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CHAPTER 15 ACCIDENT ANALYSES

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CHAPTER 15.0 ACCIDENT ANALYSES

15.0 GENERAL

This section describes the categorization and identification of events which were analyzed by the NSSS vendor for the initial fuel cycle. The reload fuel vendor has determined that certain events are limiting which require analysis for each fuel loading cycle. A discussion of the event analyses which have been updated for the current cycle is provided below in subsection 15.0.3.

In this chapter the effects of anticipated process disturbances and postulated component failures are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and events.

The scope of the situations analyzed includes anticipated (expected) operational occurrences (e.g., loss of electrical load), off-design abnormal (unexpected) transients that induce system operations condition disturbances, postulated accidents of low probability (e.g., the sudden loss of integrity of a major component), and finally hypothetical events of extremely low probability (e.g., an anticipated transient without the operation of the entire control rod drive system).

15.0.1 Analytical Objective

The spectrum of postulated initiating events is divided into categories based upon the type of disturbance and the expected frequency of the initiating occurrence; the limiting events in each combination of category and frequency are quantitatively analyzed. The plant safety analysis evaluates the ability of the plant to operate within regulatory guidelines, without undue risk to the public health and safety.

15.0.2 Analytical Categories

Transient and accident events contained in this report are discussed in individual categories as required by Reference 1. Each event evaluated is assigned to one of the following applicable categories:

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

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

c. Decrease in reactor core coolant flow rate: A reduction in the core coolant flow rate threatens to overheat the cladding as the coolant becomes unable to adequately remove the heat generated by the fuel.

d. Reactivity and power distribution anomalies: Transient events included in this category are those which cause rapid increases in power which are due to increased core flow disturbance events. Increased core flow reduces the void content of the moderator increasing core reactivity and power level.

e. Increase in reactor coolant inventory: Increasing coolant inventory could result in excessive moisture carryover to the main turbine, feedwater turbines, etc.

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

g. Radioactive release from a subsystem or component: Loss of integrity of a radioactive containment component is postulated.

h. Anticipated transients without scram: In order to determine the capability of plant design to accommodate an extremely low probability event, a multi-system maloperation plus multi single active component failures (SACF) situation is postulated.

15.0.3 Event Evaluation

The NSSS vendor examined the effects of anticipated process disturbances and postulated component failures to determine the ability of the plant to control or accommodate these failures and events. The process of examining these disturbances and failures resulted in the identification of a number of transients and accidents which were analyzed in detail. These are discussed in the appropriate sections of Chapter 15. ECCS performance is addressed in Section 6.3. The NSSS vendor performed additional analyses of some of the Chapter 15 events to justify operation with feedwater heaters out of service (FWHOS), Single Loop Operation (SLO), and maximum extended operating domain (MEOD). These analyses are documented in References 9,4,and 21, respectively, and discussed in Appendices 15B, 15C, and 15D, respectively.

The reload fuel vendor performed an assessment of the Chapter 15 events. The assessment showed that while several transients are inherently non-limiting, others would have to be re-evaluated for each fuel cycle. Only the events in this subset were re-evaluated by the reload fuel vendor for the current cycle. The events that require re-evaluation on a cycle-specific basis for GGNS are:

- MSIV Closure (flux scram only) (Section 5.2.2)
- Loss of Feedwater Heating (LOFWH) (Section 15.1.1)
- Feedwater Controller Failure (FWCF) (Section 15.1.2)
- Pressure Controller Failure-Closed (Pressure Control Downscale Failure) (Section 15.2.1)
- Generator Load Reject w/o Bypass (LRNB) (Section 15.2.2)
- Turbine Trip w/o Bypass (TTNB) (Section 15.2.3)
- Rod Withdrawal Error (Section 15.4.1)
- Control Rod Withdrawal Error (CRWE) (Section 15.4.2)
- Fuel Loading Error (Section 15.4.7)
- Control Rod Drop Accident (CRDA) (Section 15.4.9)

Details of the evaluations are addressed in the individual UFSAR sections for each event listed above. A summary of results for the current cycle are provided in Reference 5. The LOCA analyses performed by the reload fuel vendor are addressed in Section 6.3. The MAPLHGR reduction factor for SLO is addressed in Section 6.3.

The overpressurization analysis was performed for current cycle conditions as part of the MSIV closure event (flux scram only) to ensure that the peak vessel pressure is within the ASME code limit. Overpressure protection is discussed in subsection 5.2.2.

