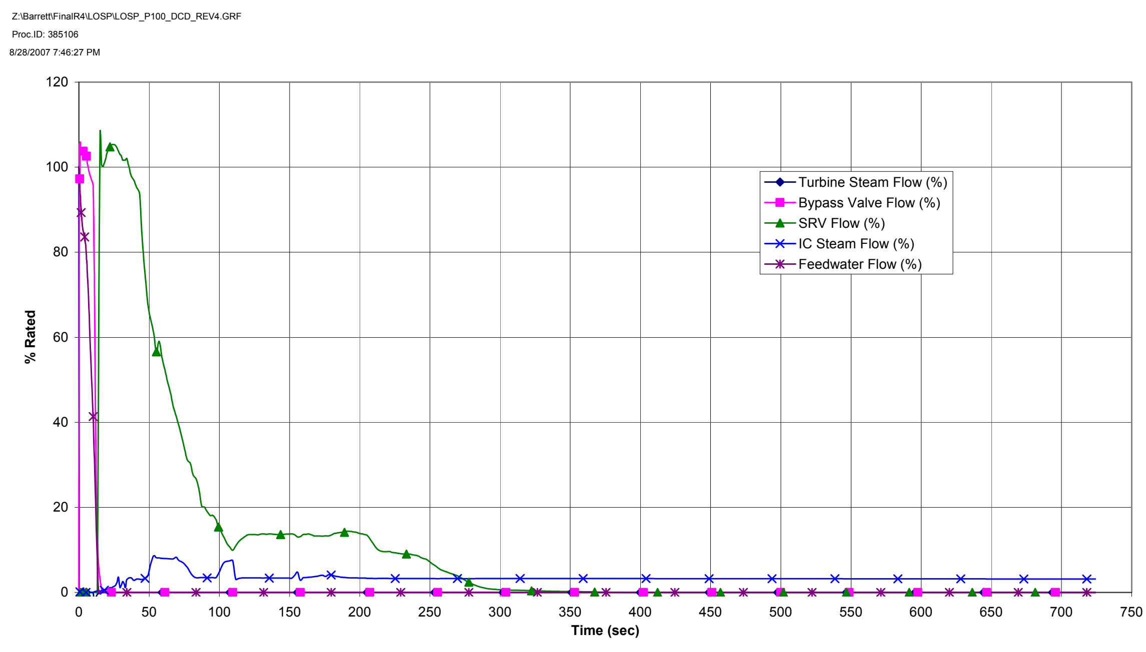
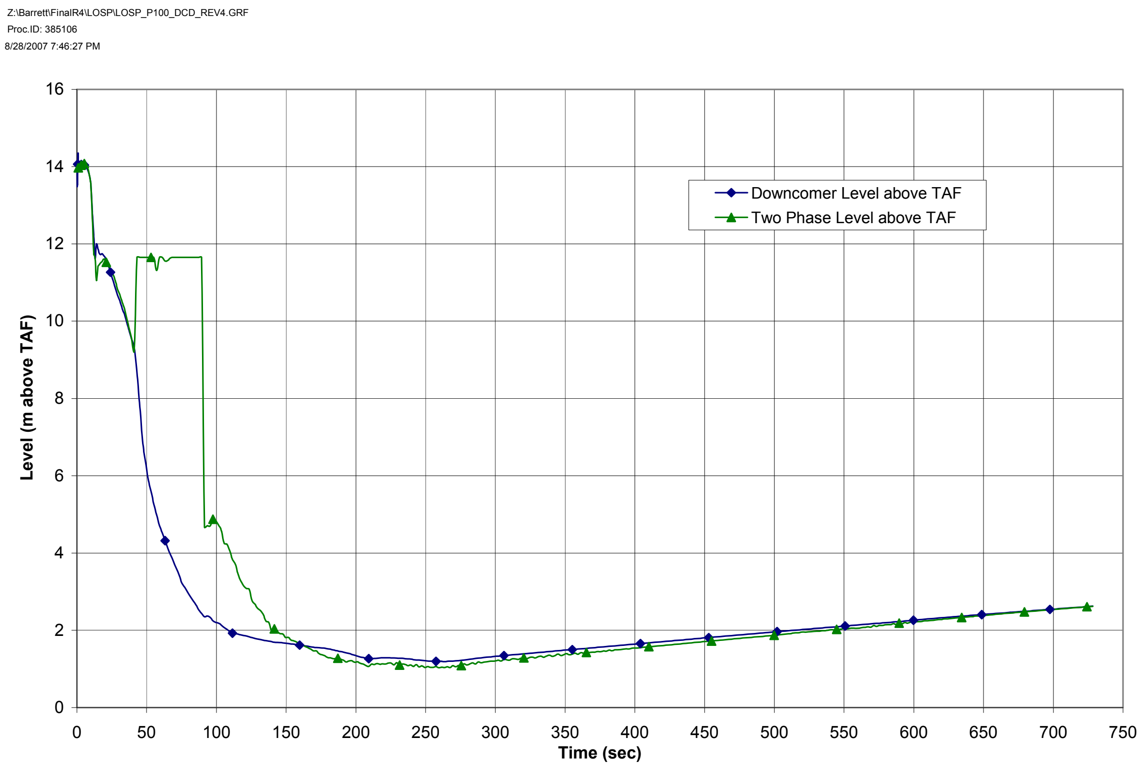


Figure 15.5-6a. Loss of Normal AC Power to Station Auxiliaries with Boron Injection



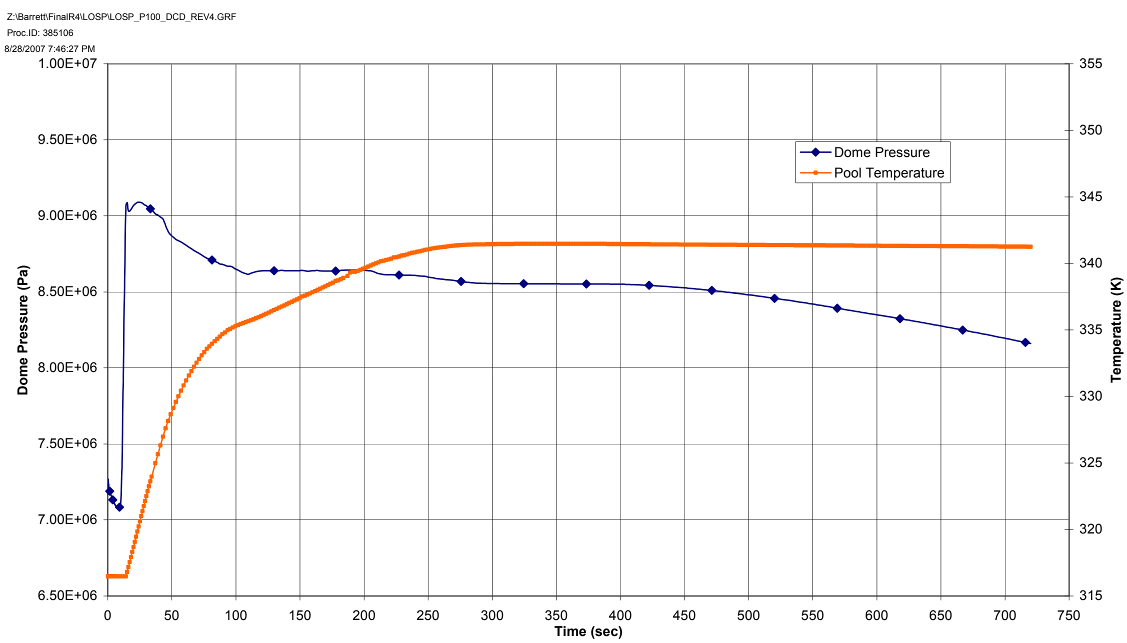
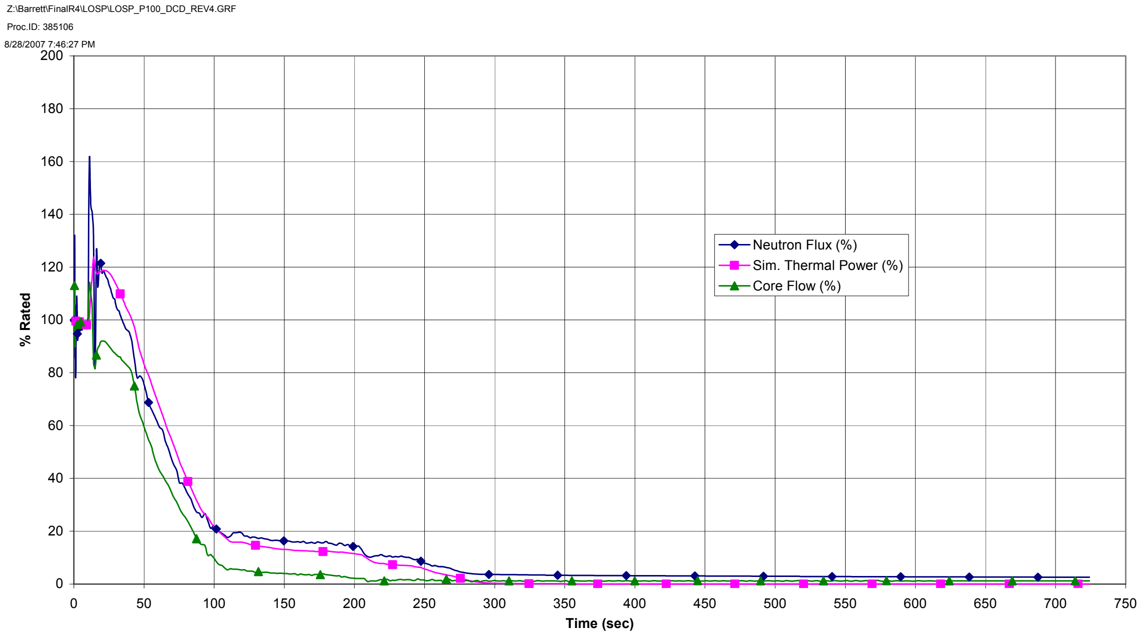


Figure 15.5-6b. Loss of Normal AC Power to Station Auxiliaries with Boron Injection



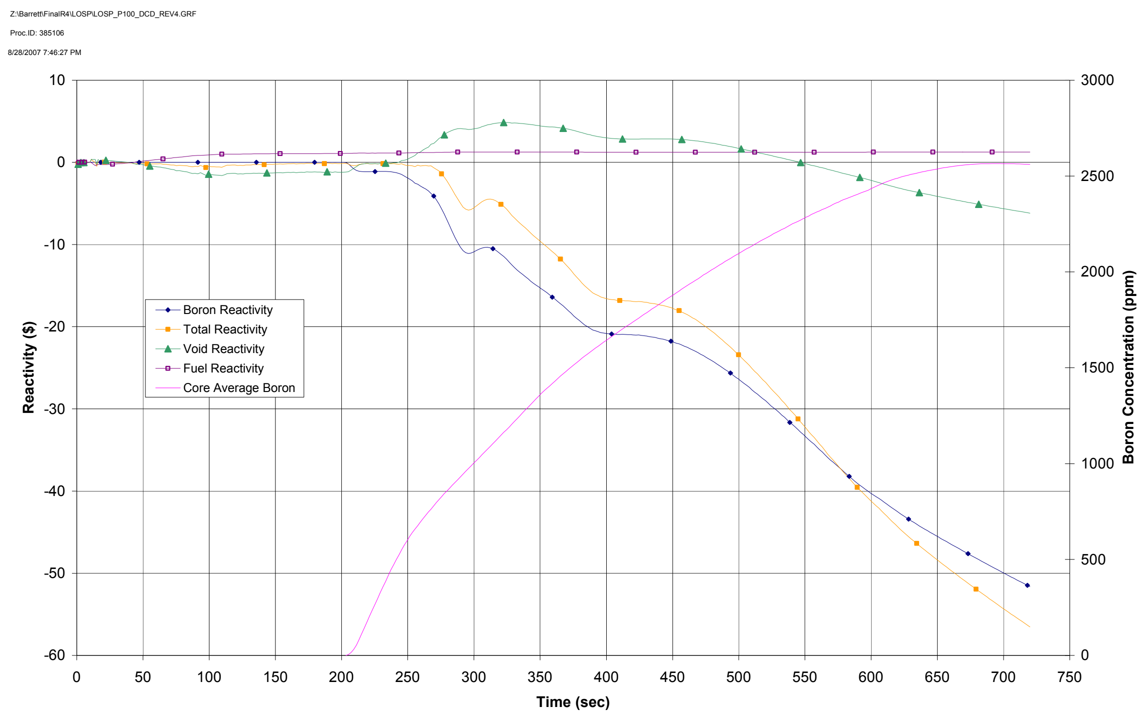


Figure 15.5-6c. Loss of Normal AC Power to Station Auxiliaries with Boron Injection

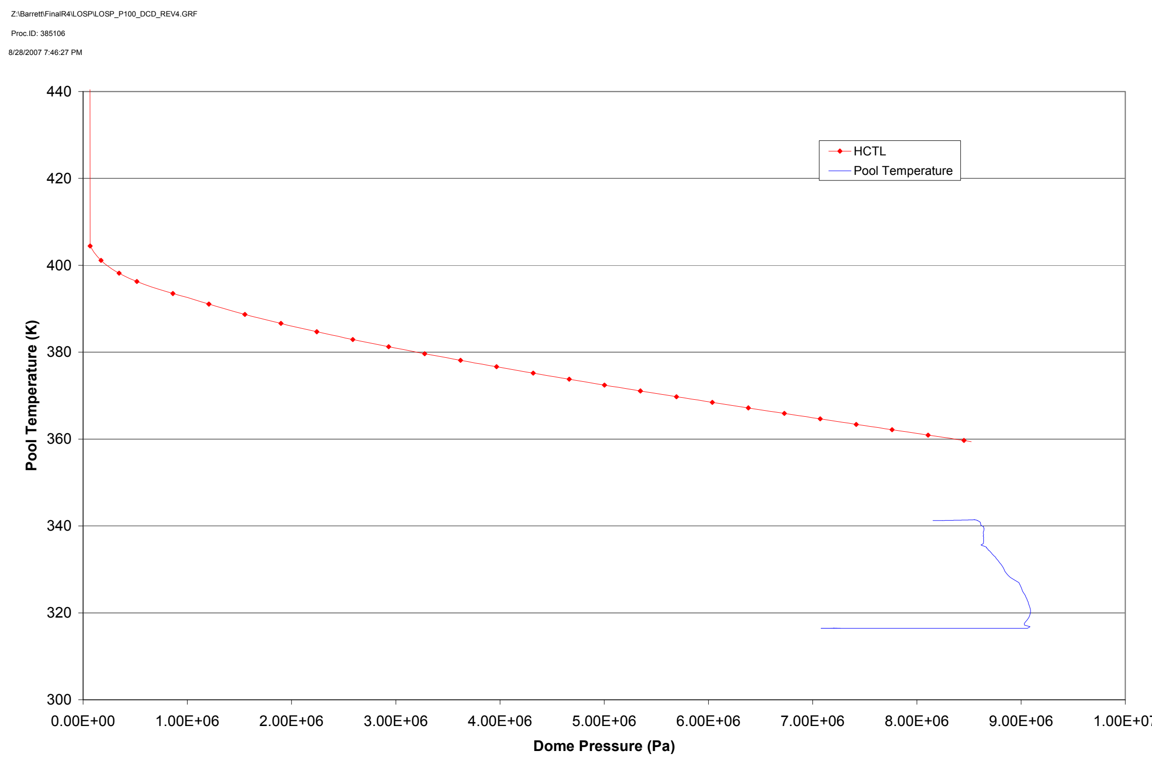
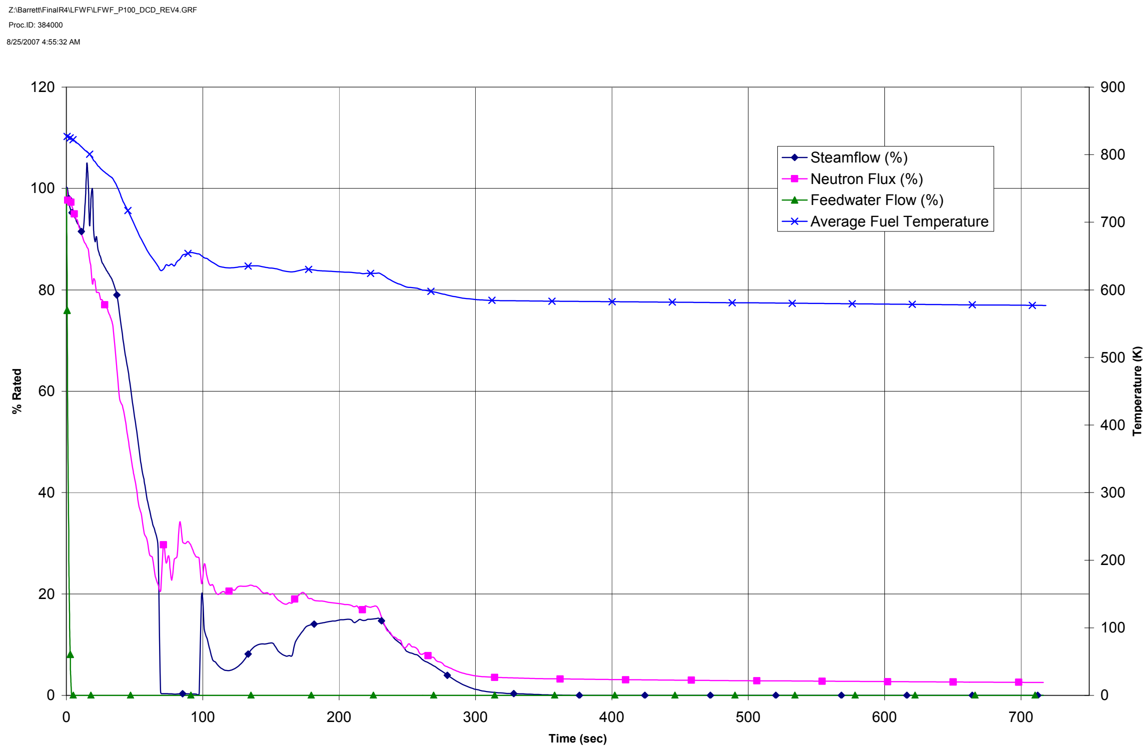


Figure 15.5-6d. Loss of Normal AC Power to Station Auxiliaries with Boron Injection



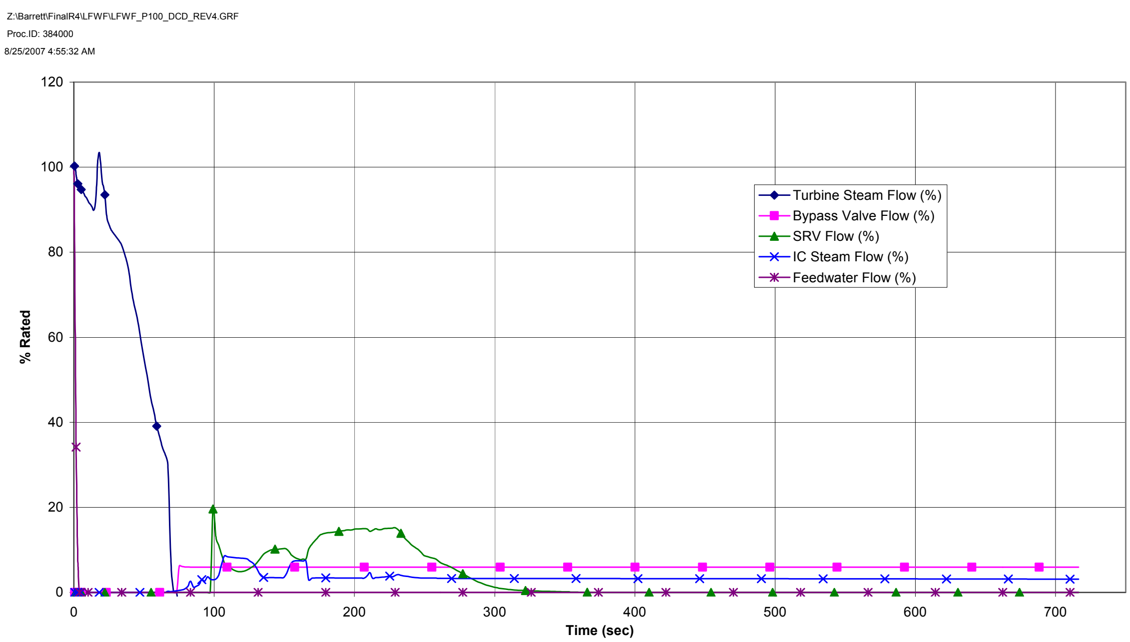
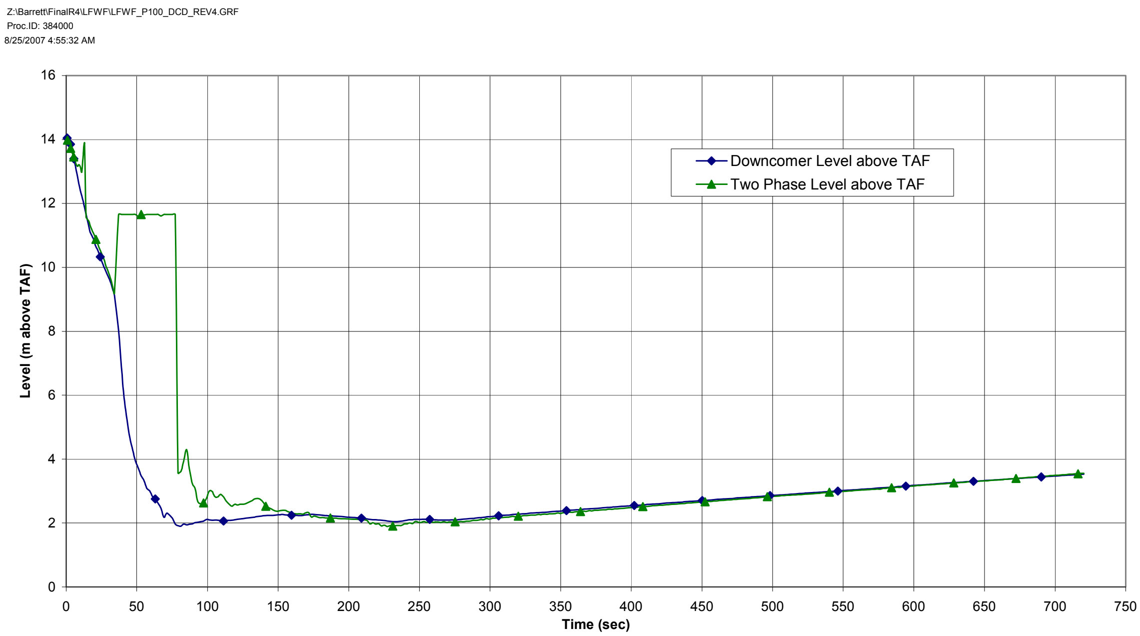