Based on the above discussions of event analyses performed for the initial cycle by the NSSS vendor and event analyses performed by the reload fuel vendor for each core reload cycle, the analysis of record for each event is described in their respective sections in Chapter 15. For events which are not re-analyzed for each refueling cycle, the original NSSS vendor analysis represents the current licensing basis because it has been determined to envelop any subsequent analyses. However, the calculational data and results showed in the tables and figures have been noted to contain initial cycle data information because the results are based on the initial licensed power of 3833 MWt and, therefore, does not represent results for the current licensed power of 4408 MWt. For event analyses performed by the reload fuel vendor for each core reload, the current cycle analysis represents the current licensing basis and is described in the respective Chapter 15 section. A description of the original evaluation by the NSSS vendor is also provided in Subsection 6 (i.e., 15.X.X.6) of that respective section and is designated as historical information. The corresponding initial cycle tables and figures are identified as historical, but are retained for informational purposes.

15.0.3.1 Identification of Causes and Frequency Classification

Situations and causes which lead to the initiating event analyzed are described within the categories designated above. The frequency of occurrence of each event is summarized based upon currently available operating plant history for the transient event. Events for which inconclusive data exists are discussed separately within each event section.

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

a. Incidents of moderate frequency - these are incidents that may occur during a calendar year to once per 20 years for a particular plant. This event is referred to as an anticipated (expected) operational transient.

b. Infrequent incidents - these are incidents that may occur during the life of the particular plant (spanning once in 20 years to once in 100 years). This event is referred to as an abnormal (unexpected) operational transient.

c. Limiting faults - these are occurrences that are not expected to occur but are postulated because their consequences may result in the release of significant amounts of radioactive material. This event is referred to as a design basis (postulated) accident.

d. Normal operation - operations of high frequency are not discussed here but are examined along with a, b, and c in the nuclear systems operational analyses in Appendix 15A to Chapter 15.

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

The following are considered to be unacceptable safety results for incidents of moderate frequency (anticipated operational transients):

a. A release of radioactive material to the environs that exceeds the limits of 10 CFR 20.

b. Reactor operation induced fuel cladding failure

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

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

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

The following are considered to be unacceptable safety results for infrequent incidents....abnormal operational transients:

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

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

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

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

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

The following are considered to be unacceptable safety results for limiting faults....design basis accidents:

a. Radioactive material release which results in dose consequences that exceed the guideline values of 10 CFR 50.67

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

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

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

e. Radiation exposure to plant operations personnel in the main control room in excess of five Rem total effective dose equivalent in accordance with 10 CFR 50.67.

15.0.3.2 Sequence of Events and Systems Operations

Each transient or accident is discussed and evaluated in terms of:

a. A step-by-step sequence of events from initiation to final stabilized condition

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

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

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

e. The operation of engineered safety systems that is required

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

Additionally, sequence of event tables are included and list the opening and closing times of valves significant to the analysis such as turbine control valves, turbine stop valves, main steam bypass valves, MSIV and safety/relief valves. The times corresponding to the reactor vessel water level trips; e.g., L9 turbine and feedwater trip, L3 scram, and L2 recirculation pumps trip and initiation of RCIC and HPCS; are also included in the sequence of events tables. The associated delay times are applied consistently to the relevant events. The function of referenced valves and the action initiated by the water level trips are important while evaluating the consequences of transients. Therefore, the associated timings are necessary in the sequence of events tables. However, since operator action is not usually necessary following a water level alarm, the documentation of the time at which the water level alarm is attained is considered to be unnecessary. Should any event include operator action for a protection function, the timing of the alarm signal(s) and assumed operator response time would be given.

15.0.3.2.1 Single Failures or Operator Errors

15.0.3.2.1.1 General

This subsection discusses a very important concept pertaining to the application of single failures and operator errors analyses of the postulated events. Single active component failure (SACF) criteria have been required and successfully applied on past NRC approved docket applications to design basis accident categories only. Reference 1 infers that a "single failures and operator errors" requirement should be applied to transient events (both high, moderate, and low probability occurrences) as well as accident (very low probability) situations.

Transient evaluations have been judged against a criterion of one single equipment failure "or" one single operator error as the initiating event with no additional single failure assumptions to the protective sequences although a great majority of these protective sequences utilized safety systems which can accommodate SACF aspects. Even under these postulated events, the plant damage allowances or limits were very much the same as those for normal operation.