Figure 15.5-7a. Loss of Feedwater Flow with Boron Injection



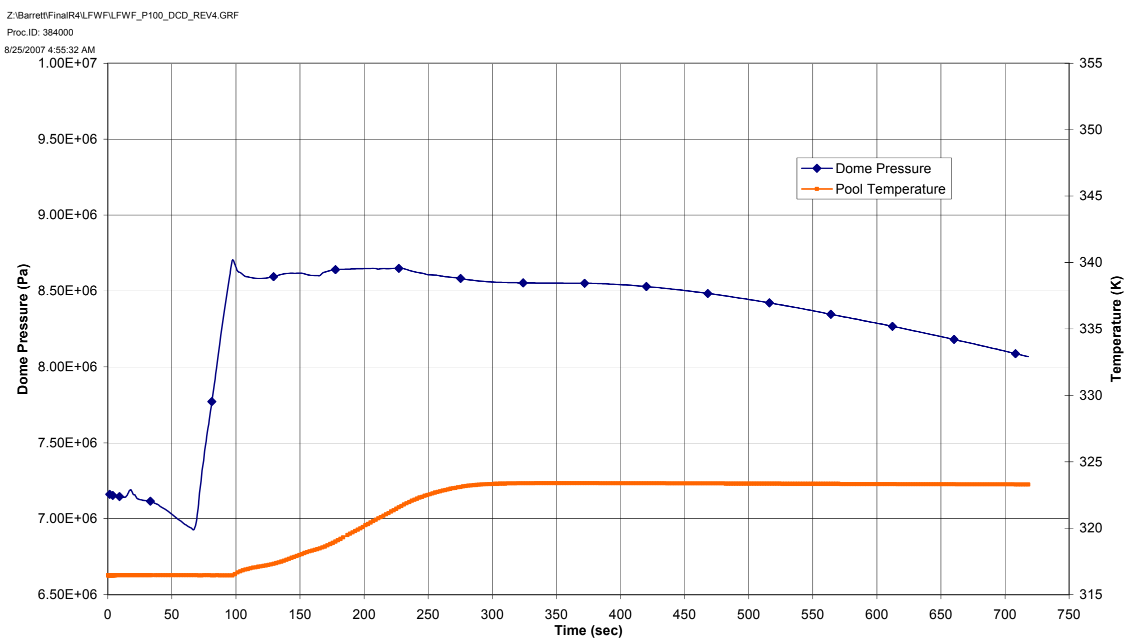


Figure 15.5-7b. Loss of Feedwater Flow with Boron Injection

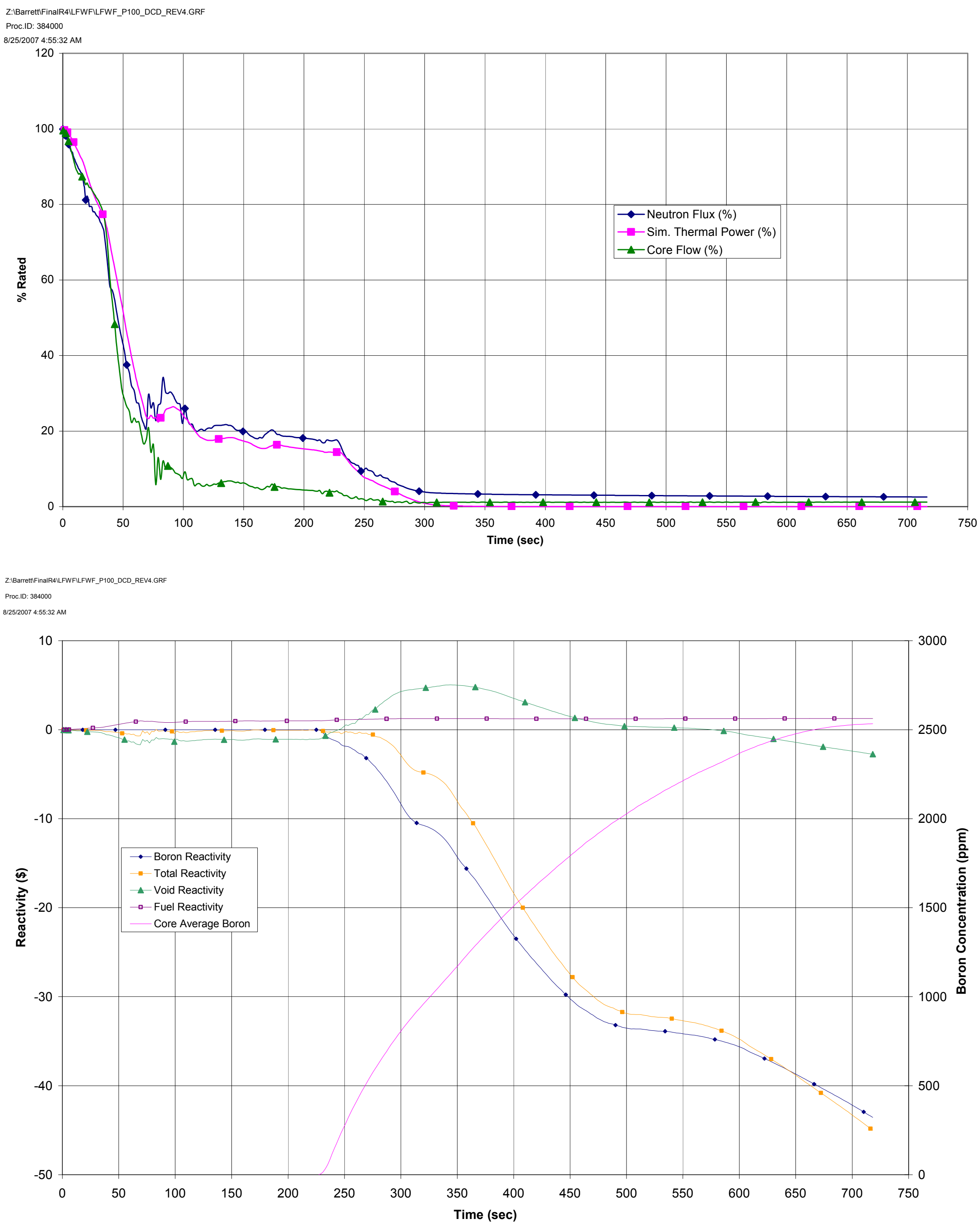


Figure 15.5-7c. Loss of Feedwater Flow with Boron Injection

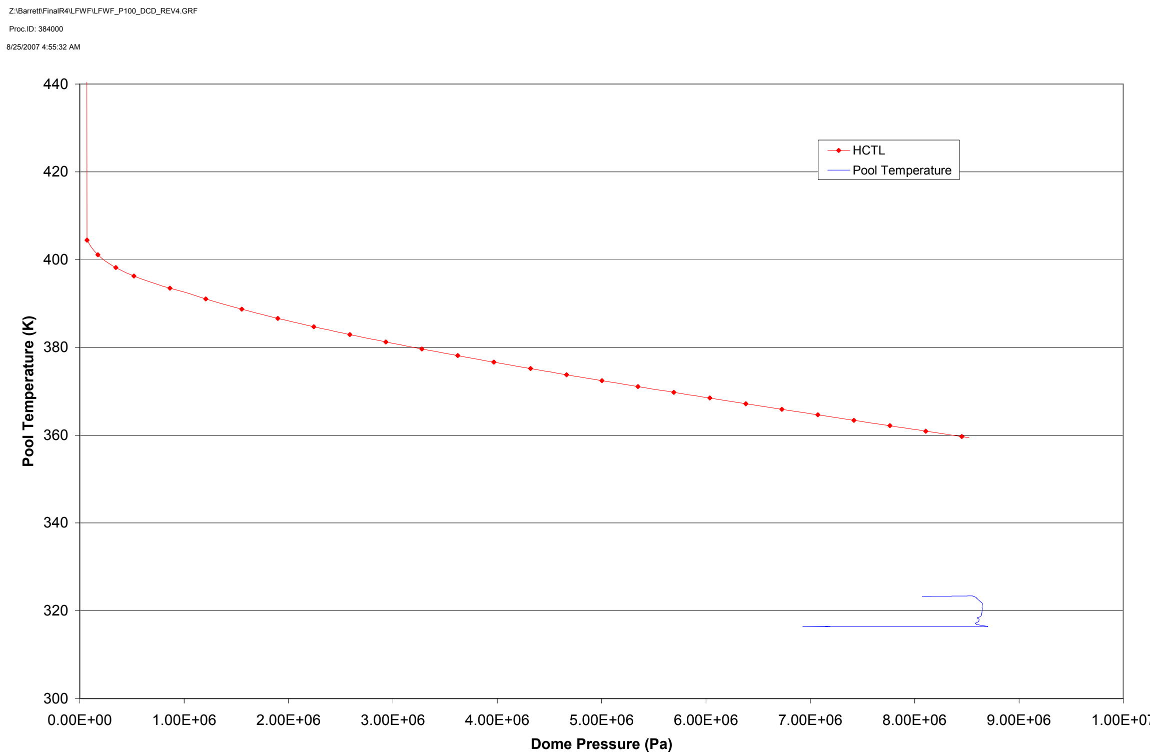
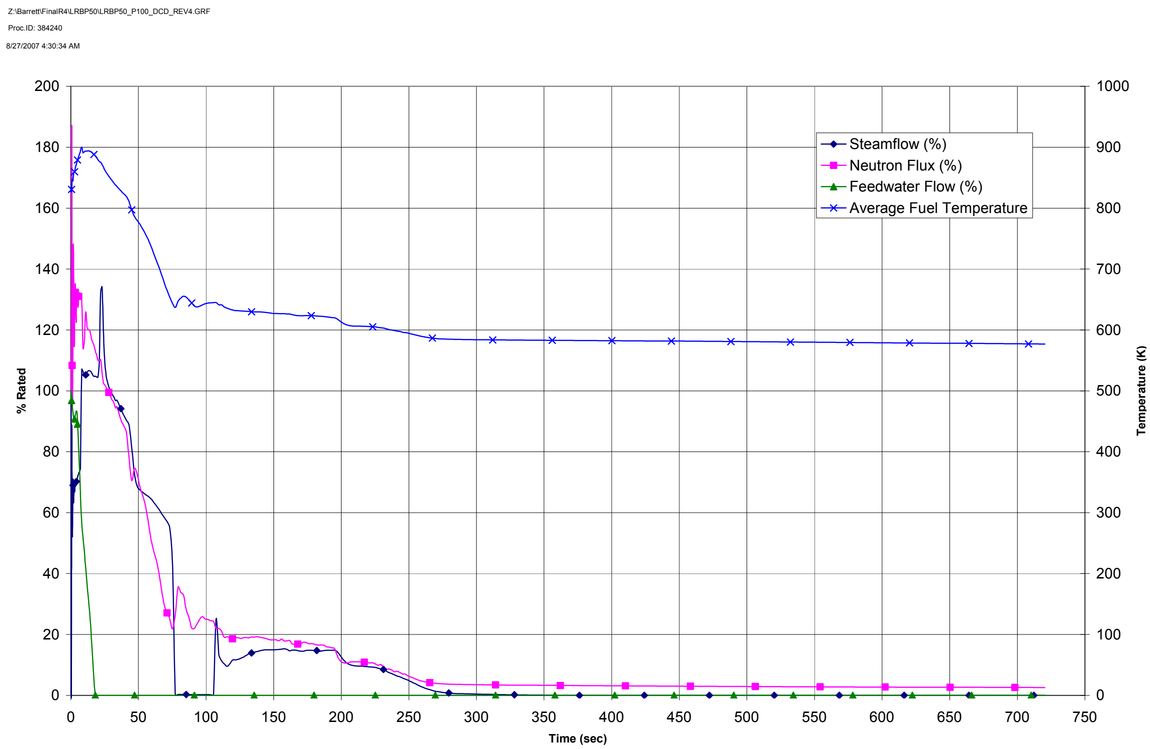


Figure 15.5-7d. Loss of Feedwater Flow with Boron Injection



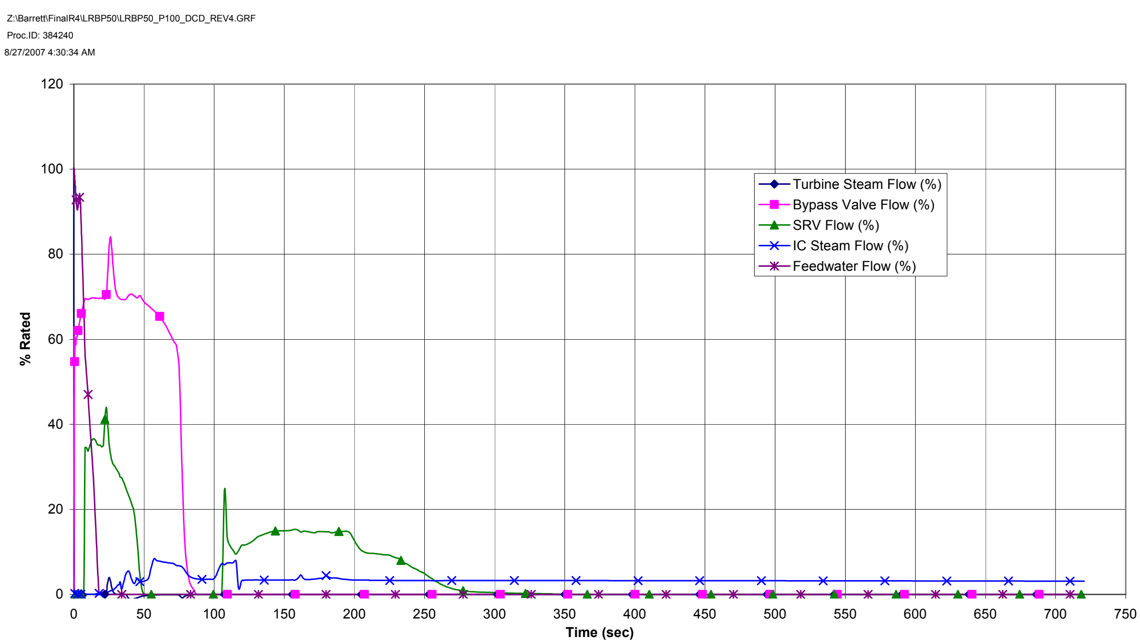


Figure 15.5-8a. Load Rejection with a Single Failure in the Turbine Bypass System with Boron Injection

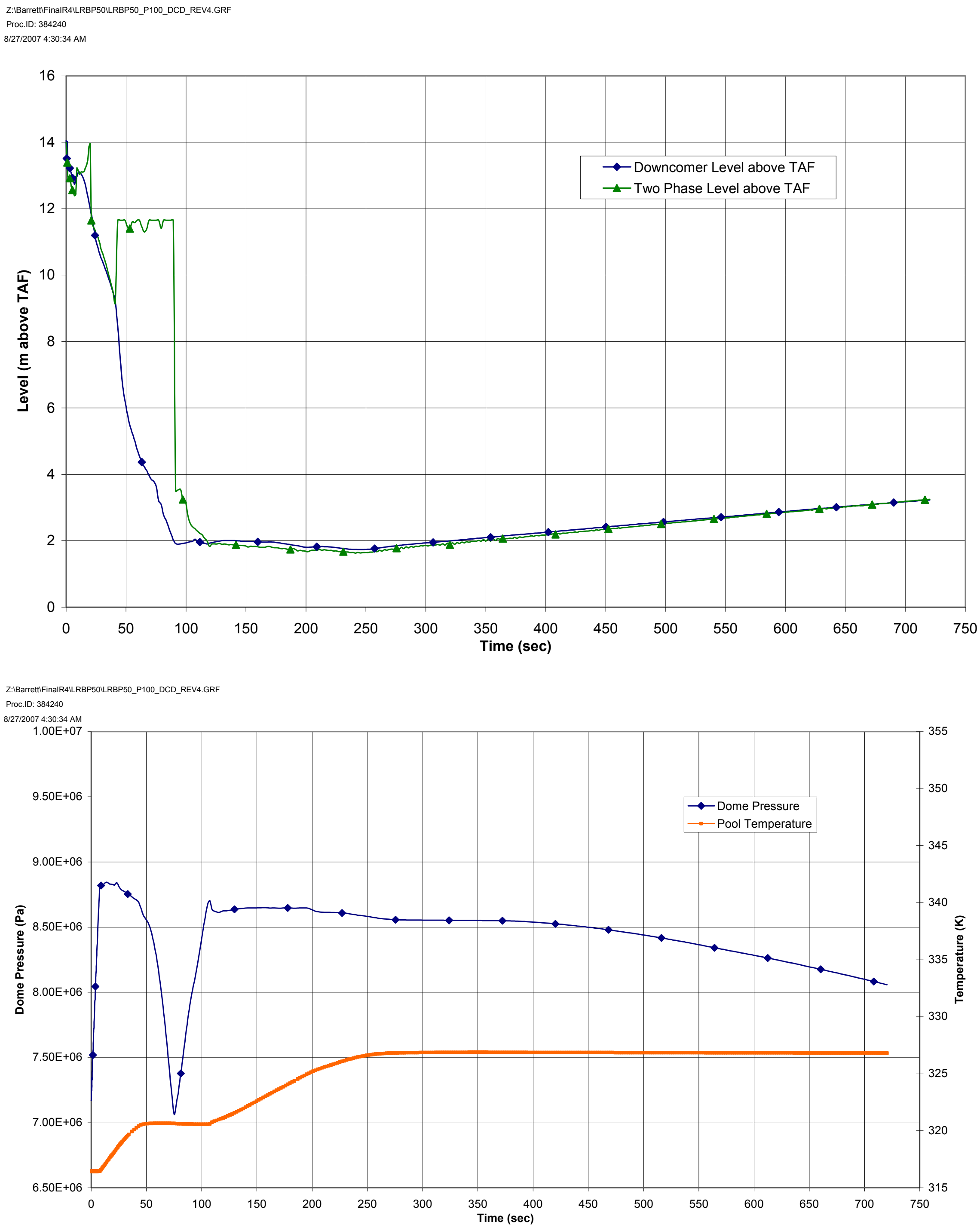
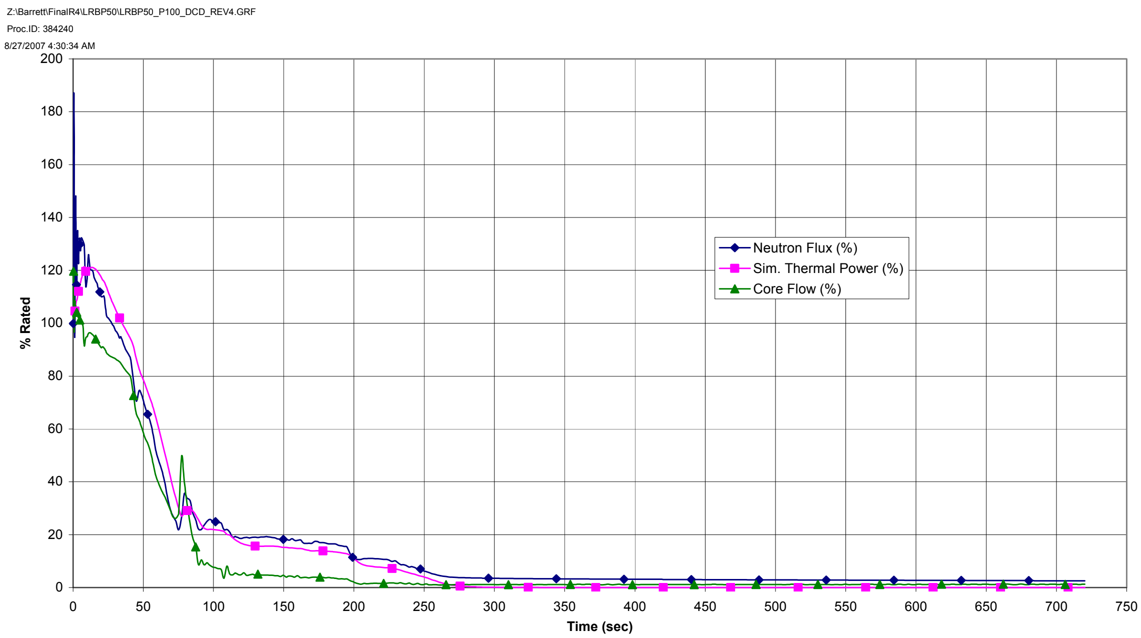


Figure 15.5-8b. Load Rejection with a Single Failure in the Turbine Bypass System with Boron Injection



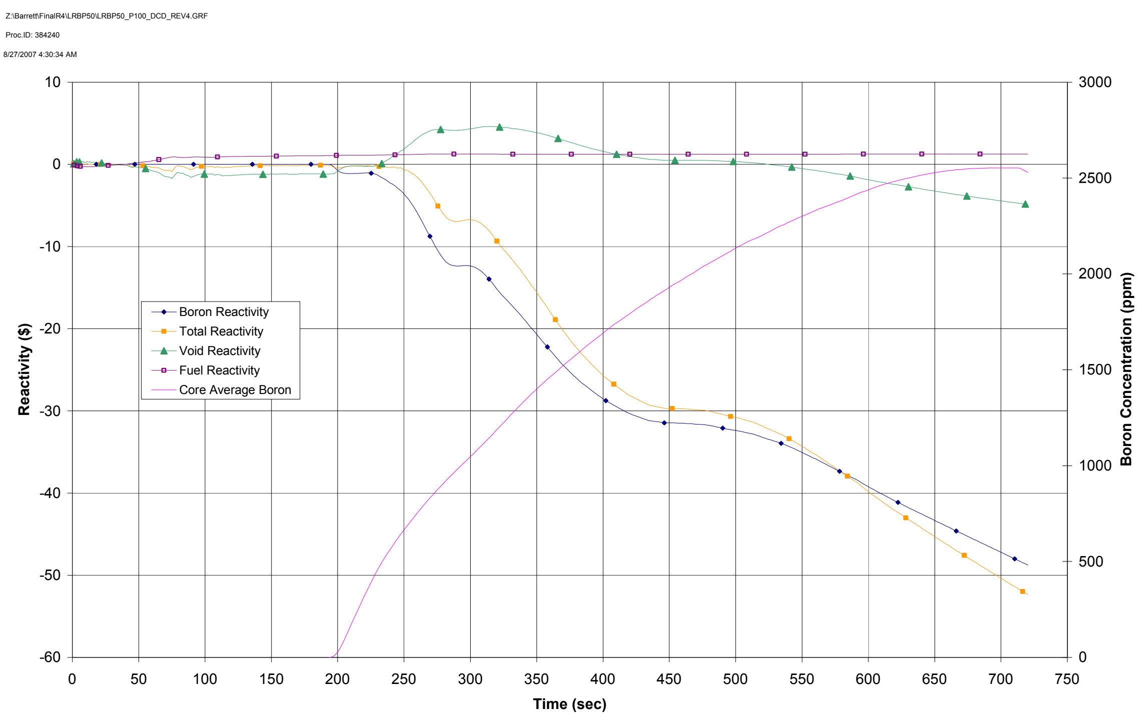


Figure 15.5-8c. Load Rejection with a Single Failure in the Turbine Bypass System with Boron Injection

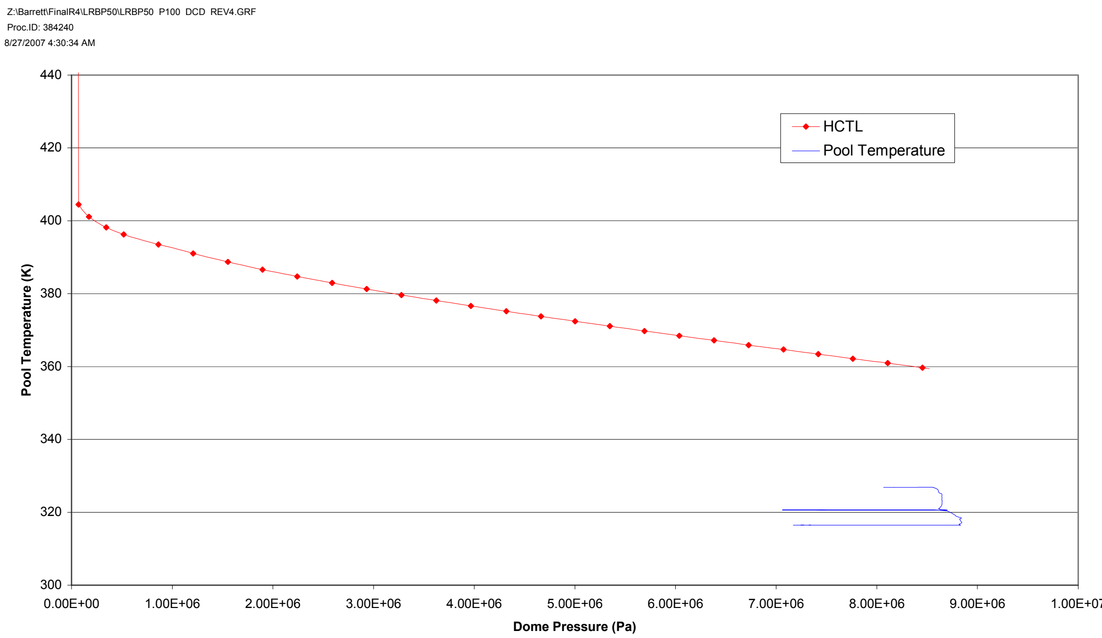


Figure 15.5-8d. Load Rejection with a Single Failure in the Turbine Bypass System with Boron Injection

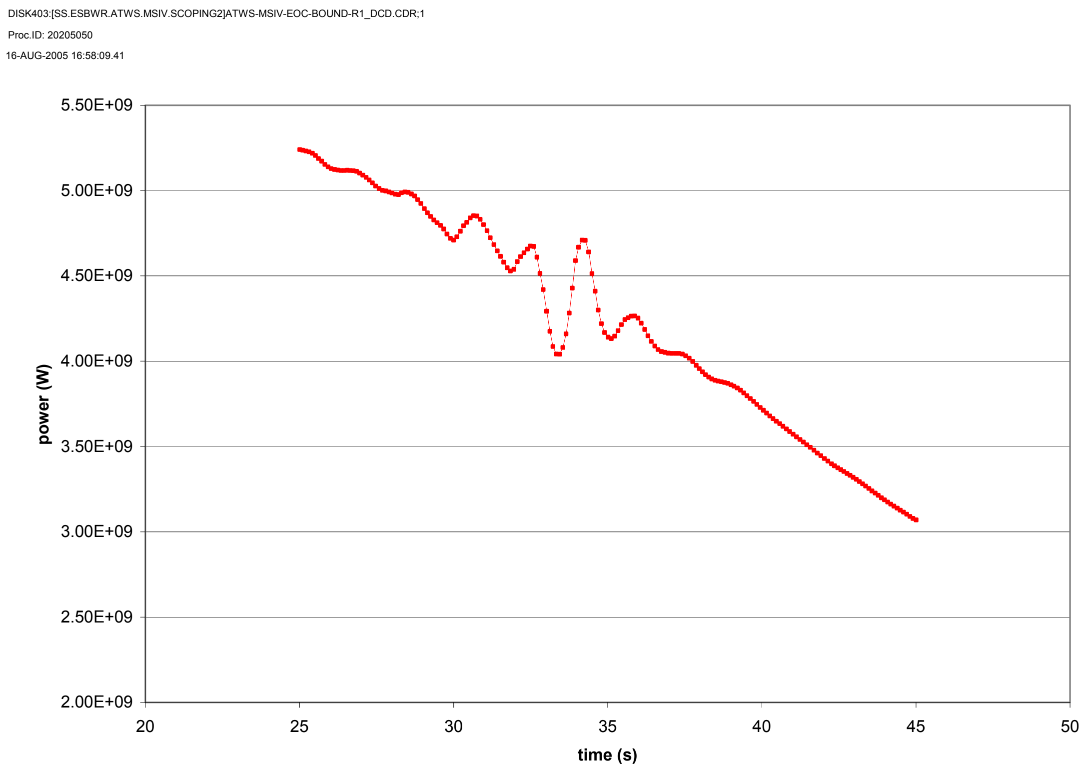


Figure 15.5-9. Core Stability during ATWS MSIV Closure Event

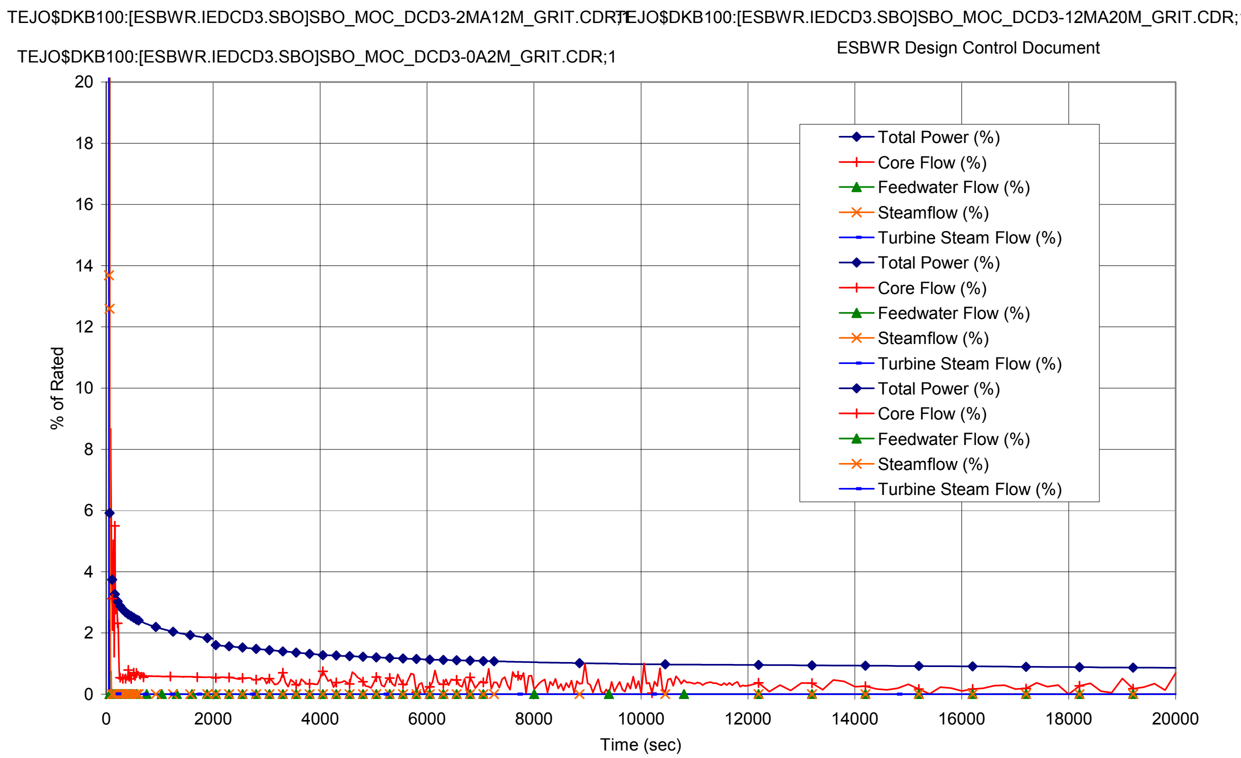


Figure 15.5-10a. Pressure Vessel Response for SBO

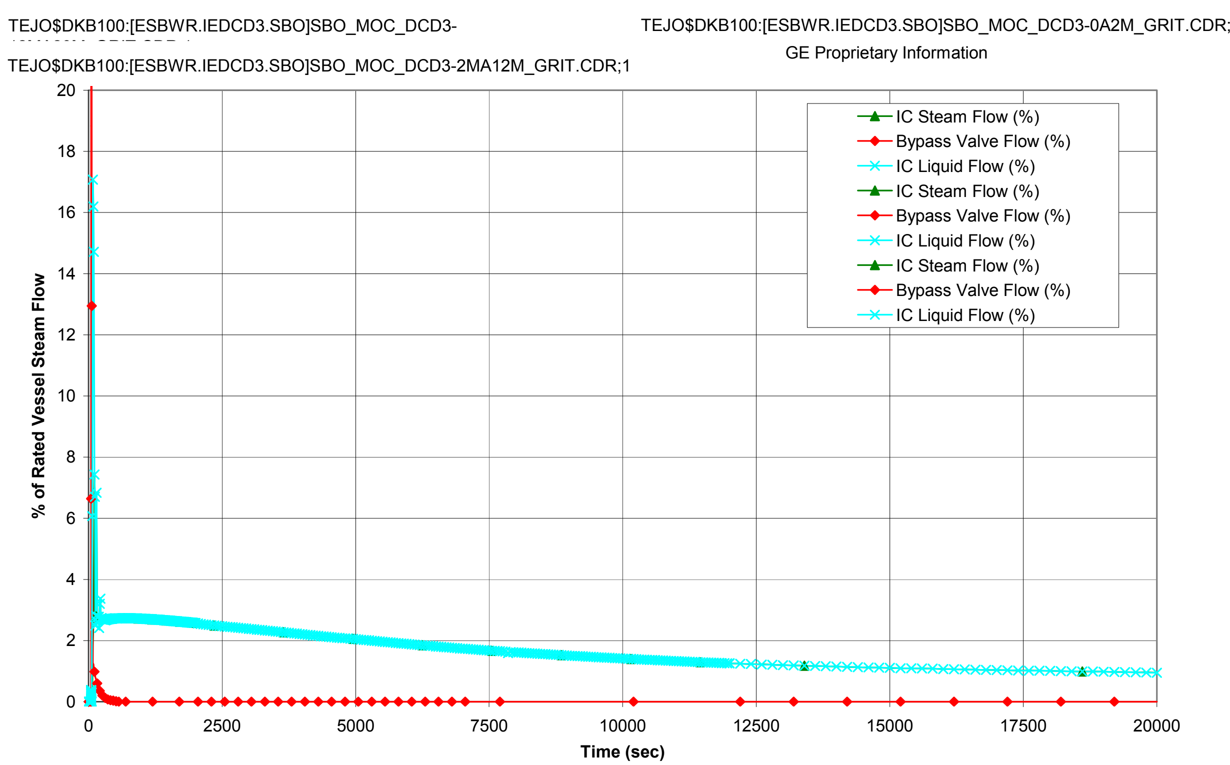


Figure 15.5-10b. Vessel inventory Makeup Flow Response for SBO

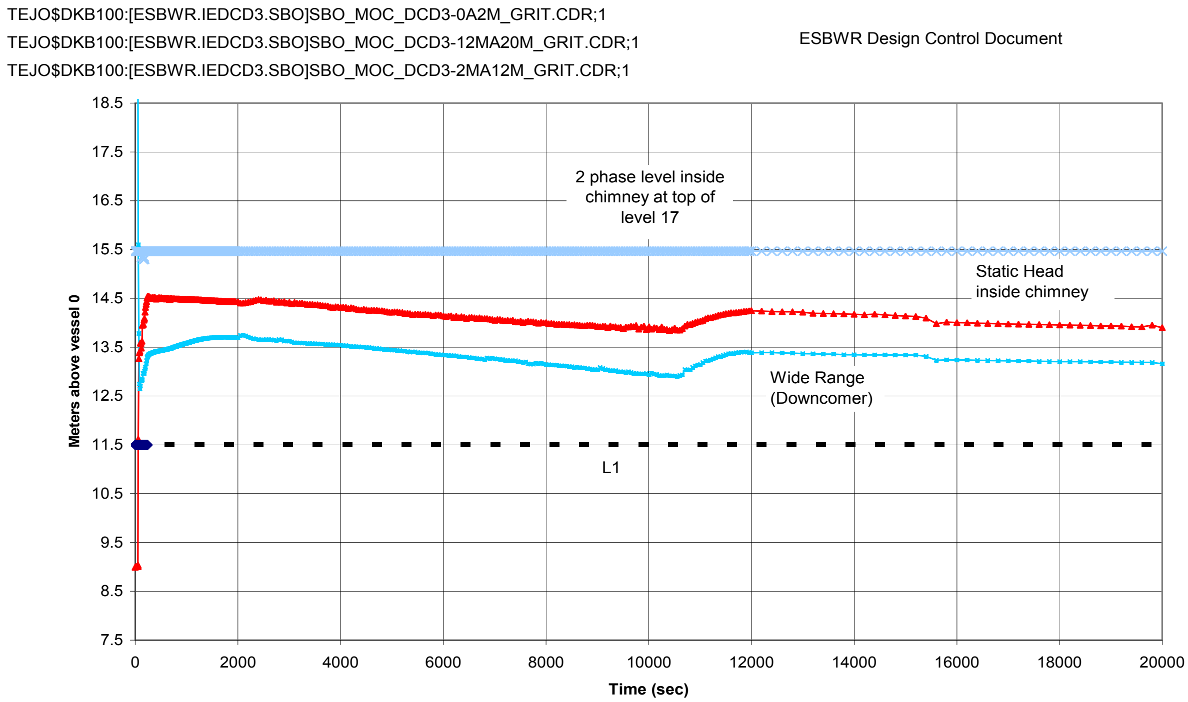


Figure 15.5-10c. Water Level Response for SBO

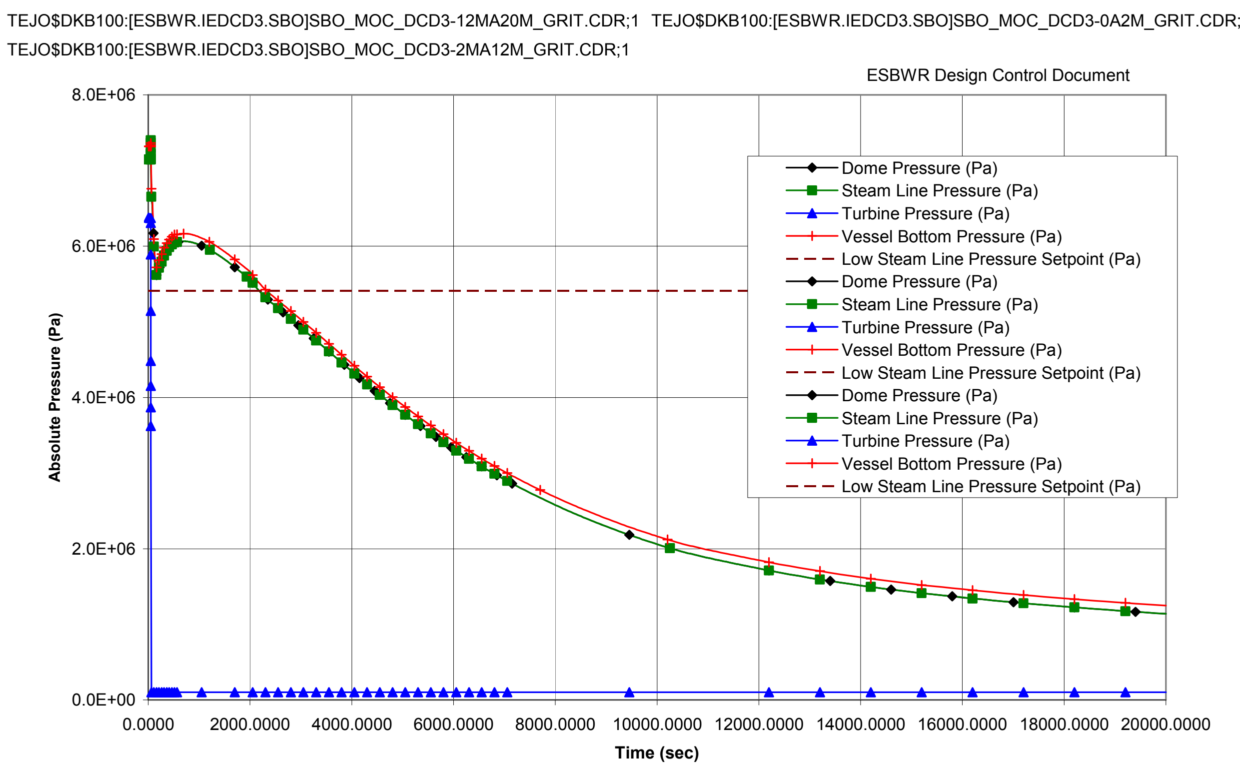


Figure 15.5-10d. Pressure Response for SBO

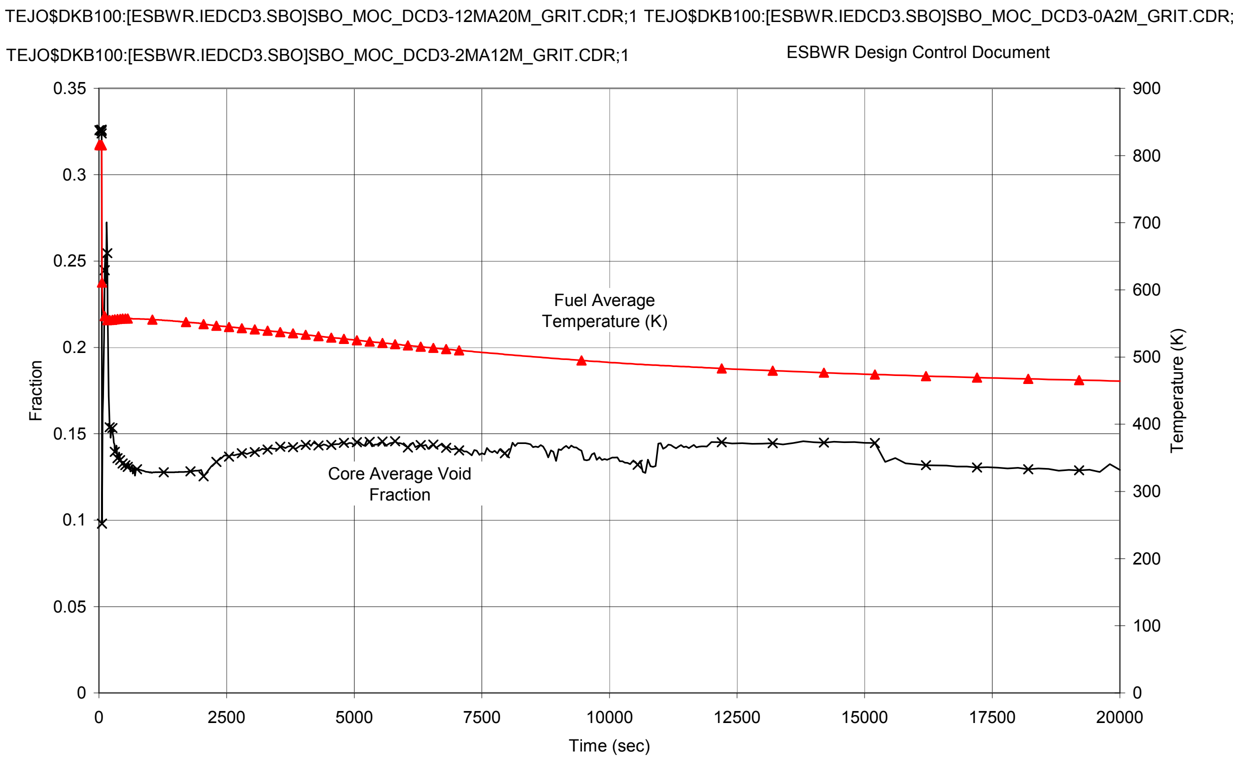


Figure 15.5-10e. Core void fraction and fuel temperature Response for SBO

15A. EVENT FREQUENCY DETERMINATION

15A.1 SCOPE

This Appendix provides the analysis to determine the frequency of occurrence of events classified as infrequent events in Table 15.0-7. The overall objective of this analysis is to determine the frequency of occurrence for these events, to allow them to be categorized as Anticipated Operational Occurrences or Infrequent Events. Events less frequent than 1 event in 100 years are classified as infrequent events.

15A.2 METHODOLOGY

The methodology used in this evaluation is based on industry established methods given in Probabilistic Risk Assessment (PRA) guidelines described in Reference 15A-1. The following types of analysis were applied in determining the event frequency:

- Where an initiating event is explicitly modeled in the ESBWR PRA, the frequency for this event is taken directly from the PRA. However, for some cases where more detail is required, additional analyses not given in the PRA were conducted. The frequencies of events that were not modeled in the PRA are addressed in this analysis.
- The event frequency is determined from actual BWR operating experience, modified to reflect the ESBWR improved design features. Where the analysis depends on specific assumed design features or testing, these features and tests are identified as ESBWR design requirements.
- Several events involve multiple independent hardware failures or human errors. For these events, the event frequency is based on conservative estimates of the hardware failures (including common cause failures) and human errors.

To account for any data or modeling uncertainties, the final event frequencies have been reviewed to ensure a factor of 3 times above the criterion for an infrequent event. This factor increase is consistent with current PRA practices dealing with uncertainties.

15A.3 RESULTS

The analysis for each event includes a description of the event, a discussion of the analysis used to determine the event frequency, and a summary of the results. The following subsections present the analysis results for each event.

15A.3.1 Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves

15A.3.1.1 Introduction

The Steam Bypass and Pressure Control (SB&PC) System controls the reactor pressure during plant operation. The SB&PC system controllers, which take input from the reactor dome pressure and other operating parameters, regulate the reactor pressure during normal operation by sending control signals to the Turbine Control Valves (TCVs). However, whenever the total steam flow demand from the SB&PC system exceeds the effective TCV steam flow demand, the SB&PC controllers send a signal to the Turbine Bypass Valves (TBVs) to open. While the SB&PC system is designed to a high degree of reliability, multiple failures in the system could lead to a failure of the controller in the upscale position, which would send a demand signal to all the TCVs and TBVs to open. Such an event is identified as the "Pressure Regulator Failure – Opening of All Turbine Control & Bypass Valves" event. The occurrence frequency of this event is evaluated in this subsection.

15A.3.1.2 Analysis

The description of the SB&PC system is provided in Subsection 7.7.5.

The SB&PC system is equipped with a triple-redundant, fault-tolerant digital controller (FTDC) including power supplies, and input/output signals. The FTDC consists of three parallel processing channels, each containing the hardware and software for execution of the control algorithms. The FTDC is designed to a high degree of reliability. Based on Subsection 7.7.5, the Mean Time to Failure (MTTF) of the SB&PC Controller is at least 1,000 years.

The actual reliability of the SB&PC controller is expected to be much better than the specified minimum MTTF requirement of 1,000 years. The controller can either fail high causing maximum demand or fail low causing minimum demand. Assuming that either failure mode is equally possible, the frequency of controller failing in a manner to cause maximum demand is estimated to be once in 2,000 years.

15A.3.1.3 Result

The frequency of pressure regulator failure – opening of all turbine control and bypass valves, is once in 2,000 years and therefore, the event frequency meets the criterion of being less than once in 100 years.

15A.3.2 Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves

15A.3.2.1 Introduction

The Steam Bypass and Pressure Control (SB&PC) System controls the reactor pressure during plant operation. The SB&PC system controllers, which take input from the reactor dome pressure and other operating parameters, regulate the reactor pressure during normal operation by sending control signals to the Turbine Control Valves (TCVs). However, whenever the total steam flow demand from the SB&PC system exceeds the effective TCV steam flow demand, the SB&PC controllers send a signal to the Turbine Bypass Valves (TBVs) to open. While the SB&PC system is designed to a high degree of reliability, multiple failures in the system could lead to a failure of the controller in the downscale position, which would send a demand signal to all the TCVs and TBVs to close. Should this occur, it would cause full closure of all TCVs as well as closure of any bypass valves that are open. Such an event is identified as the "Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves" event. The occurrence frequency of this event is evaluated in this subsection.