Reference 1 suggests that the transient and accident scenarios should now include “and” (multi-failure) event sequences. The format request follows:

<u>For initiating occurrence</u>	1) an equipment failure or an operator error, and
<u>For single equipment failure or operator error analysis</u>	2) another equipment failure or failures and/or another operator error or errors

This certainly is considered a new requirement and the impact will need to be completely evaluated. While this is under consideration GEH has evaluated and presented the transients and accidents in this chapter in the above new requirement manner.

Event categorization relative to transient and accident analysis is discussed here. If the evaluation is done per the new multi-failure methods, the event frequency categories should be modified.

The original categorization of events was based on frequency of the initiating event alone and thus the allowance or limit was accordingly established based on that high frequency level. With the introduction of additional assumptions and conditions (initial event and single component failure and/or single operator error), the total event would now fall into a lower frequency/probability category. Thus, less restrictive limits or allowances should be applied in the analysis of transients and accidents. This certainly needs to be considered and evaluated.

GEH has evaluated and presented the transients and accidents in this chapter by the more restrictive old allowances and limits of the event categorization presently in effect.

Most events postulated for consideration are already the results of single equipment failures or single operator errors that have been postulated during any normal or planned mode of plant operations. The types of operational single failures and operator errors considered as initiating events and subsequent protective sequence challenges are identified in the following paragraphs:

15.0.3.2.1.2 Initiating Event Analysis

One of the following is considered an initiating event:

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

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

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

d. Any single electrical component failure.

e. Any single operator error.

Operator error is defined as an active deviation from written operating procedures or nuclear plant standard operating practices. A single operator error is the set of actions that is a direct consequence of a single erroneous decision. The set of actions is limited as follows:

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

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

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

Examples of single operator errors are as follows:

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

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

c. An incorrect calibration of an average power range monitor.

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

15.0.3.2.1.3 Single Active Component Failure (SACF) or Single Operator Failure (SOF) Analysis

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

or

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

15.0.3.2.1.4 Operator Response

With the exception of Anticipated Transients Without Scram (ATWS), the design and protection basis assumes no operator action for 10 minutes. A lapse time of 10 minutes for these situations is considered appropriate. The necessity and justification of the operator corrective actions are discussed below. The ATWS event analysis credits certain Time Critical Operator Actions as described in Section 15.8.

All immediate short-term Design Basis Accident (DBA) event safety functions are automatic as well as manual. For NSSS-ESF systems and equipment, long-term safety actions might involve operator action at or after the 10-minute mark (as previously allowed and approved by the NRC). Long-term required NSSS-ESF action can obviously be met since they involve the same equipment as safety function systems.

For all anticipated operational occurrences cited in Chapter 15, no operator corrective action is required to prevent the plant from exceeding safety design basis limits.

Any operator action will be taken in accordance with approved site specific emergency operating procedures and their supporting instructions. [HISTORICAL INFORMATION] [The site specific emergency procedures were developed from the Boiling Water Reactor Owner's Group Emergency Procedure Guidelines in accordance with NUREG-0737. These procedures are symptomatic in nature and provide adequate guidance to the operators to maintain adequate core cooling, shut down the reactor, cool down the RPV to cold shutdown condition, protect the equipment in the primary containment with respect to the consequences of all mechanistic events, maintain secondary containment or limit radioactive release from the secondary containment, and limit radioactive release into areas outside the primary and secondary containments: Because these procedures are symptomatic in nature and the actual event will not exactly follow the design bases assumptions, it is impossible to predict the exact operator actions for any event. Vital parameters are monitored by the operators, and the action taken by the operator is based on the magnitude and direction of change of the parameters as directed by the Emergency Operating Procedures.]

In summary, the general rules utilized in BWR technology include the following:

For a DBA, no operator action is taken prior to 10 minutes.

For a transient, immediate operator action is allowed to preclude unwarranted shutdown, ESF operation, unnecessary operation, and other non-safety actions. For a hypothetical event of extreme low probability (e.g., ATWS) certain operator actions are credited as described in Section 15.8.

The 10-minute restriction for operator action for DBAs has been justifiable since the safety actions required are limited and require simple control initiations.