15A.3.2.2 Analysis

The description of the SB&PC system is provided in Subsection 7.7.5.

The SB&PC system is equipped with a triple-redundant, fault-tolerant digital controller (FTDC) including power supplies, and input/output signals. The FTDC consists of three parallel processing channels, each containing the hardware and software for execution of the control algorithms. The FTDC is designed to a high degree of reliability. Based on Subsection 7.7.5, the Mean Time to Failure (MTTF) of the SB&PC Controller is at least 1,000 years.

The actual reliability of the SB&PC controller is expected to be much better than the specified minimum MTTF requirement of 1,000 years. The controller can either fail high causing maximum demand, or fail low causing minimum demand. Assuming either failure mode is equally possible, the frequency of controller failing in a manner to cause minimum demand is once in 2,000 years.

15A.3.2.3 Result

The frequency of pressure regulator downscale failure – closing of all turbine control and bypass valves is once in 2,000 years, and therefore, the pressure regulator failure (maximum demand) event frequency meets the criterion of being less than once in 100 years.

15A.3.3 Turbine Trip with Total Bypass Failure

15A.3.3.1 Introduction

ESBWR is designed with 110% steam bypass capability, such that in case of a turbine trip event, the bypass valves open and send the steam to the main condenser thus avoiding a reactor trip. The bypass valves are part of the Turbine Bypass System (TBS) described in Subsection 10.4.4.

The TBS provides the capability to discharge main steam from the reactor to the condenser to minimize step load reduction transient effects on the Nuclear Boiler system. The TBS consists of twelve Turbine Bypass Valves (TBV) connected to the Turbine Main Steam System (TMSS) upstream of the turbine main stop valves. The outlets of TBVs are connected to the Main Condenser via piping with reducing assemblies.

15A.3.3.2 Analysis

Turbine trip without bypass requires first a turbine trip to occur, and this is followed by failure of sufficient number of TBVs to open on demand. Turbine trip frequency is obtained from the ESBWR PRA. The failure of TBVs to open on demand depends upon the control signal failure, mechanical failure of TBVs to open and support system failure. Each of these items is discussed below:

Turbine Trip Frequency: The frequency of Generic Transients, 1.18 per year is conservatively assumed to represent the turbine trip frequency. The value is taken from Table 2.3-3 of Reference 15A-2.

TBV Mechanical Failure: Subsection 15.2.2.5 documents the safety analysis of the turbine trip with a single failure in the turbine bypass system. This analysis shows that even with a failure of 50% of the bypass capability, the safety analysis results are within acceptable limits. This means that even with six of the twelve TBVs inoperable, results are within acceptable limits. Therefore, failure of seven or more out of twelve valves is considered unacceptable.

The ESBWR TBVs are designed to be significantly more reliable than the ones used in operating BWR and ABWR plants. The probability of random failure of seven valves, which involves the seventh power of a low number, is negligible compared to the common cause failure probability of seven valves. The common cause failure probability of seven valves is estimated by multiplying the individual TBV failure rate by a beta factor of 0.02. The value of 0.02 is judged to be a conservative value, especially since each valve is equipped with its own accumulator. Each group of 6 TBVs is actuated by hydraulic fluid from the main hydraulic lines. The hydraulic fluid for each group is isolated from other groups by check valves. If the hydraulic line for a particular group is lost for some reason, the accumulator for each of the TBVs is designed with sufficient capacity to open the associated valve for at least six seconds. TBVs with individual accumulators is a design improvement made for the ESBWR and makes these TBVs less susceptible to common cause failures compared to the TBVs in operating BWR plants. The common cause failure probability is 0.02 times the TBV failure rate, which yields 4.4E-4 per demand.

Signal Failure: The TBVs are controlled by triple redundant signals from the Steam Bypass and Pressure Control System. The TBVs receive redundant signals to open whenever the actual steam pressure exceeds the preset steam pressure by a small margin. This occurs when the amount of steam generated by the reactor cannot be entirely used by the turbine. Triple-redundant, fault tolerant digital controllers (FTDC) using triplicated feedback signals from the reactor vessel dome pressure sensors generate command signals for the TBVs and pressure regulation demand signals used by the Turbine Generator Control System (TGCS) to generate valve position demand signals for the TCVs. The signal (instrumentation and control) reliability is generally significantly better than the mechanical component reliability. Since the controllers are required for continued plant operation, there is a high probability that the controller is available following a turbine trip event. The signal failure is judged to be negligible.

Support System Failure: The only relevant support system is the AC power and loss of AC power results in a different category of initiating event. Therefore, the failure of AC power is not considered in this evaluation.

Operator action: Operator action is not needed for TBVs to open and operator action cannot cause a failure of TBVs to open on demand. Therefore, operator error is not considered in this evaluation.

The failure probability of TBVs is 4.4 E-4 per demand, based on the above discussion.

15A.3.3.3 Result

The frequency of turbine trip without bypass event is evaluated as a product of the turbine trip frequency (1.18 per year) and probability of failure of TBVs (4.4E-4 per demand).

The event frequency = (1.18)*(4.4E-4) = 5.19E-4 per year.

This translates to one event in over 1,900 years. Therefore, the event frequency meets the criterion of being less than once in 100 years.

15A.3.4 Generator Load Rejection with Total Turbine Bypass Failure

15A.3.4.1 Introduction

Following a load rejection, the Turbine Control Valves (TCVs) are commanded to close rapidly. At the same time the Steam Bypass and Pressure Control (SB&PC) System sends a signal to the Turbine Bypass Valves (TBVs) to open and throttle to maintain reactor pressure. The bypass valves are part of the Turbine Bypass System (TBS) described in Subsection 10.4.4. The SB&PC system is described in Subsection 7.7.5.

The TBS provides the capability to discharge main steam from the reactor to the condenser to minimize step load reduction transient effects on the Nuclear Boiler system. The TBS consists of twelve Turbine Bypass Valves (TBV) connected to the Turbine Main Steam System (TMSS) upstream of the turbine main stop valves. The outlets of TBVs are connected to the Main Condenser via piping with pressure reducing assemblies.

15A.3.4.2 Analysis

Generator load rejection with bypass failure event requires first a generator load rejection event to occur which is then followed by failure of sufficient number of TBVs to open on demand. The failure of TBVs to open on demand depends upon the control signal failure, mechanical failure of TBVs to open, and support system failure. Each of these items is discussed below.

Generator Load Rejection Frequency: The frequency of 0.45 per year is taken to represent the generator load rejection frequency. The value is taken from Table 9 of Reference 15A-5.

TBV Mechanical Failure: Subsection 15.2.2.5 documents the safety analysis of the turbine trip with a single failure in the turbine bypass system. This analysis shows that even with a failure of 50% of the bypass capability, the safety analysis results are within acceptable limits. This means that even with six of the twelve TBVs inoperable, results are within acceptable limits. Therefore, failure of seven or more out of twelve valves is considered unacceptable.

The ESBWR TBVs are designed to be significantly more reliable than the ones used in operating BWR and ABWR plants. . The probability of random failure of seven valves, which involves the seventh power of a low number, is negligible compared to the common cause failure probability of seven valves. The common cause failure probability of seven valves is estimated by multiplying the individual TBV failure rate by a beta factor of 0.02. The value of 0.02 is judged to be a conservative value, especially since each valve is equipped with its own accumulator. Each group of 6 TBVs is actuated by hydraulic fluid from the main hydraulic lines. The hydraulic fluid for each group is isolated from other groups by check valves. If the hydraulic line for a particular group is lost for some reason, the accumulator for each of the TBVs is designed with sufficient capacity to open the associated valve for at least six seconds. TBVs with individual accumulators is a design improvement made for the ESBWR and makes these TBVs less susceptible to common cause failures compared to the TBVs in operating BWR plants. The common cause failure probability is 0.02 times the TBV failure rate, which yields 4.4E-4 per demand.

Signal Failure: The TBVs are controlled by triple redundant signals from the Steam Bypass and Pressure Control System. The TBVs receive redundant signals to open whenever the actual steam pressure exceeds the preset steam pressure by a small margin. This occurs when the amount of steam generated by the reactor cannot be entirely used by the turbine. Triple-redundant, fault tolerant digital controllers (FTDC) using triplicated feedback signals from the reactor vessel dome pressure sensors generate valve position command signals for the TBVs and pressure regulation demand signals used by the Turbine Generator Control System (TGCS) to generate demand signals for the TCVs. The signal (instrumentation and control) reliability is generally significantly better than the mechanical component reliability. Since the controllers are required for continued plant operation, there is a high probability that the controllers are available following a turbine trip event. The signal failure probability is judged to be negligible.

Support System Failure: The only relevant support system is the AC power and loss of AC power results in a different category of initiating event. Therefore, the failure of AC power is not considered in this evaluation.

Operator action: Operator action is not needed for TBVs to open and operator action cannot cause a failure of TBVs to open on demand. Therefore, operator error is not considered in this evaluation.

The failure probability of TBVs is 4.4 E-4 per demand, based on the above discussion.

15A.3.4.3 Result

The frequency of generator load rejection with bypass failure is evaluated as a product of the generator load rejection frequency (0.45 per year) and probability of failure of TBVs (4.4E-4 per demand).

The event frequency = (0.45)*(4.4E-4) = 1.98E-4 per year.

This translates to one event in over 5,000 years. Therefore, the event frequency meets the criterion of less than once in 100 years.

15A.3.5 Feedwater Controller Failure - Maximum Demand

15A.3.5.1 Introduction

The Feedwater Control System (FWCS) is a power generation system, which is designed to maintain proper water level in the reactor during operation. The event of concern is one that results from one or more failures in the FWCS that causes multiple FW pumps to go to maximum output. This results in the feedwater pumps delivering a large amount of water, which increases the reactor water level to Level 8, at which time the feedwater pumps are tripped by an independent system, the main turbine is tripped and a reactor scram is initiated. Such an event is called the "Feedwater Controller Failure – Maximum Demand" event. The frequency of this event is evaluated in this subsection.

15A.3.5.2 Analysis

The description of the FWCS is provided in Subsection 7.7.3.

The FWCS is designed to maintain proper reactor pressure vessel water level in the operating range from high water level (Level 9) to low water level (Level 2). During normal operation, feedwater flow is delivered to the reactor vessel through three Reactor Feedpumps (RFPs), which operate in parallel. Each RFP is driven by an induction motor that is controlled by an adjustable speed drive (ASD). The fourth RFP is in standby mode and auto-starts if any operating feedpump trips while at power.

The FWCS is equipped with a triple-redundant, fault-tolerant digital controller (FTDC) including power supplies, and input/output signals. The FTDC consists of three parallel processing channels, each containing the hardware and software for execution of the control algorithms.

The FTDC is designed to a high degree of reliability. Based on Subsection 7.7.3, the Mean Time to Failure (MTTF) of the Feedwater System Controller is at least 1,000 years.

The actual reliability of the Feedwater controller is expected to be much higher than the specified minimum MTTF requirement of 1,000 years. It is assumed that the feedwater controller can fail high or fail low with equal probability. If any one of the three controllers fails either high causing maximum demand (or fails low causing minimum demand), the other two controllers would continue to function and the frequency of two or three controllers failing in a manner to cause maximum demand is once in 2,000 years.

15A.3.5.3 Result

The frequency of the feedwater controller failing in a manner to cause maximum demand of feedwater is less than once in 2,000 years and therefore, the event frequency meets the criterion of being less than once in 100 years.

15A.3.6 Loss of Feedwater Heating with Failure of SCRRI and SRI

15A.3.6.1 Introduction

The loss of feedwater heating causes the feed temperature to go down which increases the reactivity level. The ESBWR is designed such that the loss of feedwater results in insertion of selected control rods, so the reactivity level is adjusted appropriately. The failure of feedwater heating followed by the failure of the selected control rods to insert is the event of concern. The assessment of the mitigation capability given a loss of FW heating is estimated based on the failure of either of two functions: SCRRI or SRI. Failure of either method for inserting control rods would fail the mitigation function.

15A.3.6.2 Analysis

The loss of feedwater heating can occur at any given time during normal power range operation (e.g. the feedwater heaters are not operational during low power startup/shutdown conditions when the main turbine is not operational). When this event happens, the feedwater temperature goes down. This is detected by redundant temperature sensors in the feedwater piping lines that lead to the reactor pressure vessel that provide input signals to the Feedwater Control System (FWCS). The description of the FWCS is provided in Subsection 7.7.3. The primary purpose of the FWCS is to maintain proper reactor pressure vessel water level in the operating range. In addition, the FWCS sends signals to the Rod Control and Information System (RC&IS), via the Nonsafety-Related DCIS equipment, to insert selected control rods to mitigate the consequence of the loss of feedwater heating event. The description of the RC&IS is provided in Subsection 7.7.2. The RC&IS equipment in the control room back panel area sends signals to individual control rod logic implemented in the local RC&IS equipment in the reactor building in order to complete the run-in of each of the selected control rods to its associated, pre-defined Selected Control Rod Run-In and Select Rod Insertion (SCRRI/SRI) target position.

The RC&IS is equipped with dual-redundant, digital controller equipment including power supplies, and input/output signals. The design consists of two parallel processing channels, each containing the hardware and software for execution of the control algorithms. The controller equipment is designed to a high degree of reliability.

The RC&IS is also equipped with an Emergency Rod Insertion Control Panel and two associated Emergency Rod Insertion Panels that provide a parallel, redundant set of hardwired-based relay logic that also receives redundant signals from the FWCS (via the Nonsafety-Related DCIS equipment) when the loss of feedwater heating event is detected. For each control rod, one hardwired signal is provided from an associated Emergency Rod Insertion Panel to the individual control rod logic equipment that also must be activated in order for the individual local control rod logic to accomplish the selected control rod run-in of that control rod.

Therefore, in order to accomplish the selected control rod run-in of an individual control rod, the dual-channel logic must send the required command signals to the individual local control rod logic; and the Emergency Rod Insertion Panel must send the hardwired discrete signal to the individual local control rod. For the dual-redundant logic portion, if one channel is manually bypassed or if an operational channel detects the other channel has failed, there is logic in the operational channel that accomplishes bypass of the other channel so that the selected run-in function can still be accomplished as long as the hardwired discrete signal from the Emergency Rod Insertion Panel is operable. For this analysis it is assumed that, if a failed redundant channel situation exists, it has been manually bypassed by the operator to allow continued plant operation.

In parallel with the SCRRI, the RC&IS generates a Select Rod Insert (SRI) signal that is processed through an independent path in the Diverse Protection System (DPS) and then through the Scram Timing Panels to pre set selection switches for the rod pairs that are scrammed by using the scram solenoids.

Failure of the power supply to the individual rod controller equipment can also cause this failure event, but it is not considered in this analysis because it is considered to be a more significant initiating event, which is addressed separately in the design.

The frequency of the failure of feedwater heating followed by the failure of selected control rods to insert is determined by multiplying the failure frequency of feedwater heating with the probability of failure on demand of selected control rods to insert. These items are discussed below.

Failure of the Feedwater Heating: The frequency of feedwater heater failure in BWR plants is 0.02 per year based on operating experience, as reported in Table 9 of Reference 15A-5. The trend for initiating event frequencies has shown a steady decline since the 1980s when the Reference 15A-5 data was collected. Generally, the initiating event frequencies have decreased by a factor of 4. This is sufficient to indicate that the future frequency for this initiating event is likely no higher than 0.02/yr (95% confidence upper bound) and is likely a factor of 4 lower, i.e., 5E-3/yr. Recognizing this trend in initiating event frequencies, reasonable engineering judgment based on these data trends supports the use of 0.02/yr as a conservative characterization of the Loss of Feedwater Heating initiating event frequency.

The failure probability of selected control rods to insert results from the quantification of the fault tree is shown on Figure 15A-4. The fault tree includes three sub-trees in an OR-gate:

- Failures common to SRI and SCRRI
- Mechanical failures of SRI and SCRRI
- Independent system electrical failures

Failures Common to SRI and SCRRI (SRI-SCRRI-C)

This sub-tree includes the following basic events.

Failure of Redundant Temperature Sensors (T-SIGNAL-F)

The failure of the redundant temperature sensors to detect the loss of feedwater heater(s) is estimated to be small. The failure rate of a temperature transmitter is 3.5E-7 per hour, as documented in Table A.3-1 of Reference 15A-1. Even though the temperatures are displayed in the main control room, no credit is taken in this analysis for detection by the operator of sensor failure. Assuming that these sensors are tested during refueling outages every two-years, the probability that each sensor is unavailable is failure rate times the test interval divided by two = (3.5E-7)*(17,520 / 2) = 3.07E-3. The probability that both sensors fail upon demand is the product of unavailability of individual sensors, which is equal to (3.07E-3)*(3.07E-3) = 9.42E-6 per demand. In addition, common cause failure (CCF) is estimated based on a beta factor of 0.01. Thus the unavailability due to CCF is (0.01)*(3.07E-3) = 3.07E-5 per demand. The combined unavailability = 9.42E-6 + 3.07E-5 per demand, which is equal to 4.01E-5 per demand.

Failure of FWCS Redundant Signals (FWCS-SIG-F)

The FWCS is required for continued plant operation; therefore, there is a high probability that the controller is available following a loss of feedwater heater event. The failure of the FWCS to generate the select control rod run-in signals (and provide the redundant signals to the RC&IS equipment via the Nonsafety-Related DCIS) is judged to be negligible. However, to establish an upper bound failure probability sensitivity, the FWCS redundant signals are assessed to fail with a probability of 1.0E-03 per demand based on possible common cause failures in the FWCS. The probability of 1.0E-03 is estimated based on a single element failure probability of 0.05 and a common cause factor of 0.02.

Failure of RC&IS Dual-Redundant Channel Signals (RCIS-SIG-F)

The RC&IS has dual-redundant controller equipment of which only one is required for continued plant operation. When a failure occurs in one of the controllers, the failure is announced and the other controller continues to operate thus the plant operator can bypass the failed equipment and then repair or replace the failed part. For the RC&IS to fail, the second failure has to occur during the time when the first failed controller is being repaired while in the bypass condition, generally within a shift. The failure of the both controllers in this short period is very low, especially compared to the hardwired signal, which is also required required. The probability of this event is estimated to be 0.001 per demand, based on the discussion for hardwired signals below.

Failure of RC&IS Hardwired Signals via the Emergency Rod Insertion Control Panel and Emergency Rod Insertion Panels (IRLP-SIG-F)

The RC&IS provides hard-wired signals to individual control rod logic equipment. Even though the failure of this signal not to actuate when required is not announced, failure of one hard-wired signal only impacts the run-in function for one control rod and redundant relays are used for actuation of each hardwired output signal to the individual control rod logic equipment. The failure probability of the hard-wired signal is conservatively estimated to be better than, (i.e., lower than), 0.001 per demand. (For reference, failure rate of a single relay is 1.0E-4 for demand).

Failure of individual control rod logic equipment to insert selected rods has an insignificant contribution, and therefore, is not included in the fault tree model. The capability for movement of all control rods by the individual control rod logic equipment is tested during the monthly double-notch movement surveillance testing. Therefore, the only credible failure mode that prevents multiple control rods from being inserted upon command is considered to be loss of electrical power. However, the loss of power is a separate and more significant initiating event not considered in this analysis, and addressed separately in the design..

Independent System Electrical Failures (ELEC-FAIL)

Using values from current generation BWRs for logic and electrical support failures, results in a conservative estimate of system failure of 4E-3 per system. This conservative failure probability is used in estimating SRI and SCRRI independent failure probability modeled by the following basic events:

- Electrical or Logic Failures in SRI (SRI-LOGIC)
- Electrical or Logic Failures in SCRRI (SCRRI-LOGIC)

Mechanical Failures of SRI and SCRRI (MECH-FAIL)

This sub-tree includes the following basic events.

Common Cause Mechanical Failure of SRI (SRI-CM-F)

This basic event models the common cause mechanical failure of multiple control rods from the SRI group to insert. The probability of 1.0E-2 assigned to this basic event is a conservative estimate based on operating experience at BWRs.

Common Cause Failure of SCRRI due to Rod Sticking (SCRRI-CM-F)

This basic event models the common cause mechanical failure of multiple control rods from the SCRRI group to insert. The probability of 1.0E-2 assigned to this basic event is a conservative estimate based on operating experience at BWRs.

Common Cause Failure of SCRRI due to FMCRD Motors (FMCRD-F)

The probability assigned to this basic event is a conservative estimate based on the failure rate of frequently exercised motors. Given 4 groups of SCRRI rods, with 32 rods per group, the failure probability (Pf) of 2 or more FMCRDs is calculated as follows:

$$Pf = \lambda \frac{T}{2} * CCF * \frac{32\ Combinations}{group} * 4\ groups$$

$$Pf = 1E-5/hr * \frac{720\ hrs}{2} * 0.1 * 32 * 4$$

$$Pf = 4.61E-02$$

Where:

λ: Motor failure rate (standby) = 1.0E-5/hr

T: Test interval = 720hr

CCF: Common cause failure of second FMCRD = 0.1

The failure probability of selected control rods to insert, resulting from the quantification of the fault tree shown in Figure 15A-4 is 7.54E-2.

The frequency of the event of concern is obtained by multiplying the frequency of loss of FW heater (0.02/year), by the probability of failure of the total conditional mitigation failure probability(7.54E-2 per demand).

The event frequency = (0.02)*(7.54E-2) =1.51E-3 per year. This translated to one failure in more than 600 years.

15A.3.6.3 Result

The frequency of the failure of feedwater heating followed by the failure of the selected control rods to insert is less than once in more than 600 years and therefore, this event frequency meets the criterion of being less than once in 100 years.

15A.3.7 Inadvertent Shutdown Cooling Function Operation

15A.3.7.1 Introduction

The ESBWR is equipped with the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system, which is designed to perform Shutdown Cooling in one of its operating modes. The operator initiates shutdown-cooling mode of operation after the plant is shutdown, either normally, or after a reactor scram. It should not be possible for the operator to initiate shutdown-cooling mode of operation when the reactor is at power. However, combination of undetected failures and operator errors could lead to inadvertent shutdown cooling operation. The frequency of inadvertent shutdown cooling operation is estimated in this subsection. The RWCU/SDC system is described in Subsection 7.4.3.

15A.3.7.2 Analysis

The RWCU/SDC system design includes an interlock feature that prevents the operator from inadvertently engaging the system in the SDC mode of operation while the reactor is at power. Based on Subsection 7.4.3, this interlock feature is designed to be single-failure proof. The operator is not likely to engage the RWCU/SDC system in the SDC mode when the plant is in operation. However, if the interlock does not work for some reason, and the operator commits this error, then there is a potential for the RWCU system to be placed in the inadvertent SDC mode.

The postulated failure modes are identified in the fault tree of Figure 15A-3a as follows (Top Gate SDC-E):

- Inadvertent SDC Function Initiation During Power Operations
- SDC Initiation During Interlock Testing (at-power)
- Valves Spuriously Open
- Automatic Actuation of SDC at-power

Inadvertent SDC Function Initiation During Power Operations (SDC-E-F)

This failure mode requires that the crew incorrectly manipulate the SDC controls while at-power and coincident with this that the interlock is failed. Gate: SDC-E-F describes this logic. The bases for the inputs to the logic diagram are as follows:

Crew Error of Commission (SDOP-EOC-SDC--H--)

Operating experience indicates that inadvertent operation of SDC while the reactor is at-power is unlikely. Assuming there are no such events that have occurred (assume 1 incipient failure) and there are 23 BWRs * 20 years of operation, then the frequency of inadvertent SDC operation is less than 1/460 Rx Yr or 2.17E-03/RxYr.