To address concerns expressed by the NRC for the 10- to 20-minute time frame, the only operator actions assumed for the LOCA analysis in the 10- to 20-minute time frame are:

- DBA LOCA assumes the operator diverts partial ECCS core cooling to containment cooling. This is a conservative assumption in that flow into the core is being diverted.

15.0.3.3 Core and System Performance

15.0.3.3.1 Introduction

Section 4.4, Thermal and Hydraulic Design, describes the various fuel failure mechanisms. Avoidance of unacceptable safety limits (subsection 4.4.1.4) for incidents of moderate frequency is verified statistically with consideration given to date, calculation, manufacturing, and operating uncertainties. An acceptable criterion was determined to be that 99.9 percent of the fuel rods in the core would not be expected to experience boiling transition (Ref. 2). This criterion is met by demonstrating that incidents of moderate frequency do not result in a minimum critical power ratio (MCPR) less than the MCPR safety limit. The reactor steady-state CPR operating limit is derived by determining the decrease in MCPR for the most limiting event. All other events result in smaller MCPR decreases and are not reviewed in depth in this chapter. The MCPRs during significant abnormal events are calculated using transient core heat transfer analysis computer programs. The computer programs are based on multinode, single channel thermal-hydraulic models which require simultaneous solution of the partial differential equations for the conservation of mass, energy, and momentum in the bundle, and which accounts for axial variation in power generation. The primary inputs to the models include a physical description of the bundle, and channel inlet flow and enthalpy, pressure and power generation as functions of time.

As discussed in subsection 4.4.1, in order to ensure that fuel cladding integrity is maintained during the process disturbances and component failures, MCPR operating limits, maximum average planar linear heat generation rates (MAPLHGR), and linear heat generation rates (LHGR) multiplication factors have been established.

The MCPR safety limit will be maintained if it can be shown that when it is added to the maximum change in CPR (ΔCPR) for any event the sum is still less than or equal to the MCPR operating limit. That is,

$$MCPR_{OPERATING\ LIMIT} \geq MCPR_{SAFETY\ LIMIT} + \Delta CPR$$

If this condition is met, there is adequate margin between the operating limit and the safety limit so that the safety limit will not be compromised for any event. The largest ΔCPR for the limiting events analyzed is used to establish the MCPR operating limit.

The value of ΔCPR is a function of power, flow and fuel exposure (core life). At lower flows, lower powers, and higher exposures, the value of ΔCPR has been found to be higher. Flow, power and exposure dependent MCPR limits have been established as a result of the analyses performed.

The MAPLHGR limits are used as input to the LOCA ECCS analyses.The LOCA ECCS analyses confirm the acceptability of the limits on MAPLHGR.

The design LHGRs for each fuel type are determined by the mechanical design of the fuels. The limits ensure that the design LHGRs are not compromised. The power and flow dependent LHGR multiplication factors are determined by the transient analyses for reduced power and flow conditions. These factors when applied to the LHGR limit ensure that the design LHGRs will not be exceeded for any event.

The methodology and major codes used by the reload vendor for analyses performed in support of the current fuel reload are discussed in References 5, 6, 23, 24, and 25.

Statistical analyses were performed by the reload vendor for the control rod withdrawal error (RWE) (Section 15.4.2) events.

The methodologies are also addressed in each of the sections describing the events.

For situations in which fuel damage is sustained, the extent of damage is determined by correlating fuel energy content, cladding temperature, fuel rod internal pressure, and cladding mechanical characteristics.

These correlations are substantiated by fuel rod failure tests and are discussed in Section 4.4, Thermal and Hydraulic Design, and Section 6.3, Emergency Core Cooling Systems.

The relief mode of 6 valves and the safety mode of 9 valves for safety/relief valve (SRV) actuation has been applied to current cycle Chapter 15 transient pressurization events. There is no requirement to assume simultaneous failure of these valves for the transient assessment. Any increase in peak pressure is addressed by the bounding, worst ASME Code case analysis presented in Chapter 5. These analyses show that overpressure protection is provided even for the worst cases when credit is only taken for accepted ASME valve operation.

All equipment and components required for safety/relief valve actuation are safety grade.

The operating limit MCPR is not impacted at rated conditions by using less than 100 percent capacity of relief actuation of safety/relief valves in Chapter 15. However, at some off-rated conditions (less than full power) the assumption of only 6 SRVs in relief mode and 9 SRVs in safety mode increases the operating limit MCPR. The peak pressure for each Chapter 15 analysis is bounded by the worst ASME code case presented in Chapter 5. The design SRV grouping and setpoints are addressed in detail in Section 5.2.

15.0.3.3.2 Input Parameters and Initial Conditions for Analyzed Events

In general, the events analyzed by the reload fuel vendor within Chapter 15 have values for input parameters and initial conditions as listed in Table 15.0-4.

Input parameters and initial conditions used by the NSSS vendor in the FWHOS, SLO, and MEOD analysis are specified in Appendices 15B, 15C and 15D, respectively. Analyses which assume data inputs different than these values are designated accordingly in the appropriate event discussion. Refer to Section 15.1.3.6 for a description of the input values and initial conditions used by the NSSS vendor in the initial cycle analyses.

The input parameters and initial conditions used for the reload transient analysis are listed in Table 15.0-4.

For the reload vendor analysis, the scram reactivity, doppler, and void coefficients are calculated internally by the code. Furthermore, the reload vendor analysis assumes the slowest allowable control rod insertion time based on measured mean scram times or on the Technical Specification.

15.0.3.3.3 Power/Flow Operating Constraints

The analyses basis for most of the transient safety analyses performed by the NSSS vendor for the initial cycle, and by the reload fuel vendor for the current cycle, is the thermal power at rated core flow (100 percent) corresponding to 105 percent of the licensed nuclear boiler rated steam flow. This operating point is the apex of a bounded operating power/flow map which, in response to any classified abnormal operational transients, will yield the minimum pressure and thermal margins of any operating point within the bounded map.

The power/flow domain for the initial fuel cycle is described in Section 15.0.3.6. Expanded power/flow domains have been defined for GGNS subsequent to plant startup to improve operational flexibility while at the same time complying with all regulatory requirements. This includes the maximum extended operating domain (MEOD), the MELLLA extended operating domain, and the MELLLA+

extended operating domain and is described in Appendix 15D. All normal and abnormal transients and all design basis accidents were evaluated to show that the applicable regulatory requirements are met when operating within the expanded operating domains. The expanded operating domains include additional regions beyond the 100% power/100% flow domain shown in Figure 15.D.3-1 for the initial cycle and in the COLR for the current fuel cycle. The regions are: (a) above the 100% rod line (the extended load line regions of MELLLA and MELLLA+) and (b) to the right of the 100% flow line (the increased core flow region).

The reload fuel vendor determined the limiting events for the expanded operating domain and re-evaluated these events in detail at various state points for the current cycle [5]. Transient analyses were performed for each of the events that require re-evaluation on a cycle specific basis as identified in Section 15.0.3 at several power/flow statepoints within the expanded power/flow map, as well as at current exposure conditions, for validation of operating limits. Furthermore, various operational flexibility options and equipment out-of-service considerations have been evaluated by the reload fuel vendor as described below and in Reference 5. Refer to the appropriate subsections for discussion of the power/flow and exposure conditions examined for each transient.

GGNS can be operated with one active recirculation pump. This operating condition is called Single Loop Operation (SLO). The reload fuel vendor evaluated the effects of SLO on accident and abnormal operational transients (Appendix 15C). The current cycle analysis of the SLO pump seizure event was performed for a power/flow condition of 61.4% current licensed rated power and 54.1% core flow. This state point bounds the SLO operating domain (References 6 and 25) hence, SLO is not allowed in the MELLLA+ operating domain. For the current cycle, the SLO pump seizure event and other events were evaluated and found to not be limiting for MCPR values. The Two Loop LHGRFAC limits for reduced power and flow were found to be appropriate for SLO for the current cycle.

Other operating flexibility options were evaluated for the current cycle which included feedwater temperature reduction, safety/relief valves out of service (OOS), automatic depressurization system (ADS) OOS, end of cycle recirculation pump trip, main steam valve OOS, and turbine bypass valves OOS. Operation with one or more feedwater heaters out of service (FWHOS) is allowed so long as the reduction in feedwater temperature entering the reactor vessel at rated conditions does not exceed 100°F. For the current cycle, transient analyses were evaluated at reduced feedwater temperatures for the FWCF, LRNB, and TTNB at several exposure points and power flow levels. The results of these analyses are reported in Reference 5.

Analytical evaluations were performed by the NSSS vendor for each event in Chapter 15.