F = 2.17E-03/RxYr

Alternatively, with the use of the THERP analysis, in Reference 15A-7, of the RWCU/SDC system and the assumption that the SDC controls are not uniquely designated or segregated from the RWCU controls, then the following errors could occur during a 2 year refuel period:

RWCU control manipulation once per week

Incorrect manipulation of the SDC controls 3E-3 (Table 20-12 Item (2) of Reference 15A-7)

Recovery from the inadvertent operation of the SDC controls 0.05 (Table 20-22 Item (3) of Reference 15A-7)

$$F = 52\frac{demands}{RxYr} * 3\text{E-03} * 5\text{E-02} = 7.8\text{E-03/RxYr}$$

This can be approximated by 1E-2/RxYr as an upper bound.

Interlock Failure Probability (SDC-E-FA-C)

A simplified model of the interlock is included to estimate a single failure proof design.

The failure probability of a single failure proof system can be estimated by a fault tree analysis. It is estimated here by two common cause failures:

Common cause miscalibration of sensors feeding the logic for the SDC interface valve logic estimated based on existing BWR PRAs and use of Reference 15A-7 to be 8E-05.

Common cause failure of multiple logic circuits conservatively estimated as 1E-02/circuit and 0.05 common cause contributions.

SDC Initiation During Interlock Testing (at-power) (SDC-E-I)

The possibility of the SDC interlock being tested during power operation is considered remote. It is estimated here as 0.1 probability per year. Given this test, the interlock is assumed bypassed. Coincident with this testing the crew must incorrectly manipulate the valves for SDC. This treatment is under Gate SDC-E-I.

This modeling is described as follows:

SDIN-LOGICTST--: This is the frequency that during power operation that the RWCU/SDC interlock would be in test. This frequency is judged small because the testing would likely be restricted to shutdown operational conditions, but is represented by a frequency of 0.1/yr.

SDPH-RESPONSEH--: This action is the conditional probability that during a test of the RWCU/SDC interlock while at-power the crew would be required to take actions to manipulate RWCU controls. Because these actions would likely be restricted during any such interlock tests, this conditional probability is judged to be quite low, but is conservatively estimated at 0.1.

SDOP-SDCINIT-H--: This is the Human Error Probability (HEP) that the crew while manipulating RWCU/SDC controls performs an incorrect series of operations that causes SDC initiation. This HEP is judged to be quite low based on the expected control design and expected crew training. Nevertheless, a conservative HEP of 1E-2 is used in the analysis. The error by the crew of 1E-2 is based on Reference 15A-7.

<u>Valves Spuriously Open (SDC-E-FA-V)</u>

While this gate results in a double count of some failure modes already addressed, it is included for completeness. It may subsequently be subsumed by more explicit modeling. The spurious open MOV frequency is 5.0E-8/hr. Reference 15A-8. The CCF basic event is the frequency failure for the SDC function to inadvertently initiate given a full year of power operation (8760 hours). A conservative common cause factor of 0.1 (NRC common cause data is ~3E-2 for 2 of 2 MOVs failing to operate) is applied and provides a result of 5.00E-08/hr * 0.1 * 8760 hours/year = 4.38E-05/yr.

<u>Automatic Actuation of SDC At-Power (SDC-E-AUTO)</u>

The SDC system is designed to automatically initiate when the control rods are fully inserted. An erroneous initiation signal combined with the failure of the SDC interlocks leads to the inadvertent operation of the SDC function. Gate SDC-E-AUTO provides the assessment of this combinations of failures.

The failure probabilities used in the fault tree are upper bound estimates.

15A.3.7.3 Result

The result of the fault tree analysis is a calculated frequency of inadvertent SDC operation at-power of approximately 1.6E-04/yr, which includes interlock failure or bypass. The frequency of this failure is less than once in 6,200 years. Therefore, this event frequency meets the criterion for an infrequent event because it is less than once in 100 years.

15A.3.8 Inadvertent Opening of a Safety Relief Valve

15A.3.8.1 Introduction

ESBWR is equipped with four Isolation Condensers and eighteen Safety Relief valves (SRVs). ESBWR is designed with the capability to handle reactor overpressurization using only the Isolation Condensers.

Subsection 5.2.2 states that for overpressure protection, the Isolation Condensers have sufficient capacity to preclude actuation of the SRVs, during normal operational transients. The SRVs are therefore a backup to the Isolation condensers and are also needed for ATWS conditions. Of the

18 SRVs, ten SRVs discharge through lines routed to quenchers in the suppression pool. The remaining eight SRVs are arranged in two groups of four. Each group discharges to a horizontal header that has a rupture disc at the end. Each header has a discharge line that is routed to a quencher in the suppression pool. These SRVs discharge through the rupture discs to the drywell or through the discharge line to the suppression pool.

The SRVs provide two main protection functions:

- Overpressure relief function (all 18 SRVs are actuated by the inlet steam pressure to prevent nuclear steam overpressurization); and
- Depressurization operation (ten SRVs are actuated by the Automatic Depressurization System, (ADS), as part of the ECCS.

Eight of the SRVs are opened by steam pressure if the direct and increasing static inlet steam pressure overcomes the restraining spring and the frictional forces acting against the inlet steam pressure at the main or pilot disk and the main disk moves in the opening direction at a faster rate than corresponding disk movements at higher or lower inlet steam pressures.

The remaining ten of the SRVs are opened by either of the following two modes of operation:

- The safety (steam pressure) mode of operation is initiated by the direct and increasing static inlet steam pressure as described above for the eight SRVs.
- The ADS (power) mode of operation is initiated when an electrical signal is received at any of the solenoid valves located on the pneumatic actuator assembly. The solenoid valve(s) open, allowing pressurized pneumatic fluid to enter the lower side of the pneumatic cylinder piston, which pushes the piston and the rod upwards.

The power-actuated SRVs can be operated individually by remote manual controls from the main control room. Remote manual actuation of the SRVs from the control room is recommended to minimize the total number of these discharges with the intent of achieving extended valve seat life.

The inadvertent opening of the SRVs is termed an "Inadvertent opening of a Relief Valve" or IORV event. The IORV event frequency is estimated in this subsection.

15A.3.8.2 Analysis

There are five ways in which an SRV can open inadvertently:

(1) Incorrect setpoint or spring adjustments

(2) Vibration induced

(3) Excess nitrogen pressure

(4) Spring relaxation

(5) Spurious opening signal

(6) Operator error

Each of these modes is discussed in more detail below:

Incorrect Setting: Incorrect (low) setpoint setting or improperly locked setpoint spring, allowing the spring adjustments to back off with vibration can potentially lead to an inadvertent opening. This calibration action, as well as the maintenance action, are very important actions that are performed with a lot of care and are checked and verified before the valve is put in service. The failure of undetected operator actions leading to an incorrect setting or spring adjustments is estimate based on Reference 15A-7 (Table 20-7 Item (1) and Table 20-22 Item (4)).

Vibration induced error: This value is based on operating experience with current SRVs in operating BWRs.

Excess Nitrogen Pressure: Excess nitrogen pressure could result in inadvertent valve opening. Based on Subsection 9.3.8, no single failure in the nitrogen system can lead to an IORV event. Failure of control valves that can lead to this condition is estimated at 2.29 E-04.

Spring Relaxation: Spring relaxation of solenoid valve with normal nitrogen pressure can potentially lead to an IORV event. However, this has never occurred in operating BWR history, and hence it is judged that this postulated failure mechanism has a negligible probability of occurrence.

Spurious Actuation Signal: Spurious actuation can occur from a failure in the control logic of the SRVs. There are 10 SRVs actuated by ADS. The other 8 SRVs are opened only by steam pressure overcoming a restraining spring; therefore, not subject to spurious signal actuation. The ADS logic was analyzed for spurious actuation in Subsection 15A.3.9 for the DPV inadvertent opening, resulting in a frequency of 5.75E-04 per year. This frequency is increased to 6.0E-04 to estimate the SRVs, to account for the fact that there are 10 SRVs, and only 8 DPVs.

Operator Error: The power-actuated SRVs can be operated individually by remote manual controls from the main control room. The operator is expected to use this feature only after the SRVs open initially, with the intent of minimizing total number of discharges. He should not be opening the SRVs inadvertently and he cannot do it accidentally because a deliberate action is required to open the SRVs. Since the primary means of controlling reactor overpressure in the ESBWR is the Isolation Condenser, the operator does not have a reason to actuate SRVs to relieve reactor pressure. The probability of an IORV resulting from an operator action in judged to be negligible.

In summary, the frequency of an IORV, based on the above discussion is as follows:

Incorrect setpoint or spring adjustments:	1.8 E-03per year
Vibration Induced:	1.8 E-04 per year
Excess nitrogen pressure:	2.29E-04 per year
Spring relaxation:	0.0 per year
Spurious opening signal:	6.0E-04 per year
Operator error:	0.0 per year
Total:	2.81E-03 per year

The resulting IORV frequency is 2.81 E-03 per year, or one event in over 300 years.

15A.3.8.3 Result

The ESBWR IORV frequency is less than once in 300 years of operation. Thus the event frequency meets the criterion of being less than once in 100 years.

15A.3.9 Inadvertent Opening of a Depressurization Valve

15A.3.9.1 Introduction

The Depressurization Valves (DPVs) are part of the Automatic Depressurization System (ADS). ADS consists of 10 SRVs and 8 DPVs and their associated instrumentation and controls.

The DPVs are described in Subsection 5.4.13. In summary, the DPVs are of a non-leak/non-simmer/non-maintenance design. They are straight-through, squib-actuated, non-reclosing valves with a metal diaphragm seal. The DPV is closed with a cap covering the inlet chamber. The cap shears off when pushed by a valve plunger that is actuated by the explosive initiator-booster.

Four initiators (igniter charges or squibs), singly or jointly, ignite a buster assembly explosive charge, which drives the shearing plunger. Each initiator is activated by an independent firing circuit. The firing of one initiator is adequate to ignite the buster, and open the valve.

The firing circuits of three DPV initiators are actuated by the ADS logic, which is part of the Engineered Safety Features (ESF) systems. In addition, the Diverse Protection System (DPS) can independently actuate the fourth DPV firing circuits. The DPS is implemented as a Nonsafety-Related system. The ESF and DPS logics are presented in Chapter 7, sections 7.3.1.1, and 7.8.1.2, respectively.

A simplified diagram of the DPV initiation logic is shown schematically in Figure 15A-1.

The conceptual design shown in Figure 15A-1, with two 2-out-of-4 logics and two load drivers per initiating (firing) circuit is generic for ECCS, including the SRV portion of ADS. The DPV actuation includes a third load driver (controlled by a third 2-out-of-4 logic) in the firing circuit for increased reliability against spurious actuation. For generality and conservatism, the analysis in this section, regarding the inadvertent opening of a DPV, is performed for the case of only two 2-out-of-4 logics and two load drivers per DPV firing circuit.

The safety-related ADS logic is implemented in four divisions. Each division has an instrument channel consisting of a level transmitter, trip-decision-making logic, and 10-second timer. All four divisions share the trip decision generated by each instrument channel. Each of the four divisions makes a 2-out-of-4 trip decision from each of the four divisional trip decisions. The system has single-channel bypass capability, that is, one channel at a time can be manually removed from the voting logic for maintenance purposes. In this case, the trip decision process reverts to a 2-out-of-3 logic.

Each division has two trains of 2-out-of-4 trip logic to support the requirement that single divisional failures do not inadvertently open any ADS valve. These trains are named Voting Logic Units (VLUs) for the purpose of this analysis. As shown in Figure 15A-1, each pair of VLUs actuates one pair of series-connected load drivers in one firing circuit of each of six DPVs. Before reaching the load drivers, the trip signals from the VLUs initiate timers, one per load driver. These timers are set such that predetermined groups of SRVs and DPVs open at staggered times, to minimize reactor water level swell. These timers are not shown in Figure 15A-1, but are accounted for in the analysis by including their failure rate in the Load Driver (LD) reliability component.

The information received from the four safety-related instrument channels is compared for consistency at each VLU input, and inconsistencies are annunciated. Any one load driver trip in a firing circuit of a DPV is also annunciated.

The non-safety DPS logic actuates one pair of series-connected load drivers in one firing circuit of each DPV. Each load driver is actuated by a dedicated 2-out-of-3 voting logic. The voting is performed between the trip decisions generated by the three instrument channels of a Nonsafety-Related controller. This triplicate channel controller is a complex piece of equipment, with high reliability due to its ability to share information between channels at different stages of the process. A reliability analysis performed by a vendor is available for a controller of this type. Therefore, the triplicate channel controller is represented as a single reliability component in Figure 15A-1 for the purpose of this analysis.

15A.3.9.2 Analysis

15A.3.9.2.1 Analysis of Failure Causes

Inadvertent opening of a DPV can occur due to one of the following causes:

- Local failure mechanisms at the DPV level, leading to ignition of the initiator-buster without any of the load drivers in the firing circuit having been closed
- Operator error
- Spurious actuation signals

Each one of these causes is discussed in the following subsections.

15A.3.9.2.1.1 Local failure mechanisms at the DPV level

The DPV has undergone engineering development testing using a prototype to demonstrate the proper operability, reliability, and flow capability of the design. Functional tests were performed to assure proper operability and the adequacy of the initiator-booster to operate the valve assembly.

The initiator used for the DPV actuation is also used in the automotive industry to actuate the airbag restraint system. Data obtained by GEH from the manufacturer of initiators shows that between 1987 and 1993 there were no reported problems of any kind in more than 15,000,000 automotive initiators that were delivered. Because the initiators were delivered over a period of 6 years, and probably more in the later part of this period, it is assumed that the average operating time for one initiator is 2 years. Based on this data, i.e., 15,000,000 initiators operating with no failure over a time period of 2 years, the estimated failure frequency for one initiator is 1/(15,000,000*2) = 3.3E-8/year. Based on this estimate, and the fact that the propellant used in the buster is more stable than the initiator, it is assessed that the contribution from local failure mechanisms at the DPV level is insignificant.

15A.3.9.2.1.2 Operator Error

The DPV control system is designed to minimize the possibility of accidental manual actuation.

Each firing circuit of a DPV includes a key-lock switch, which has to be open when testing the load drivers in that circuit. Although the load drivers are tested sequentially, and one load driver actuation alone cannot open the DPV, the key-lock switch offers additional protection against accidental firing of an initiator-buster.

Manual actuation of the DPVs can be performed from video display units (VDU) in the main control room. Safety-related and nonsafety-related VDUs can provide a display format that allows the operator to manually open each DPV independently. Each display utilizes an "arm/fire" configuration that requires at least two deliberate operator actions. Operator use of the "arm" portion of the display causes a plant alarm. Also, ADS can be manually initiated as a system to open all SRVs and DPVs, instead of each valve individually. To perform this action, each safety-related VDU can provide a display with an "arm/fire" switch (one per division). If the operator uses any two of the four switches, the ADS sequence seals in, and starts the ADS valve sequencing. This requires at least four deliberate operator actions. For all of the manual initiations, operator use of the "arm" portion of the display causes a plant alarm.

Based on the design described above, it is considered that the probability of inadvertent opening of a DPV due to operator error is insignificant compared to the probability of a spurious actuation signal.

15A.3.9.2.1.3 Spurious Actuation Signals

This failure mechanism includes spurious initiation signals, and inadvertent closure of load drivers. This failure cause is considered dominant, and analyzed in the following subsections of this report.

15A.3.9.2.2 Analysis of Spurious Actuation Signal Frequency

A fault tree was developed based on the schematic diagram shown in Figure 15A-1. Figure 15A-2 shows the fault tree model for the inadvertent opening of one or more DPVs. The calculation of the frequency of inadvertent opening of one or more DPVs is performed by first calculating the frequencies for the different categories of failures shown on the left-hand side of Table 15A-1, based on the corresponding scenarios resulting from the failure combinations shown on the right-hand side of Table 15A-1. The following presents frequency calculations for the different combinations of failures. The input failure data used in these calculations are presented in Table 15A-2.

15A.3.9.2.2.1 Frequency contribution of combinations of Load Drivers (LD) failures, and combinations of LD and Voter Logic Unit (VLU) failures

Table 15A-1 shows 3 LD combinations and 6 LD/VLU combinations for each DPV. Each combination leads to 2 scenarios resulting in inadvertent DPV opening. There are a total of 8 DPVs. Therefore:

$$F_1 = 8*[6*F(LD)*P(LD) + 6*F(LD)*P(VLU) + 6*F(VLU)*P(LD)] =$$

$$= 48*[F(LD)*P(LD) + F(LD)*P(VLU) + F(VLU)*P(LD)]$$

where:

F(LD) and F(VLU) are the yearly failure frequencies for the LD and VLU shown in Table 15A-2.

P(LD) and P(VLU) are the unavailabilities of the LD and VLU shown in Table 15A-2.

Therefore,

F_1 = 48*(8.76E-3*1.0E-5 + 8.76E-3*5.0E-5 + 4.38E-2*1.0E-5) = 4. 63E-05/year

15A.3.9.2.2.2 Frequency contribution of VLU failure combinations

F_2 = 8*F(VLU)*P(VLU) = 1.75E-05/year

15A.3.9.2.2.3 Frequency contribution of Instrument Channel (IC) failure combinations, with failure to detect first IC failure

F_3 = 12*F(IC)*P(IC_F)*P(FD) = 2.32E-6/year

where:

F(IC): Yearly failure frequency of the IC

P(IC_F): Unavailability of an IC when its failure is detected only during the channel functional test

P(FD): Probability of failing to automatically detect an IC failure

15A.3.9.2.2.4 Frequency contribution of IC failure combinations, with first IC failure detected and bypassed

F_4 = 12*F(IC)*P(IC)^2 = 1.05E-8/year

15A.3.9.2.2.5 Frequency contribution of Triplicate Channel Controller(IC_T)

F_5 = F(IC_T) = 2.65E-04/year

15A.3.9.2.2.6 Frequency contribution of LD and Voter Logic (LDV) failure combinations

F_6 = 16*F(LDV)*P(LDV) = 5.05E-5/year

15A.3.9.2.2.7 Frequency contribution of Common Cause Failures (CCFs)

To account for failure modes that are not well understood (e.g., CCF due to software), CCFs of groups of redundant and identical components were included in the model. Three groups of components subject to CCF were identified for the ESF part of the design, and one for the non-safety part of the design. A beta-factor of 1.0E-3 was assumed for the calculation of these CCF probabilities, applied to the failure rate of each type of component listed in Table 15A-2. The following are the four CCFs used in the analysis.

CCF of 2 or more LDs (LDccf):	8.76E-06/year
CCF of 2 or more VLUs (VLUccf):	4.38E-05/year
CCF of ICs (ICccf):	8.76E-05/year
CCF of 2 or more LDVs (LDVccf):	5.26E-05/year

The frequency contribution of the CCF of ESF and non-safety components is as follows:

F_7 = F(LDccf) + F(VLUccf) + F(ICccf) = 1.40E-4/year

F_8 = F(LDVccf) = 5.26E-5/year

15A.3.9.3 Results

The total frequency of inadvertent opening of one or more DPVs due to Instrumentation and Control failures is:

$F = F_1 + F_2 + F_3 + F_4 + F_5 + F_6 + F_7 + F_8$ = 5.75E-04/year

The frequency of 5.75E-04 per year translates to one DPV inadvertent opening in more than 1,700 years. Thus the event frequency meets the criterion of being less than once in 100 years. .

15A.3.10 Stuck Open Safety Relief Valve

15A.3.10.1 Introduction

ESBWR is equipped with four Isolation Condensers and eighteen Safety Relief valves (SRVs). ESBWR is designed with the capability to handle reactor overpressurization using only the Isolation Condensers.

Subsection 5.2.2 states that for overpressure protection, the Isolation Condensers have sufficient capacity to preclude actuation of the SRVs, during normal operational transients. The SRVs are therefore a backup to the Isolation Condensers and are also needed for ATWS conditions. Of the 18 SRVs, ten SRVs discharge through lines routed to quenchers in the suppression pool. The remaining eight SRVs are arranged in two groups of four. Each group discharges to a horizontal header that has a rupture disc at the end. Each header has a discharge line that is routed to a quencher in the suppression pool. These SRVs discharge through the rupture discs to the drywell or through the discharge line to the suppression pool.

The SRVs provide two main protection functions:

- Overpressure relief function (all 18 SRVs are actuated by the inlet steam pressure to prevent nuclear steam overpressurization)
- Depressurization operation (ten SRVs are actuated by the Automatic Depressurization System, (ADS), as part of the ECCS.

Even though the SRVs are not required or expected to open during a transient, under some rare conditions when all the Isolation Condensers not available, one or two SRVs may open. When the SRVs are opened, there is a chance that they will get stuck in the open position. The event in which the SRV sticks open is identified as a Stuck Open Relief Valve (SORV) event, and its frequency is evaluated in this subsection.

There is a potential for SRVs to stick open if the SRVs are tested at power. However, as stated in Subsection 5.2.2.4, it is not practical to test the SRV setpoints while the reactor is at power. Therefore, the potential for an SORV to occur following a SRV test at power is not considered.

15A.3.10.2 Analysis

For an SORV event to occur, first, there should be a transient event in which there is a potential for reactor over pressurization, and second, one of the Isolation Condensers which is designed to actuate on demand does not open, and third, a number of SRVs open to relieve the pressure, and then finally, one of the SRVs fails to reclose after opening. It is assumed that four SRVs open when the Isolation Condenser is unavailable following a pressurization transient.

From Table 2.3-3 of Reference 15A-2, the following events are identified as overpressurization events:

- Transient with PCS unavailable: 1.97E-1 events /year
- Loss of Feedwater: 1.17E-1 events/year
- Loss of Preferred Power: 3.59E-2 events/year
- Total 3.50E-1 events/year

The probability that Isolation Condenser System is not available on demand is conservatively estimated to be 0.1. The actual value is expected to be significantly lower.

In Reference 15A-6, there are five SORV events occurred in BWR plants (it is noted that only one of these instances were with direct-acting SRVs, so therefore this is a conservative approach to getting the frequency of SORV). The number of overpressurization events in BWRs in that database is estimated by adding the frequency of total loss of heat sink (122 events) with loss of offsite power (33 events), a total of 155 events. The assumption of four SRVs opened during each overpressurization transient (note: lower number gives a conservative value), results in a total of 620 SRV actuations which resulted in five SORV events. Therefore, the conditional probability of any SRV sticking open after it opens initially = 5 divided by 620, which is equal to 0.0016 per valve opening.

The ESBWR overpressurization frequency in which SRVs are likely to open is obtained by multiplying the frequency of over pressurization transient by probability that the Isolation Condensers are unavailable, which is 3.50E-1 times 0.1 = 3.50E-2 events per year.

The expected number of SRV actuations = 3.50E-2 times four = 1.40E-1 SRV actuations per year.

The frequency of SORV = 1.40E-1 times 0.0016 = 2.24E-4 per year.

The frequency of SORV can also be expressed as once in over 4,400 reactor years.

15A.3.10.3 Result

The ESBWR SORV frequency is less than once in 4,400 years of operation. Thus the event frequency meets the criterion of being less than once in 100 years.

15A.3.11 Control Rod Withdrawal Error During Refueling

15A.3.11.1 Introduction

The control rod withdrawal error event during refueling involves inadvertent criticality due to the complete withdrawal or removal of the most reactive rod (or pair of control rods associated with the same Control Rod Drive system hydraulic control unit) during refueling. Two channels of instrumentation are provided to sense the position of each of the control rods. In addition, redundant signals for the position status of the refueling machine and the loading of the refueling machine main hoist are provided to both channels of the RC&IS logic. With the reactor mode switch in the refueling position, the indicated conditions are combined in redundant RC&IS logic circuits to determine if all restrictions on refueling equipment operations and control rod withdrawal are satisfied. The reactor mode switch status is sensed by four channels (i.e. divisions) of safety logic with each channel providing separate, isolated status inputs into the two channels of non-safety Rod Control and Information System (RC&IS). A rod withdrawal block based upon this redundancy in either RC&IS channel provides a control rod withdrawal block to all control rods.

While in the refueling mode, detection of an operable control rod not being at its full-in position results in activation of interlock signals being provided from the RC&IS to the refueling equipment that prevent operating the equipment over the reactor core when loaded with a fuel assembly. Conversely, when the refueling equipment is located over the core and loaded with fuel, the refueling equipment provides redundant interlock signals to the RC&IS that generates a control rod withdrawal block signal in the RC&IS to prevent withdrawing a control rod.

This event is initiated by one or more operator errors followed by failure of the refueling equipment interlocks.

15A.3.11.2 Analysis

The following is an analysis of the operational conditions during refueling that could lead to a potential control rod withdrawal error.

15A.3.11.2.1 Fuel Insertion with Control Rod Withdrawn

All operational control rods are fully inserted when fuel is being loaded into the core to minimize the possibility of loading fuel into a cell containing no control rod. Refueling interlocks associated with both rod withdrawal and movement of the refueling platform back up this requirement. 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 in the hoist. Likewise, if the refueling platform is over the core and fuel is on the hoist, control rod withdrawal is blocked by associated RC&IS logic. In addition, the control rod scram function provides backup mitigation action should a criticality occur during refueling. Since the scram function and refueling interlocks may be suspended, alternate backup protection required by Technical Specifications is obtained by assuring that an array of control rods, centered on the withdrawn control rod, are inserted and are incapable of being withdrawn (by insertion of a control rod block). Since this event requires operator error in loading the fuel plus the failure of the multiple refueling interlocks and redundant RC&IS logic or not following the procedures required by Technical Specifications, this event frequency is assessed to be significantly less than once in 1,000 years.

The capability to place individual control rods in the inoperable bypass status in the RC&IS logic can be used to allow multiple (e.g., more than one control rod or control rod pair) control rod withdrawals, control rod blade replacement, associated control rod drive (CRD) removal or repair, or any combination of these, provided all fuel has been removed from the cell if the control rod blade does not remain fully inserted. With no fuel assemblies in the core cell, the associated control rod has no reactivity control function and is not required to remain fully inserted. Prior to reloading fuel into the cell, however, the associated control rod must be inserted to ensure that an inadvertent criticality does not occur. There is a special case when loading fuel into the core with multiple control rods withdrawn under administrative controls. Special spiral reload sequences are used to ensure adequate detection of the neutron flux level by the Startup Range Neutron Monitor equipment, as such reload sequences are being performed (e.g. for providing monitoring capability for inadvertent criticality). Spiral reloading encompasses reloading a cell (four fuel locations immediately adjacent to a control rod) on the edge of a continuous fueled region (the cell can be loaded in any sequence). The occurrence of an inadvertent criticality event under this special case is assessed to be less than 0.00000001 per year or one event in 10,000,000 years based on GEH SIL 372 (Reference 15A-3).

15A.3.11.2.2 Second Control Rod Removal or Withdrawal

When the platform is not over the core (or fuel is not on the hoist), and the mode switch is in the REFUEL position, only one operable control rod can be withdrawn when the RC&IS SINGLE/GANG switch is in the SINGLE position. When the RC&IS switch is in the GANG position, only one operable control rod pair associated with the same HCU may be withdrawn. Any attempt to withdraw an additional rod results in a rod block by the redundant RC&IS logic. Because the core is designed to meet shutdown requirements with one such control rod pair (associated with the same HCU) of the maximum reactivity worth, OR one rod of maximum reactivity worth withdrawn, the core remains subcritical even with one such control rod pair (of control rod) withdrawn. Withdrawal of a second control rod or a second rod pair (with the same HCU) would require an operator error and failure of the redundant RC&IS rod withdrawal block logic and failure of the scram function. The frequency of this type of event is assessed to be significantly less than once per 1,000 years based on the multiple failures required for this event to occur.

15A.3.11.2.3 Control Rod Removal Without Fuel Removal

The installed design of the control rod incorporates a bayonet coupling system that without disassembly of the control rod drive equipment in the under-vessel area, it physically impossible to accomplish the upward removal of the control rod blade without:

- The simultaneous or prior removal of the four adjacent fuel bundles, and
- Decoupling of the control rod blade by physical rotation of the blade relative the associated coupling spud of the hollow piston tube.

Therefore, based on the required conditions for this event to occur, this event is considered not credible.

15A.3.11.3 Results

The frequency of a rod withdrawal error during refueling is evaluated to be significantly less than once in 1,000 years based the multiple failures that are required for this event to occur. This event therefore meets the criterion of less than one event in 100 years.

15A.3.12 Control Rod Withdrawal Error During Startup

15A.3.12.1 Introduction

It is postulated that, during reactor startup, a single control rod is inadvertently withdrawn continuously due to a procedural error by the operator during manual rod withdrawal, or a gang of control rods is inadvertently withdrawn due to a malfunction in the automated rod movement control system (ganged rod operation) of the Plant Automation System (PAS), when in the automatic startup mode. Rod withdrawal block signals are generated whenever selected single or ganged rod movements differ from those allowed by the reference rod pull sequence (RRPS), when the RC&IS is in either the automatic or semi-automatic rod movement mode. The RC&IS is described in Subsection 7.7.2.

The RC&IS has a dual channel rod worth minimizer (RWM) function that prevents withdrawal of any out-of-sequence rods from 100% to 50% control rod density, i.e., for Group 1 to Group 4 rods. It also has ganged withdrawal sequence restriction constraints such that, if the withdrawal sequence constraints are violated, the rod worth minimizer function of the RC&IS initiates a rod block. The RWM sequence restriction constraints are in effect from 100% control rod density up to the low power setpoint.

The Plant Automation System includes triple-redundant process controllers. It provides rod movement demand signals to the RC&IS to accomplish automatic positioning of the control rods during an automatic startup, shutdown or during automatic power range maneuvers. The Plant Automation System is described in Subsection 7.7.4.

In addition, the startup range neutron monitors (SRNMs), a subsystem of the Neutron Monitoring System (NMS), has a "period withdrawal permissive" automatic rod withdrawal interlock for each of twelve SRNM instruments, three SRNMs per NMS division. It is also possible to bypass one SRNM in each core quadrant, or all three SRNMs in one NMS division. When any of the unbypassed SRNM channels senses that the reactor period reaches the rod withdrawal block setpoint due to erroneous control rod withdrawal, control rod withdrawal is blocked and automatic control rod operation by the PAS is interrupted. As a result, continuous control rod withdrawal is stopped. If the reactivity addition by the rod withdrawal error is large enough, the SRNM scram function is also initiated (i.e. if the unbypassed SRNMs of two or more NMS divisions detect the reactor period has reached the associated scram function setpoint). The SRNM setpoints are so selected that no violation of the applicable thermal margins occurs during this event. The NMS is described in Subsection 7.2.2.

Because both the RC&IS and Plant Automation System include either a dual channel or triple-redundant processors, no single failure can cause this event to occur.

15A.3.12.2 Analysis

15A.3.12.2.1 Automatic Rod Movement during Startup

During a typical plant startup, the PAS automated rod movement control function provides command signals to the RC&IS that withdraws the rod gangs. If there were erroneous ganged rod withdrawal initiated by the PAS that result in a flux excursion with the measured SRNM period for an unbypassed SRNM shorter than 20 seconds during rod withdrawal, the SRNM function and associated redundant RC&IS logic initiates the associated rod withdrawal block function. If there is a measured flux excursion shorter than 10 seconds, as detected by the unbypassed SRNMs of two NMS divisions, a scram is initiated. Therefore, an unmitigated rod withdrawal error during the automatic startup would require a failure in the PAS automated rod movement control function followed by a failure of the SRNM rod block trip and SRNM scram initiation. Triple-redundant fault-tolerant digital controllers and redundant system controllers would have to fail to cause loss of the PAS and RC&IS control logic functions. In addition, all unbypassed channels of SRNM system and the redundant RC&IS logic would have to fail to cause loss of the rod block function. The unbypassed SRNMs of 3 of 4 NMS divisions would have to fail to cause loss of the period-based scram function. The frequency of an automatic control rod withdrawal error during startup can be calculated as:

Annual Frequency of Automatic Control Rod Withdrawal Error =

(Number of startups/year) times

(Probability of failure of redundant PAS control logic) times

(Probability of failure of both SRNM rod block trip channels)

Because of the multiple failures required for this event, it can be expected that this frequency is significantly less than 1/100 years. To demonstrate this without a detailed analysis of the systems involved, a bounding calculation was performed. The number of starts per year was conservatively assumed to be 5 starts per year. The actual number of starts based on the ESBWR design can be expected to be no more than 2 starts per year. It is assumed that the probability of failure of both the redundant PAS and redundant SRNM channels is conservatively bounded by a common cause failure that disables both systems. The failure rate for electronic processors (from Reference 15A-4, Chapter 19, Table 19D.6-7, item Division 1 Transmission Network) is 1.0E-5/hour. Assuming 24 hours per startup, the probability of failure/startup is (1.0E-05) X 24 = 2.4E-04/startup. Applying a beta-factor of 1.0E-03, the probability of a common cause failure disabling both systems is (2.4E-04 X 1.0E-03) = 2.4E-07/startup. The beta factor is also obtained from the page 19N-3 of Reference 15A-4. The final calculation of the frequency of a control rod withdrawal error during a startup using the automatic rod movement system is (5 starts/year) X (2.4E-07/start) = 1.2E-06/year = 1 event/8.3E+05 years.

15A.3.12.2.2 Manual Rod Movement during Startup

This event consists of an operator or procedural error during a single or ganged rod group withdrawal. The dual channel RWM enforces specific control rod sequences to limit the potential amount and rate of reactivity increase during control rod withdrawals. Control rod withdrawal is blocked when there is an out of sequence control rod withdrawal. The frequency of manual control rod withdrawal error during startup can be calculated as:

Annual Frequency of manual Control Rod Withdrawal Error =

(Number of startups/year) times

(Probability of operator or procedural error per startup) times

(Probability of failure of the RWM to block control rod movement)

To demonstrate the low frequency without a detailed analysis of the systems involved, a bounding calculation was performed similar to the previous section. It was conservatively assumed that there would be 5 starts per year. The probability of an operator or procedural error per startup is conservatively assumed to be 1 event in 10 startups (0.1 per startup). The final results are insensitive to these two assumptions. As in the previous section, the failure rate for electronic processors (Reference 15A-4) is 1.0E-5/hour. Assuming 24 hours per startup, the probability of failure/startup of a single channel in the RWM (1.0E-05) X 24 = 2.4E-04/startup. The probability of both channels failing is (2.4E-04)2 = 5.8E-08/startup. This failure probability assumes that the first channel is not repaired during the 24-hour period. If it can be repaired in less than 24 hours, then the probability would be even lower.

Using the same beta-factor as in the previous section, the probability of a common cause failure of both RWM channels is (2.4E-04 X 1.0E-03) = 2.4E-07/startup. The total failure probability for a startup is (5.8E-08) + (2.4E-07) = 3.0E-07. The final calculation of the frequency of a control rod withdrawal error during a startup using manual rod withdrawal is (5 starts/year) X (0.1) X (3.0E-07) = 1.5E-07/year = 1 event/6.7E+06 years.

15A.3.12.3 Results

The frequency of a rod withdrawal error due to automatic or manual startup is evaluated to be less than once in 741,000 years based the multiple failures that are required for this event to occur. Therefore, the event frequency meets the criterion of being less than one in 100 years.

15A.3.13 Control Rod Withdrawal Error During Power Operation

15A.3.13.1 Introduction

The causes of a potential rod withdrawal error (RWE) at power are either a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously, or a malfunction of the automated rod withdrawal sequence control logic during automated operation in which a gang of control rods is withdrawn continuously. In either case, the operating thermal limits rod block function blocks any further rod withdrawal when the operating thermal limit is reached. That is, the withdrawal of rods is stopped before the operating thermal limit is reached. The performance of the automated thermal limit monitor (ATLM) subsystem of the RC&IS prevents the RWE event from occurring. The core and system performance are not affected by such an operator error or control logic malfunction.

In the ESBWR, the ATLM subsystem performs the rod block monitoring function. The ATLM is a dual channel subsystem of the RC&IS. Each ATLM channel has two thermal limit monitoring functions. One function monitors the MCPR limit and protects the operating limit MCPR, and the other function monitors the MLHGR limit and protects the operating limit of the MLHGR. The rod block algorithm and setpoint of the ATLM are based on actual on-line core thermal limit information. If any one of the limits is reached, such as due to control rod withdrawal, control rod withdrawal block is initiated.

15A.3.13.2 Analysis

15A.3.13.2.1 Automatic Rod Movement during Power Operation

The analysis of the rod withdrawal error during power operation is similar to the bounding analysis for startup operation. The frequency of an automatic control rod error during power operation is calculated is follows:

Annual frequency of automatic control rod error during power operation =

(Frequency of failure of redundant PAS control logic/year) times

(Probability of failure of the dual channel ATLM subsystem)

It is assumed the failure of both redundant PAS control logic channels is dominated by common cause failure. Using the same failure rate and beta-factor as used for the startup control rod withdrawal error, the frequency of a common cause failure of the PAS control logic channels causing a gang of control rods to be withdrawn continuously is (1.0E-05 failures per hour) X (8760 hours/year) X (Beta factor 1.0E-03). This is (8.76E-02/year) X (1.0E-03) = 8.76E-05/year. As in the previous section for startup operation, the failure rate for electronic processors (from ABWR PRA) is 1.0E-5/hour. The probability of failure of both ATLM channels is calculated based on a 92 day Technical Specification test interval and no annunciation of failures or repair. This is very conservative since failures are normally annunciated and the failed channel restored in less than 12 hours. The probability of both ATLM channels being unavailable is (1.0E-05 x 24 hours x 92 days/2)2 = 1.22E-04. The probability of a common cause failure using a beta-factor of 1.0E-03 is (1E-05 x 1E-03 x 24 hours x 92 days/2) = 1.1E-05. The final calculation of the frequency of an automatic control rod withdrawal error during power operation is (8.76E-05/year x 1.33E-04) = 1.2E-09/year = 1 event/8.3E+8 years.

15A.3.13.2.2 Manual Rod Movement during Power Operation

The frequency of a manual control rod error during power operation is calculated is follows:

Annual frequency of manual control rod error during power operation =

(Frequency of operator error /year) times

(Probability of failure of the dual channel ATLM subsystem)

The frequency of operator control rod withdrawal error is dependent on the number of times an operator performs manual control rod withdrawals within a year. It is assumed that an operator makes a control rod withdrawal error 1 time every 5 years. This is considered conservative since the ESBWR design provides the operator with information on the main control panel to assist in control rod withdrawal and reduce the potential of a procedural error. Also, the results and conclusions are not very sensitive to this assumption. Using probability of failure of the dual channel ATLM subsystem from the previous section for automatic control rod withdrawal, the final calculation of the frequency of a manual control rod withdrawal error during power operation is (1 event/5 years) X (1.22E-04) = 2.44E-05/year or 1 event/40,984 years. This calculated frequency should be recognized as a very conservative bounding value. A more realistic analysis, taking into consideration the ESBWR designed test features that provides annunciation of failures and allows the restoration of a failed logic channel in a reasonable short period, could reduce the calculated frequency by one or more orders of magnitude.

15A.3.13.3 Results

The frequency of a rod withdrawal error during power operation is calculated to be one in 40,984 years based the multiple failures that are required for this event to occur. This event therefore meets the criterion of less than one event in 100 years.

15A.3.14 Fuel Assembly Loading Error, Mislocated Bundle

15A.3.14.1 Introduction

The loading of a fuel bundle in an improper location with subsequent operation of the core requires three separate and independent errors:

- A bundle must be placed into a wrong location in the core.
- The bundle that was supposed to be loaded where the mislocation occurred is also put in an incorrect location or discharged.
- The misplaced bundles are overlooked during the core verification process performed following core loading.

Proper location of the fuel assembly in the reactor core is readily verified by visual observation and assured by verification procedures during core loading. GEH provides recommended fuel assembly loading instructions for the initial core as part of the Startup Test Instructions (STIs). It is expected that the plant owners use similar procedures during subsequent refueling operations. Verification procedures include inventory checks, current bundle location logs, serial number verifications and visual or photographic inspection of the loaded core. The verification procedures are designed to minimize the possibility of the occurrence of the mislocated bundle accident.

15A.3.14.2 Analysis

The likelihood of operating the core with a mislocated bundle is low because multiple errors are required. The likelihood of a mislocation resulting in a reduced thermal margin is also low. In an initial core most mislocations do not cause adverse effects on thermal margin. For reload cores, at least two bundles have to be mislocated and fuel locations are verified. Verification procedures include inventory checks, current bundle location logs, serial number verifications and visual or photographic inspection of the loaded core.

Current operating plants have provided the basis for changing this event from an AOO event to an infrequent event. With improved design features, the ESBWR is expected to be as good as or better than current operating experience in preventing this event. Event calculations based on actual plant operating experience supports the infrequent event classification. A 2004-2005 utility survey indicates there has been no confirmed mislocated bundle events based on 25 years of operating experience for 29 plants (a total of 725 years). The estimated failure frequency based on 0 failures and 725 years at the 50% confidence interval is 0.00096 per year or 1 event in 1,046 years.

15A.3.14.3 Results

The frequency of a mislocated fuel assembly during power operation is estimated to be 1 event in 1,046 years. Thus the event frequency meets the criterion of being less than once in 100 years.

15A.3.15 Fuel Assembly Loading Error, Misoriented Bundle

15A.3.15.1 Introduction

Proper orientation of fuel assemblies in the reactor core is readily verified by visual observation and assured by verification procedures during core loading. Five separate visual indications of proper fuel assembly orientation exist:

- The channel fastener assemblies, including the spring and guard used to maintain clearances between channels, are located at one corner of each fuel assembly adjacent to the center of the control rod.
- The identification boss on the fuel assembly handle points toward the adjacent control rod.
- The channel spacing buttons are adjacent to the control rod passage area.
- The assembly identification numbers that are located on the fuel assembly handles are all readable from the direction of the center of the cell.
- There is cell–to–cell replication.

15A.3.15.2 Analysis

Current operating plants have provided the basis for changing this event from an AOO event to an infrequent event. With improved design features, the ESBWR is expected to be as good as or better than current operating experience in preventing this event. Event calculations based on actual plant operating experience supports the infrequent event classification. A 2004-2005 utility survey indicates there has been three confirmed misoriented bundle events that went undetected based on 25 years of operating experience for 29 plants (a total of 725 years). The actual frequency based on 3 errors in 725 years is 4.1 E-03 per year or 1 event in 242 years. This estimate is considered conservative since improved core verification procedures have been adopted by utilities. Zero errors during the most recent period from June 1995 to January 2005, representing 290 years of operating, confirms the effectiveness of the improved core verification procedures. Based on 0 errors and 290 years of operation, the 50% confidence estimate is 0.0024 failures per year or 1 event in 418 years.

15A.3.15.3 Results

The frequency of a misoriented fuel assembly during power operation is 1 event in 418 years based on improved core verification procedures. Thus the event frequency meets the criterion of being less than once in 100 years.

15A.3.16 Liquid-Containing Tank Failure

15A.3.16.1 Introduction

A description of this event is provided in Subsection 15.3.16.

15A.3.16.2 Analysis

To date there has not been a direct release of the contents of a waste gas decay tank or other direct release to the environment. The total U.S. reactor experience (1969–1997) is 1,392 PWR calendar years and 710 BWR calendar years as reported in NUREG/CR-5750 (Reference 15A-6). Given that there have been no events of this type in 2,102 calendar years reported in NUREG/CR-5750, the frequency of occurrence based on 0 failures and 2,102 calendar years is 1 event in 3,033 years at the 50% confidence level.

15A.3.16.3 Results

The probability of occurrence of an uncontrolled direct release of liquid waste to the environment is calculated to be 1 event in 3,033 years. Thus the event frequency meets the criterion of being less than once in 100 years.

15A.4 SUMMARY

The frequency of occurrence for each of the events classified as infrequent events in Table 15.0-7 has been analyzed. Each event has been shown to have frequency of occurrence less than once in 100 years and therefore is classified as an infrequent event. A summary of the event frequency estimates is shown in Table 15A-3.

15A.4.1 COL Information

None

15A.5 REFERENCES

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15A-3 GE Nuclear Energy, "Recommended Technical Specifications for Fuel Loading", SIL 372, June 1982.

15A-4 GE Nuclear Energy, "23A6100, ABWR Standard Safety Analysis Report"

15A-5 USNRC, "Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments", NUREG/CR-3862, May 1985.

15A-6 USNRC, "Rates of Initiating Events at US Nuclear Power Plants: 1987-1995", NUREG/CR-5750, February 1999

15A-7 USNRC, "Handbook of Human Reliability Analysis", NUREG/CR-1278, August 1983.

15A-8 Eide, S.A. et al., Generic Component Failure Data Base for Light Water and Liquid Sodium Reactor PRAs, EGG-SSRE-8875, February 1990.

15A-9 Nonelectronic Parts Reliability Data, NPRD-95, Reliability Analysis Center, Rome Laboratory, Griffiss AFB, NY 13441-5700.

15A-10 D.M. Ericson, Jr., Editor, et. al., Analysis of Core Damage Frequency Internal Events Methodology, NUREG/CR-4550, Vol. 1, Rev. 1, January 1990.

15A-11 Reactor Safety Study; An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants Main Report, WASH-1400 (NUREG-75/014), October 1975.

15A-12 The Institute of Electrical and Electronics Engineers, Inc. (IEEE), “IEEE Guide to the Collection and Presentation of Electrical, Electronic, Sensing Component, and Mechanical Equipment Reliability Data for Nuclear Power Generating Stations”, IEEE Std. 500-1984.