15.0.3.3.4 Results

The cycle-specific results applicable to the limiting transients for the current cycle, as identified by the reload fuel vendor, are provided in Reference 5. The results are based on the reload fuel vendor analysis in the power/flow domain documented for the current cycle in the COLR and provide the current cycle maximum ΔCPR for the limiting events. Additional results at conditions different than rated are provided in the appropriate subsections of this chapter.

15.0.3.4 <u>Barrier Performance</u>

This section primarily evaluates the performance of the reactor coolant pressure boundary (RCPB) and the containment system during transients and accidents.

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

15.0.3.5 <u>Radiological Consequences</u>

In this subsection, the consequences of radioactivity release during the three types of events: a) incidents of moderate frequency (anticipated operational transients), b) infrequent incidents (abnormal operational transients), and c) limiting faults (design basis accidents) are considered. For all events whose consequences are limiting a detailed quantitative evaluation is presented. For non-limiting events a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

For limiting faults (design basis accidents) two quantitative analyses are considered:

a. The first is based on conservative assumptions considered to be acceptable to the NRC for the purposes of worst case bounding the event and determining the adequacy of the plant design to meet 10 CFR Part 50.67 guidelines. This analysis is referred to as the design basis analysis, and is presented in this chapter of the FSAR.

b. The second is based on realistic assumptions considered to reflect expected radiological consequences. This analysis is referred to as the realistic analysis, and is presented in the Environmental Report.

15.0.3.6 Initial Cycle Core and System Performance

As described in section 15.0.3, the reload fuel vendor identified those events that are limiting events which require analysis for the current reload cycle. For those events that do not require analysis for the current reload cycle, the initial cycle analysis remains the current analysis of record. This subsection describes the analysis performed by the NSSS vendor for the initial fuel cycle.

In the initial cycle, the power/flow operating domain used for base case transient analyses was that shown in Figure 15.0-1. This was later augmented to include the MEOD and other extended load lines and the increased flow region as discussed in subsection 15.0.3.3.3. Referring to Figure 15.0-1, the apex of the bounded power/flow map is point A, the upper bound is the design flow control line (104.2 percent rod line A-D'), the lower bound is the zero power line H'-J', the right bound is the rated valve position line A-H', and the left bound is either the low pump speed, minimum valve position line D-J or the natural circulation line D'-J'.

The power/flow map, A-D'-J'-H-A, represents the acceptable operational constraints for abnormal operational transient evaluations.

Any other constraint which may truncate the bounded power/flow map must be observed, such as the recirculation valve and pump cavitation regions, the licensed power limit and other restrictions based on pressure and thermal margin criteria. For instance, if the licensed power is 100 percent nuclear boiler rated (NBR), the power/flow map is truncated by the line B-C and reactor operation must be confined within the boundary B-C-D'-J'-J-L-K-B. If the maximum operating power level has to be limited, such as point F, to satisfy pressure margin criteria, the upper constraint on power/flow is correspondingly reduced to the rod line, such as line F-G', which intersects the power/flow coordinate of the new operating basis. In this case, the operating bounds would be F-G'-J'-J-L-K-F. Operation would not be allowed at any point along line F-M, removed from point F, at the derated power but at reduced flow. If, however, operating limitations are imposed by GETAB derived from transient data with an operating basis at point A, the power/flow boundary for 100 percent NBR licensed power would be B-C-D'-J'-J-L-K-B. This power/flow boundary would be truncated by the MCPR operating limit for which there is no direct correlation to a line on the power/flow map. Operation is allowed within the defined power/flow boundary and within the constraints imposed by GETAB. If operation is restricted to point F by the MCPR operating limit, operation at point M would be allowed provided the MCPR limit is not violated.

Consequently, the upper operating power/flow limit of a reactor is predicated on the operating basis of the analysis and the corresponding constant rod pattern line. This boundary may be truncated by the licensed power and the GETAB operating limit.

Certain localized events were evaluated at other than the above mentioned conditions. These conditions are discussed pertinent to the appropriate event.

15.0.3.6.1 Identification of Cause and Frequency Classification (Initial Cycle)

The potential causes for the events for the initial cycle are the same as described for the current cycle. The probability of occurrence of the events is also the same as for the current cycle. Refer to subsection 15.0.3.1.

15.0.3.6.2 Sequence of Events and Systems Operation (Initial Cycle)

The sequence of event and systems operation description for the events for the initial cycle are the same as described for the current cycle. Refer to subsection 15.0.3.2.