Table 15A-1

I&C Failures Leading to Inadvertent Opening of DPVs

Category			Combinations of Failures	
ESF Portion of DPV Control	Common Cause Failures		IC_CCF	
			LD_CCF	
			VLU_CCF	
	Combinations of Load Driver Failures and Combinations of Load Driver and VLU Failures	DPV 1	LD11	LD12
			LD11	VLUA2
			LD12	VLUA1
			LD13	LD14
			LD13	VLUB2
			LD14	VLUB1
			LD15	LD16
			LD15	VLUC2
			LD16	VLUC1
		DPV 2	LD21	LD22
			LD21	VLUB2
			LD22	VLUB1
			LD23	LD24
			LD23	VLUC2
			LD24	VLUC1
			LD25	LD26
			LD25	VLUD2
			LD26	VLUD1
		DPV 3	LD31	LD32
			LD31	VLUC2
			LD32	VLUC1
			LD33	LD34
			LD33	VLUD2
			LD34	VLUD1
			LD35	LD36
			LD35	VLUA2
			LD36	VLUA1
		DPV 4	LD41	LD42
			LD41	VLUD2
			LD42	VLUD1
			LD43	LD44
			LD43	VLUA2
			LD44	VLUA1
			LD45	LD46
			LD45	VLUB2
			LD46	VLUB1

Table 15A-1

I&C Failures Leading to Inadvertent Opening of DPVs (continued)

Category			Combinations of Failures	
ESF Portion of DPV Control (continued)	Combinations of Load Driver Failures and Combinations of Load Driver and VLU Failures (continued)	DPV 5	LD51	LD52
			LD51	VLUA2
			LD52	VLUA1
			LD53	LD54
			LD53	VLUB2
			LD54	VLUB1
			LD55	LD56
			LD55	VLUC2
			LD56	VLUC1
		DPV 6	LD61	LD62
			LD61	VLUB2
			LD62	VLUB1
			LD63	LD64
			LD63	VLUC2
			LD64	VLUC1
			LD65	LD66
			LD65	VLUD2
			LD66	VLUD1
		DPV 7	LD71	LD72
			LD71	VLUC2
			LD72	VLUC1
			LD73	LD74
			LD73	VLUD2
			LD74	VLUD1
			LD75	LD76
			LD75	VLUA2
			LD76	VLUA1
		DPV 8	LD81	LD82
			LD81	VLUD2
			LD82	VLUD1
			LD83	LD84
			LD83	VLUA2
			LD84	VLUA1
			LD85	LD86
			LD85	VLUB2
			LD86	VLUB1

Table 15A-1

I&C Failures Leading to Inadvertent Opening of DPVs (continued)

Category			Combinations of Failures		
ESF Portion of DPV Control (continued)	Combinations of VLU Failures		VLUA1	VLUA2	
			VLUB1	VLUB2	
			VLUC1	VLUC2	
			VLUD1	VLUD2	
	Combinations of Instrument Channel Failures	Failure of First Channel is not Detected	ICA_F	ICB_F	FD
			ICA_F	ICC_F	FD
			ICA_F	ICD_F	FD
			ICB_F	ICC_F	FD
			ICB_F	ICD_F	FD
			ICC_F	ICD_F	FD
		First Failed Channel is Bypassed	ICA	ICB	ICC
			ICA	ICB	ICD
			ICA	ICC	ICD
			ICB	ICC	ICD
Non-Safety (DPS) Portion of Control	Triplicate Controller		IC_T		
	Load-Driver/Voter CCF		LDV_CCF		
	Combinations of Load-Driver/Voter Failures - One per DPV -		LDV17	LDV18	
			LDV27	LDV28	
			LDV37	LDV38	
			LDV47	LDV48	
			LDV57	LDV58	
			LDV67	LDV68	
			LDV77	LDV78	
			LDV87	LDV88	

NOTE: Naming of failure events is based on schematic diagram in Figure 15A-1. The combinations of failures result from the fault tree in Figure 15A-2.

Table 15A-2

Failure Data

Component / Event		Failure Rate [/h]	Failure Frequency [/y]	MTTR [h]	Test Interval [h]	Unavailability
Description	Acronym					
Load Driver	LD	1.00E-06	8.76E-03	10		1.00E-05
Voting Logic Unit	VLU	5.00E-06	4.38E-02	10		5.00E-05
Instrument Channel	IC	1.00E-05	8.76E-02	10		1.00E-04
IC with Failure not Detected	IC_F	1.00E-05	8.76E-02		4416	2.21E-02
Load Driver CCF	LDccf	1.00E-09	8.76E-06			
Voting Logic Unit CCF	VLUccf	5.00E-09	4.38E-05			
Instrument Channel CCF	ICccf	1.00E-08	8.76E-05			
Failure to Detect IC Failure	FD					P = 1.0E-04
LD and Voter Group	LDV	6.00E-06	5.26E-02	10		6.00E-05
LDV CCF	LDVccf	6.00E-09	5.26E-05			
Triplicate Channel Controller	IC_T	3.03E-08	2.65E-04			

References and Notes for Table 15A-2:

Load Driver (LD). This component includes the actual load driver and associated timer. Per Reference 15A-1, Page A.A-28, the failure rate for a solid-state relay spurious operation is 2.0E-07/h, and the failure rate of a solid-state time-delay relay premature operation is 5.0E-07/h. The failure rate of the LD component, for the purpose of this calculation, was conservatively assumed 1.0E-06/h, or 8.76E-03/y. Actuation of a load driver is annunciated. Therefore, the unavailability of the LD component is calculated based on its Mean Time to Repair (MTTR). Assuming an MTTR of 10 hours, the LD unavailability is 1.0E-05.

Voting Logic Unit (VLU). The failure rate of the VLU generating a spurious signal is conservatively assumed to be 5.0E-06/h, or 4.38E-2/y. This is a conservative estimate, based on reliability requirements in Reference 15A-4. This value assumes there are multiple circuit boards included in this reliability component. The VLU unavailability, based on a 10 hour MTTR is 5.0E-5.

Instrument Channel (IC). This component includes the level transmitter, trip-decision-making logic, and associated timers. Based on its complexity, a failure rate twice the value of the VLU failure rate was assumed. The failure rate used for the IC generating a spurious signal is 1.0E-05/h, or 8.76E-02/y. The failure of an instrument channel is normally detected immediately, by comparing the inputs to the VLUs. In this case, the IC unavailability, assuming an MTTR of 10 hours is 1.0E-4. If the IC failure was not detected, it is assumed it is detected and repaired during the channel functional test, once every 184 days, or 4416 hours. In this case, the IC unavailability is 2.21E-02.

Failure to Detect an IC Failure (FD). The failure detection mechanism for a faulty trip signal generated by one of the instrument channels is fairly simple. It consists of annunciating any inconsistency between the inputs to the VLUs. There are four VLUs, each getting all four IC inputs from the four ESF divisions. The failure to detect an inconsistency should be negligible.

However, for the purpose of this conservative analysis, a failure probability of 1.0E-4 was assumed.

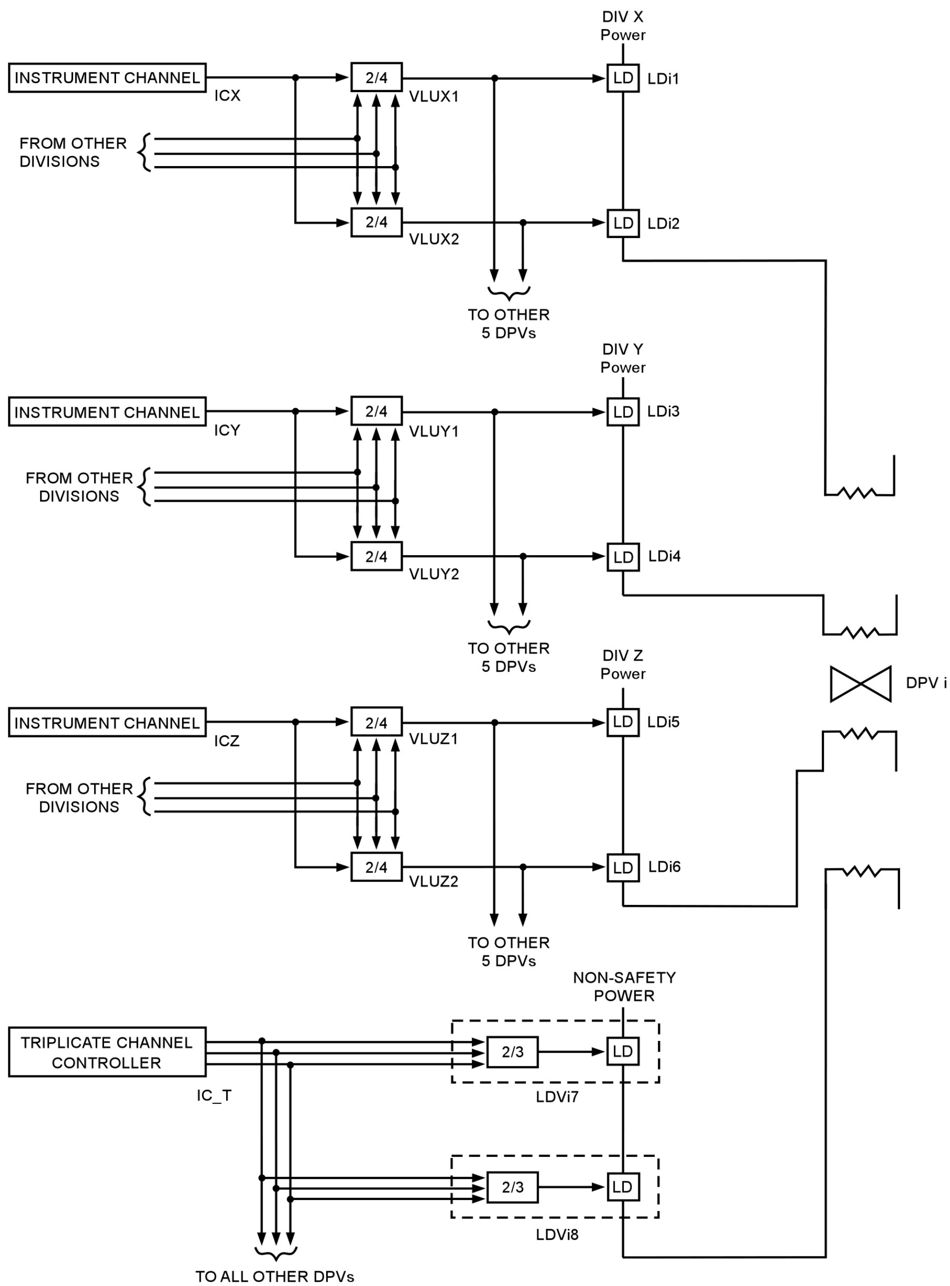
Load Driver and Voter (LDV). Compared to the LD component, this component also includes a dedicated two-out-of-three voting logic. Its failure rate is assumed to be the sum of the LD and VLU failure rates described above. Therefore, a failure rate of 6.0E-06/h, or 5.26E-2/y was assumed for this component. The unavailability of the LDV, based on a 10 hour MTTR, is 6.0E-05.

Triplicate Channel Controller (IC_T). A reliability analysis was performed by the vendor of a similar controller. The result of that analysis was assumed applicable to this controller. Therefore, the failure rate for IC_T inadvertently opening the DPVs is assumed to be 3.03E-8/h, or 2.65E-04/y.

Table 15A-3

Summary of Event Frequency Estimates

Section Number	Event	Event Frequency
15A.3.1	Pressure Regulator Failure – Opening of All Turbine Control and Bypass Valves	1 Event in 2,000 Years
15A.3.2	Pressure Regulator Failure – Closure of All Turbine Control and Bypass Valves	1 Event in 2,000 Years
15A.3.3	Turbine Trip with Total Turbine Bypass Failure	1 Event in 1,900 Years
15A.3.4	Generator Load Rejection with Total Turbine Bypass Failure	1 Event in 5,000 Years
15A.3.5	Feedwater Controller Failure – Maximum Demand	1 Event in 2,000 Years
15A.3.6	Loss of Feedwater Heating With Failure of Selected Control Rod Run-In	1 Event in 600 Years
15A.3.7	Inadvertent Shutdown Cooling Function Operation	1 Event in 6,200 Years
15A.3.8	Inadvertent Opening of a Safety Relief Valve	1 Event in 300 Years
15A.3.9	Inadvertent Opening of a Depressurization Valve	1 Event in 1,700 Years
15A.3.10	Stuck Open Safety Relief Valve	1 Event in 4,400 Years
15A.3.11	Control Rod Withdrawal Error During Refueling	1 Event in 1,000 Years
15A.3.12	Control Rod Withdrawal Error During Startup	1 Event in 741,000 Years
15A.3.13	Control Rod Withdrawal Error During Power Operation	1 Event in 40,900 Years
15A.3.14	Fuel Assembly Loading Error, Mislocated Bundle	1 Event in 1,046 Years
15A.3.15	Fuel Assembly Loading Error, Misoriented Bundle	1 Event in 418 Years
15A.3.16	Liquid-Containing Tank Failure	1 Event in 3,000 Years



NOTES: This figure shows only 2 (instead of 3) VLUs/LDs per firing circuit for consistency with the conservative analysis performed

"i" represents the DPV number

"X," "Y," and "Z" represent the divisions assigned to each DPV

i	1	2	3	4	5	6	7	8
X	A	B	C	D	A	B	C	D
Y	B	C	D	A	B	C	D	A
Z	C	D	A	B	C	D	A	B

Figure 15A-1. DPV Initiation Logic

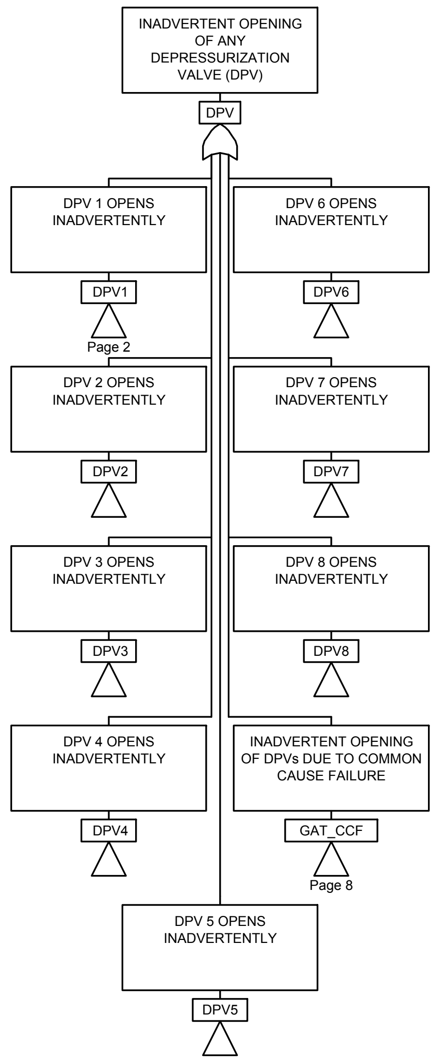


Figure 15A-2a. Fault Tree – Inadvertent Opening of a Depressurization Valve (page 1 of 8)

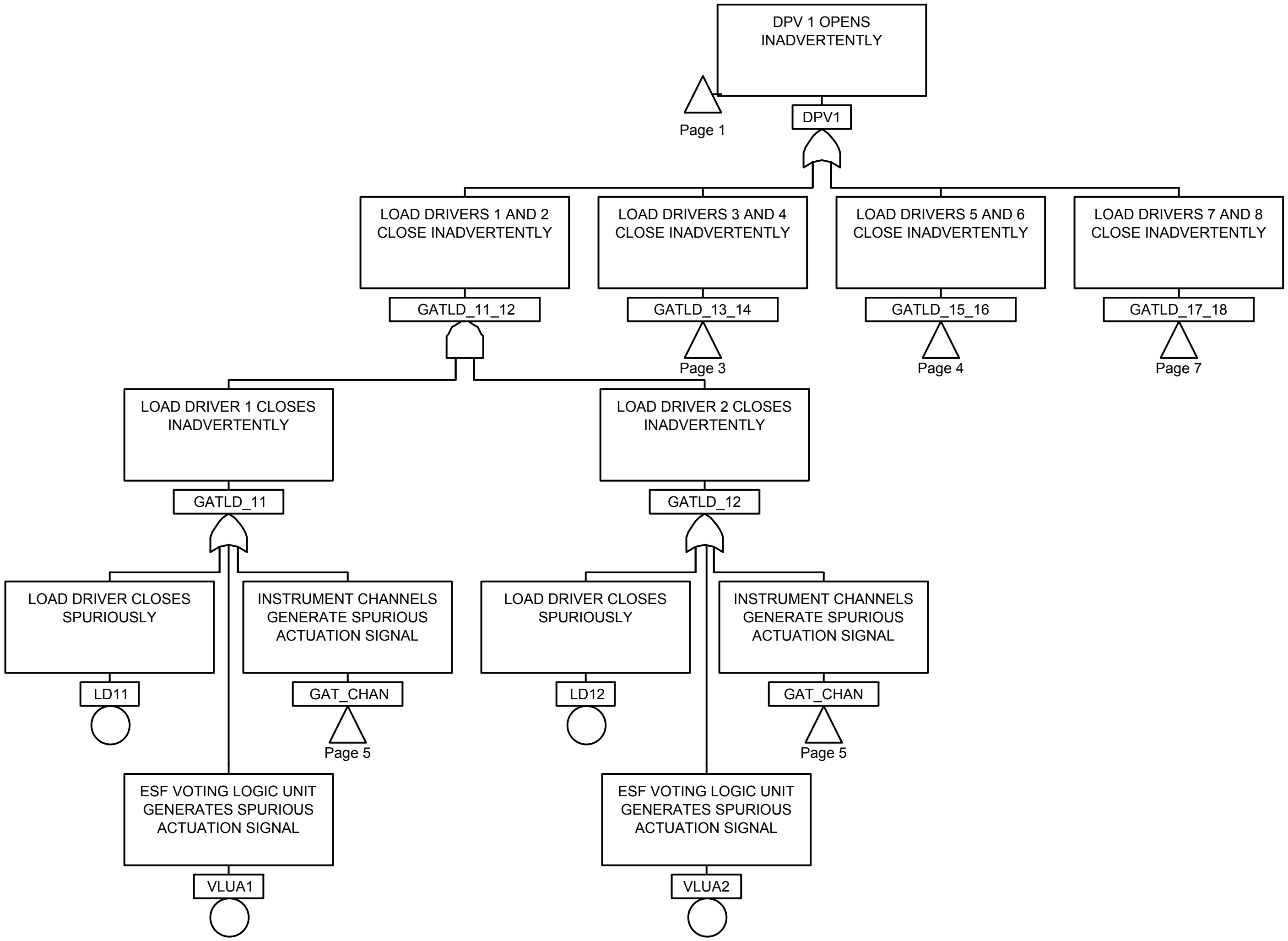


Figure 15A-2b. Fault Tree – Inadvertent Opening of a Depressurization Valve (page 2 of 8)

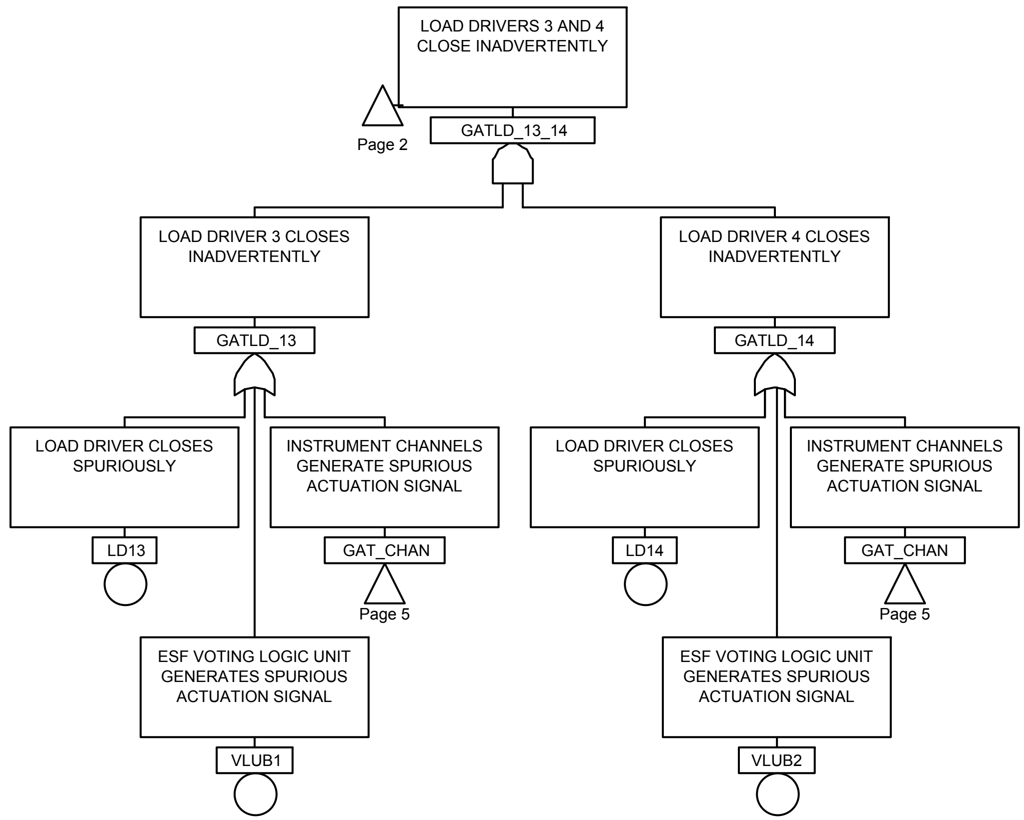


Figure 15A-2c. Fault Tree – Inadvertent Opening of a Depressurization Valve (page 3 of 8)

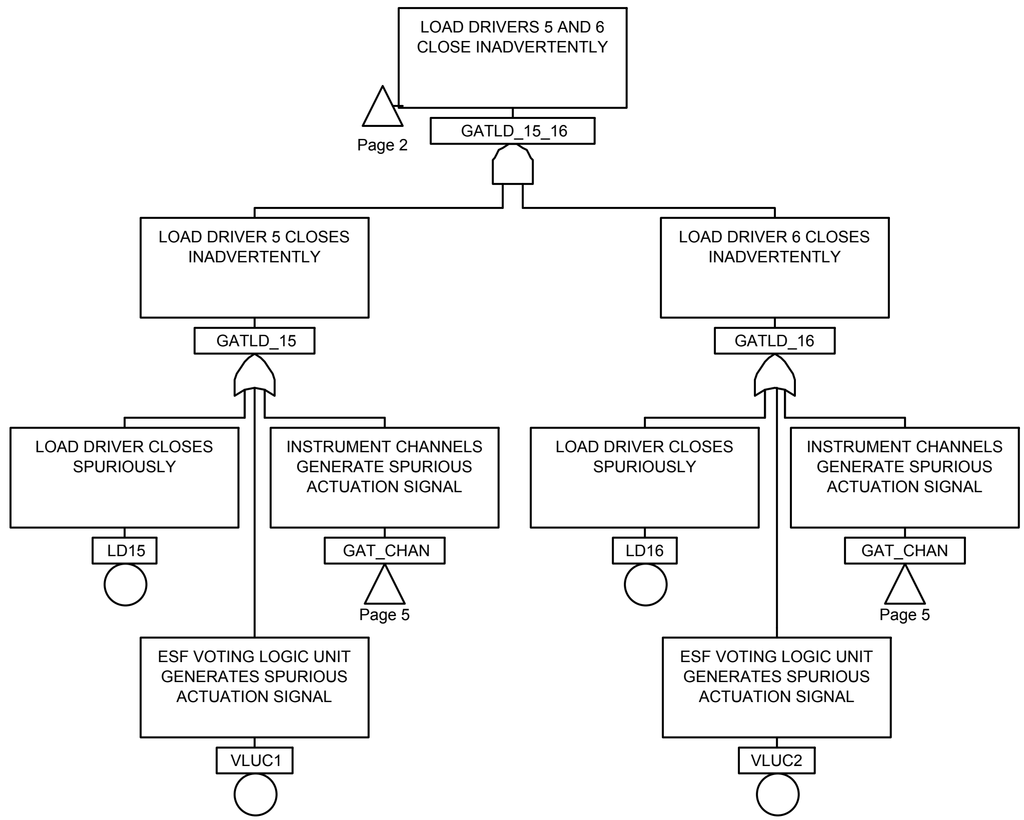


Figure 15A-2d. Fault Tree – Inadvertent Opening of a Depressurization Valve (page 4 of 8)

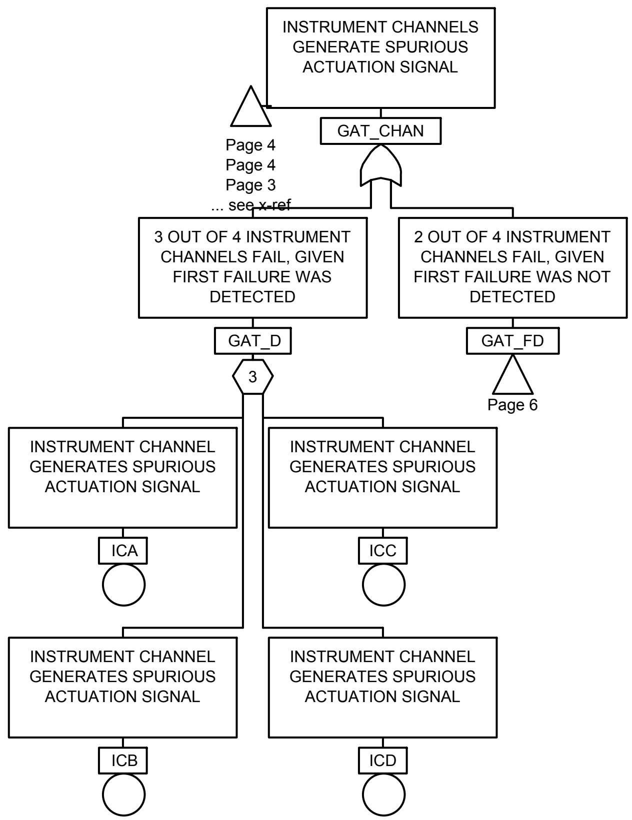


Figure 15A-2e. Fault Tree – Inadvertent Opening of a Depressurization Valve (page 5 of 8)

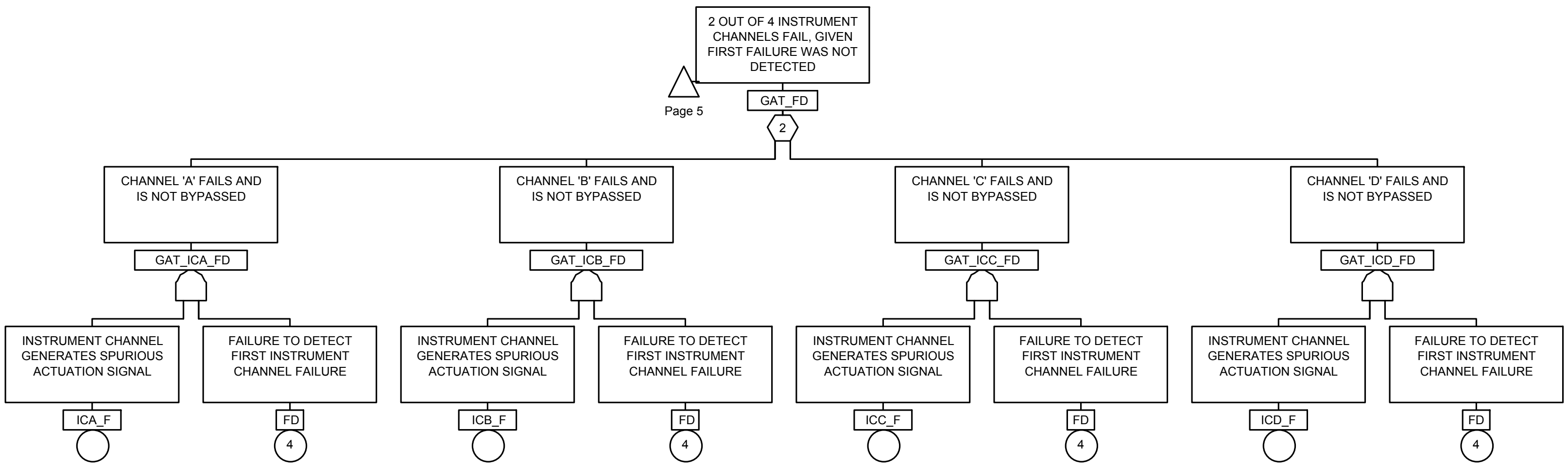


Figure 15A-2f. Fault Tree – Inadvertent Opening of a Depressurization Valve (page 6 of 8)

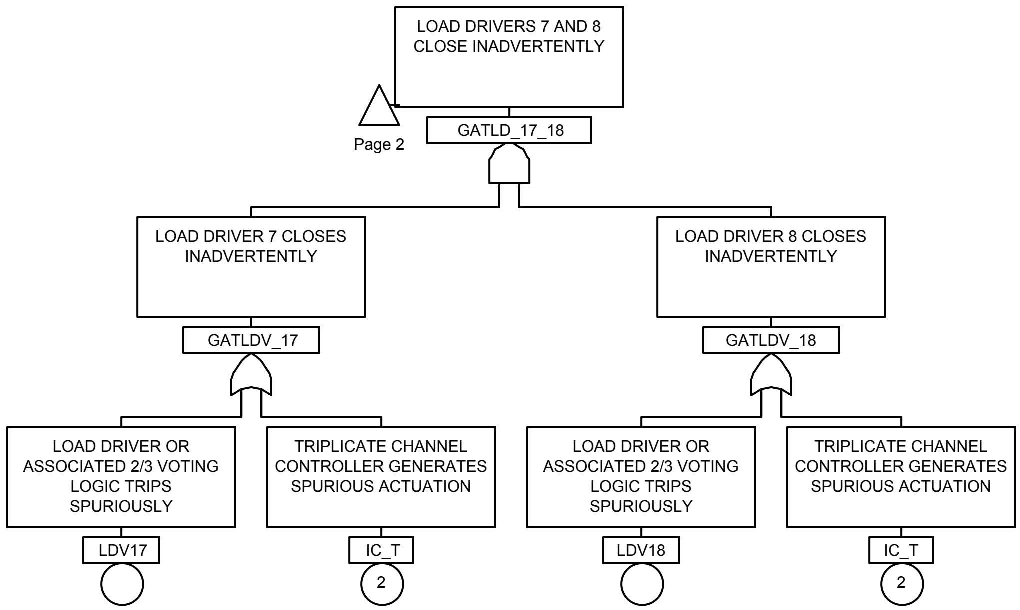


Figure 15A-2g. Fault Tree – Inadvertent Opening of a Depressurization Valve (page 7 of 8)

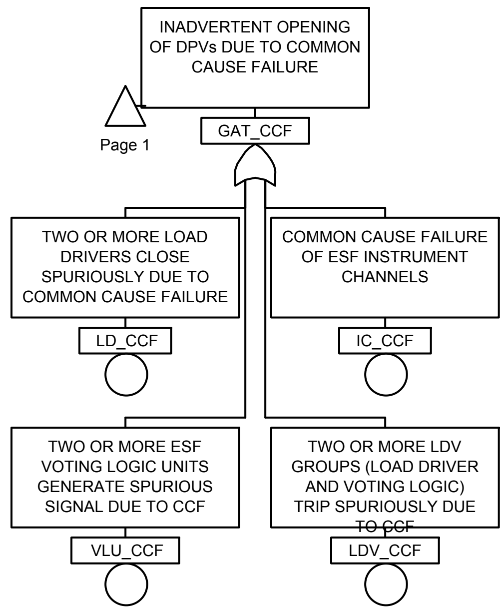


Figure 15A-2h. Fault Tree – Inadvertent Opening of a Depressurization Valve (page 8 of 8)

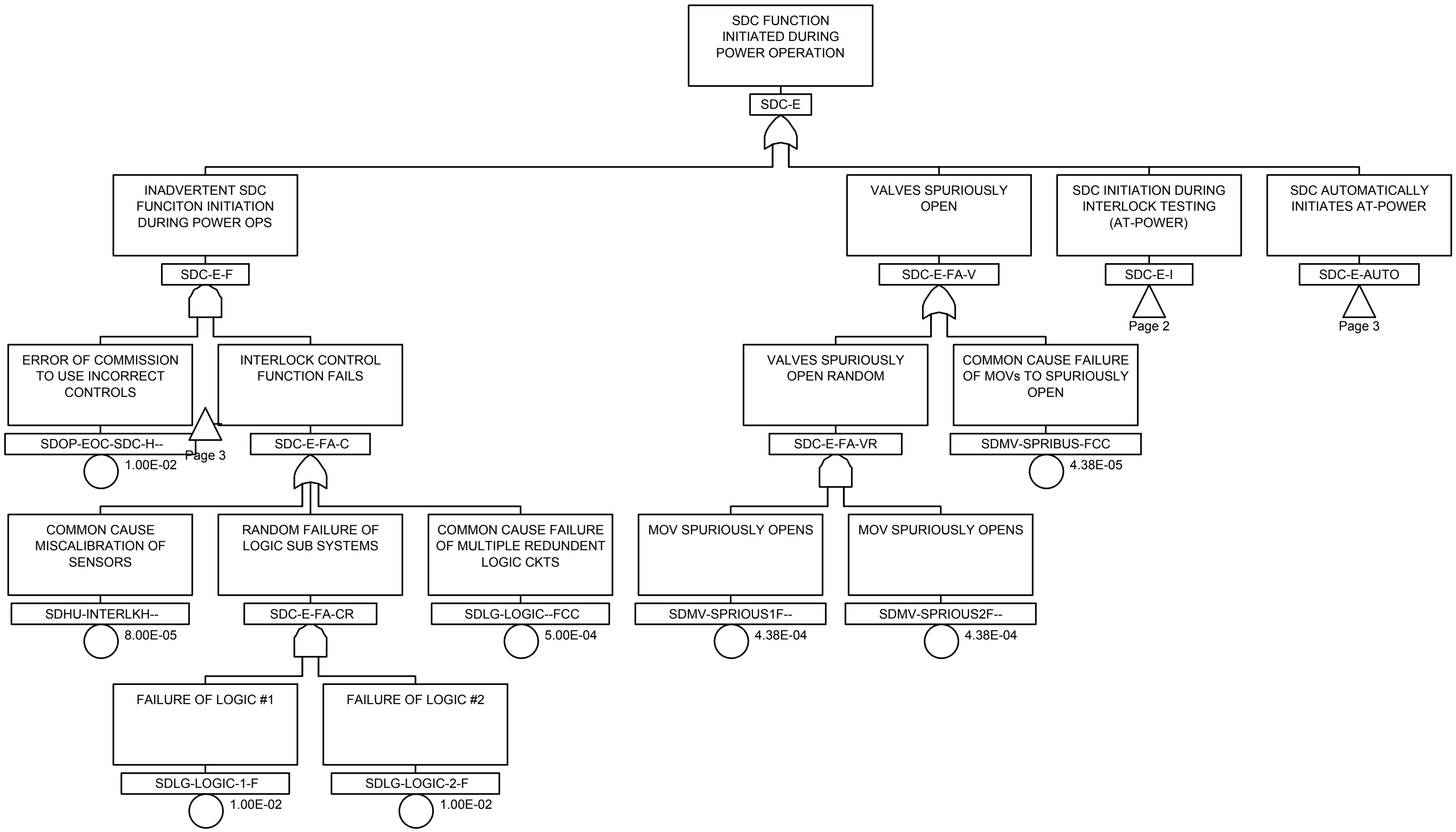


Figure 15A-3a. Fault Tree – Inadvertent Shutdown Cooling Function Operation (page 1 of 3)

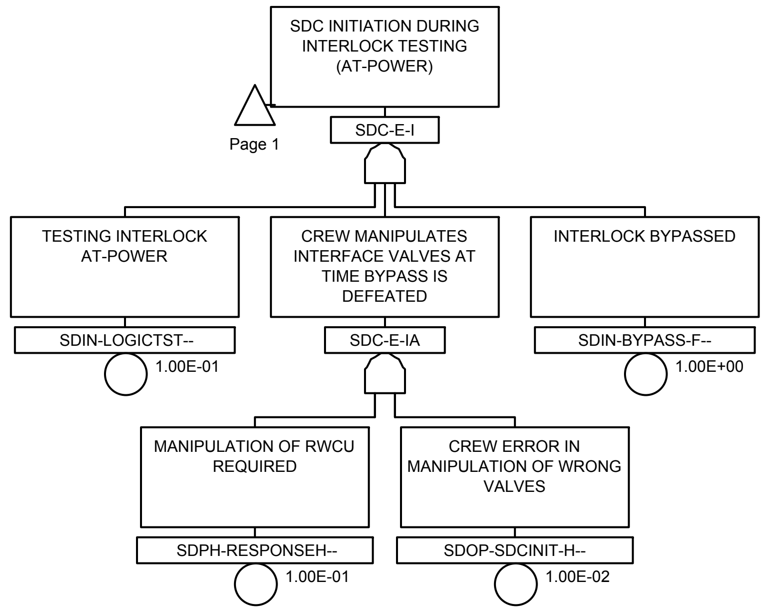


Figure 15A-3b. Fault Tree – Inadvertent Shutdown Cooling Function Operation (page 2 of 3)

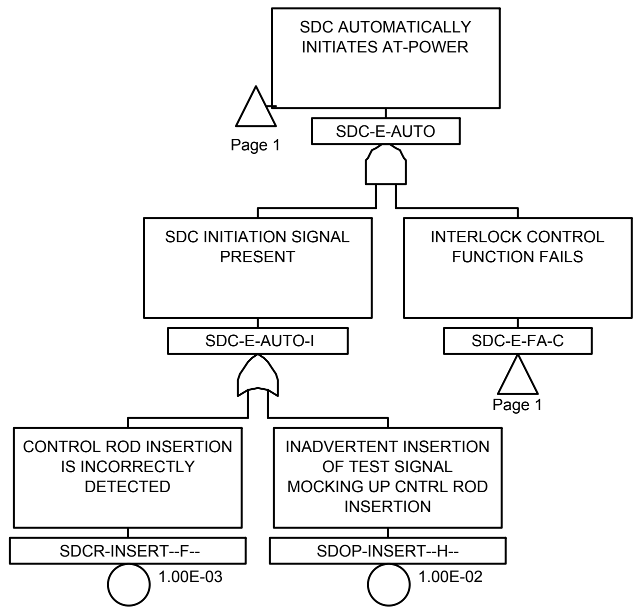


Figure 15A-3c. Fault Tree – Inadvertent Shutdown Cooling Function Operation (page 3 of 3)

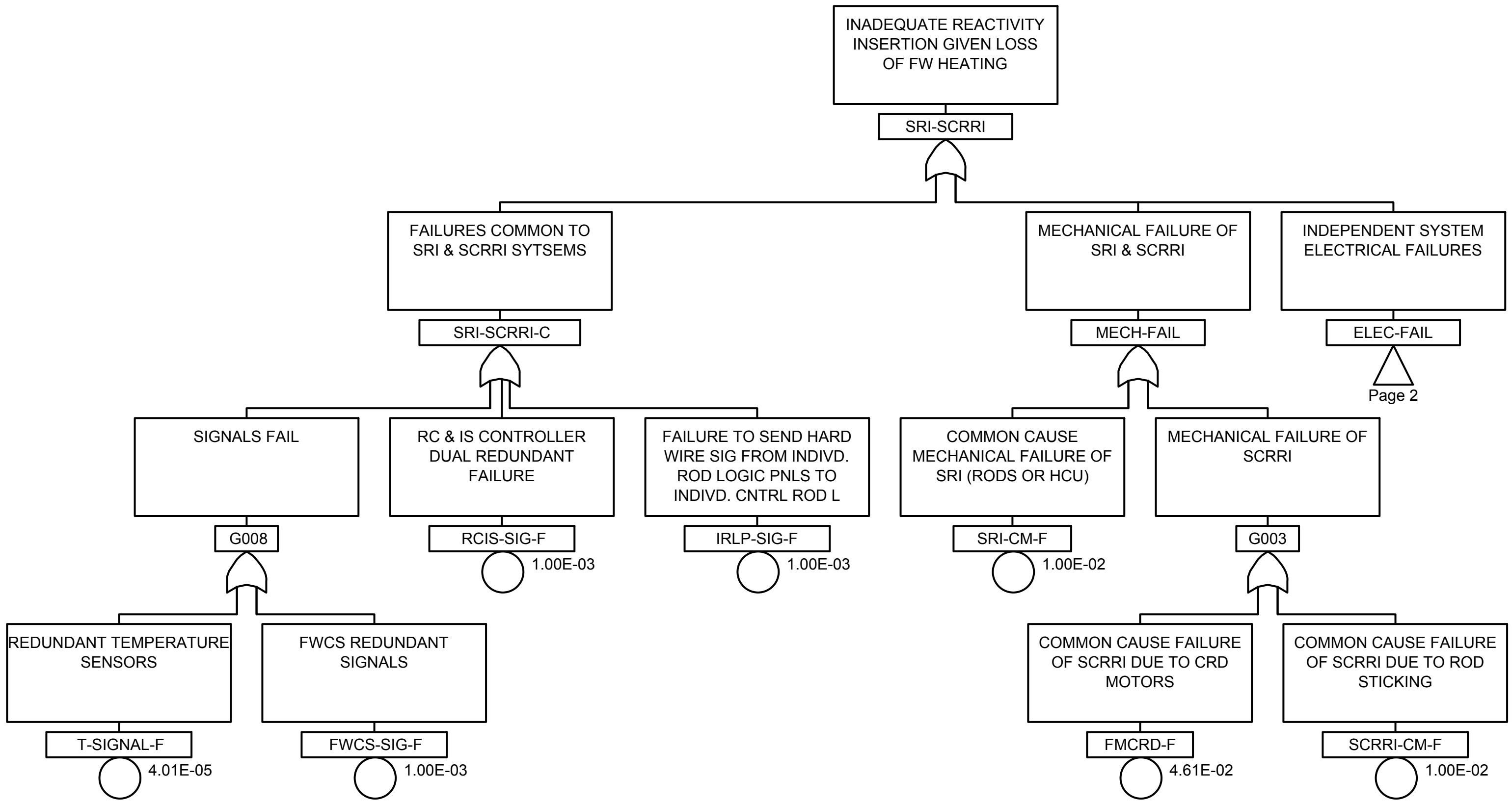


Figure 15A-4a. Fault Tree - Inadequate Reactivity Insertion Given a Loss of FW Heating (page 1 of 2)

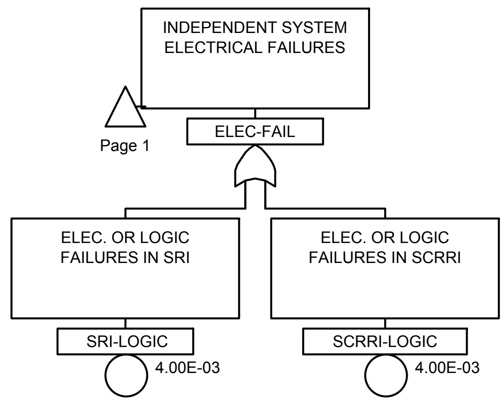


Figure 15A-4b. Fault Tree for Inadequate Reactivity Insertion Given a Loss of FW Heating (page 2 of 2)

15B. LOCA INVENTORY

This appendix provides additional detail on the design basis core source term assumed in the Chapter 15 dose consequence analyses. The source term was calculated using the computer code ORIGEN2 (Reference 15B-1). The source term meets the requirements of Regulatory Guide 1.183, Section 3.1.

The design power level for the ESBWR is 4500 MWt for a core with 1132 shortened GE14 fuel bundles. Considering a licensing power 2% above the design level gives a total core power of 4590 MWt or a bundle average power level of 4.054 MWt/bundle. The core inventory for licensing basis evaluations is based on the GE14 bounding bundle inventory. This inventory is based on a bundle enrichment of 4.6% and a core average exposure of 35 GWD/MTU. Also, it assumes a power level of 5.75 MWt/bundle. A full length GE14 bundle was used with a uranium mass of 182 kg, rather than the shorter bundle for the ESBWR, hence the higher bundle power assumption. The GE14 full length core inventory has been used for numerous power uprate licensing amendments. The linear heat generation rate is identical for both full length and ESBWR GE14 fuel. Also, when normalized to total length other parameters such as uranium mass are comparable. As such, use of a full length bundle has a negligible impact on the overall source term, thus the results are appropriate for the ESBWR.

Table 15B-1 contains values applicable to the ESBWR for the 60 isotopes used by the NRC computer code RADTRAD (Reference 15B-2).

15B.1 COL INFORMATION

None

15B.2 REFERENCES

15B-1 CCC-371, "RSICC Computer Code Collection – ORIGEN 2.1", Oak Ridge National Laboratory, May 1999.

15B-2 NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," USNRC, April 1998.

Table 15B-1

ESBWR Core Concentrations

	Isotope	Conc. (MBq/MWth)		Isotope	Conc. (MBq/MWth)		Isotope	Conc. (MBq/MWth)
1	Co-58	5.10E+06	21	Ru-103	1.50E+09	41	Cs-136	6.89E+07
2	Co-60	4.92E+06	22	Ru-105	1.00E+09	42	Cs-137	1.28E+08
3	Kr-85	1.23E+07	23	Ru-106	5.21E+08	43	Ba-139	1.84E+09
4	Kr-85m	2.73E+08	24	Rh-105	9.10E+08	44	Ba-140	1.77E+09
5	Kr-87	5.27E+08	25	Sb-127	1.03E+08	45	La-140	1.82E+09
6	Kr-88	7.42E+08	26	Sb-129	3.15E+08	46	La-141	1.68E+09
7	Rb-86	2.35E+06	27	Te-127	1.05E+08	47	La-142	1.62E+09
8	Sr-89	9.93E+08	28	Te-127m	1.37E+07	48	Ce-141	1.68E+09
9	Sr-90	9.76E+07	29	Te-129	3.10E+08	49	Ce-143	1.56E+09
10	Sr-91	1.25E+09	30	Te-129m	4.60E+07	50	Ce-144	1.36E+09
11	Sr-92	1.34E+09	31	Te-131m	1.42E+08	51	Pr-143	1.53E+09
12	Y-90	1.04E+08	32	Te-132	1.41E+09	52	Nd-147	6.69E+08
13	Y-91	1.27E+09	33	I-131	9.90E+08	53	Np-239	1.93E+10
14	Y-92	1.35E+09	34	I-132	1.44E+09	54	Pu-238	3.34E+06
15	Y-93	1.55E+09	35	I-133	2.04E+09	55	Pu-239	4.02E+05
16	Zr-95	1.79E+09	36	I-134	2.25E+09	56	Pu-240	5.21E+05
17	Zr-97	1.85E+09	37	I-135	1.91E+09	57	Pu-241	1.51E+08
18	Nb-95	1.80E+09	38	Xe-133	2.03E+09	58	Am-241	1.70E+05
19	Mo-99	1.90E+09	39	Xe-135	6.72E+08	59	Cm-242	4.01E+07
20	Tc-99m	1.68E+09	40	Cs-134	1.98E+08	60	Cm-244	1.94E+06

Figure 15B-1. Iodine Airborne Inventory in Primary Containment as a Function of Time (Deleted)