15.0.3.6.3 Core and System Performance (Initial Cycle)

15.0.3.6.3.1 Introduction (Initial Cycle)

The discussion of the thermal hydraulic design approach for the events for the initial cycle are generally the same as described for the current cycle in Section 15.0.3.3.1. A detailed description of the NSSS vendor's analytical model may be found in Appendix C of Reference 2. For the initial core, the initial condition assumed for all full power transient MCPR calculations is that the bundle is operating at both the linear heat generation rate limit and at the MCPR operating limit. Maintaining MCPR greater than the safety limit is a sufficient, but not necessary, condition to assure that no fuel damage occurs. This is discussed in Section 4.4, Thermal and Hydraulic Design.

15.0.3.6.3.2 Input Parameters and Initial conditions for Analyzed Events (Initial Cycle)

In general the events analyzed for the initial core by the NSSS vendor within Chapter 15 have values for input parameters and initial conditions as specified in Table 15.0-2 for the REDY Code analyses (Ref. 1, Section 15.1.7) and Table 15.0-3 for the ODYN Code analyses (Ref. 2, Section 15.1.7).

Total RPS initial cycle response times used in both REDY and ODYN transient analysis codes are provided below.

Function	Total Response Time (Seconds)
IRM neutron flux	0.11[4]
APRM neutron flux	0.09
Reactor vessel high pressure	0.35
Reactor vessel low water level	0.30
	(1.05)[3]
MSLIV closure	0.06
MSL high radiation	1.05[4]
Drywell high pressure	0.65[4]
Scram discharge volume high water level	1.05[4]
Turbine stop valve closure	0.07
	(0.10)[2]

Function	Total Response Time (Seconds)
Turbine control valve fast closure	0.07
	(0.10)[2]
Reactor vessel high water level	0.30
	(1.05)[3]

[1] Time delay requirements are applicable only above 0.4 percent of rated power.

[2] The total response time indicated is based on testing data as discussed in AECM-82/142. The generator load rejection with bypass, generator load rejection without bypass, and turbine trip without bypass transients were reanalyzed using this response time with no significant effect on the minimum CPRs given in Table 15.0-1 for these events.

[3] The total response time indicated is based on testing data as discussed in AECM-82/142. The feedwater controller failure (maximum demand) transient was reanalyzed using this response time with no significant effect on the minimum CPR given in Table 15.0-1 for this event.

[4] The total response time indicated is a design value and is not used for transient analysis calculations.

In the NSSS vendor's analysis input for the REDY code (Table 15.0-2), the only exposure dependent parameters are the doppler coefficient, the void coefficient, and the scram reactivity. While doppler and void reactivity effect impact transient performance, the scram reactivity dominates the transient response. To provide assurance that the transient evaluations yield the most conservative results, the evaluations are performed at core exposure conditions expected to occur with the worst scram reactivity characteristic. The minimum scram reactivity for projected operation in BWRs occurs at the end of cycle exposure point, when the control rods are completely withdrawn from the core at rated power/flow conditions.

The scram reactivity characteristic varies with exposure, but is most strongly affected by the core power distribution and the associated control rod configuration prior to a scram. The scram reactivity in Figure 15.0-2 presents a conservative lower bound on the minimum scram reactivity for Grand Gulf and also defines the minimum scram characteristic for permitted operation.

The reactivity curve in Figure 15.0-2 is the bounding physics curve multiplied by a conservative factor of 0.80.

The doppler coefficient varies slowly with exposure, and the analysis assumes that the variation is expected to be valued from -.1458 to -.2318 cents/F during rated power operation in cycle 1. There is no defined operation band for this parameter. The void coefficient varies slightly with exposure and is expected to fall in the range of 5.04 to 8.34 cents/% (rated voids) in cycle 1. Except for requiring that the void coefficient be negative, there is no defined operation band for this parameter.

15.0.3.6.3.3 Power/Flow Operating Constraints (Initial cycle)

As described in Section 15.0.3.3.3 for the current fuel cycle, the analysis basis for the initial fuel cycle for most of the transient safety analyses was the thermal power at rated core(100%) corresponding to 105% on the initial licensed nuclear boiler rated steam flow. The extended power/flow domain defined by the NSSS vendor for GGNS was referred to as the maximum extended operating domain (MEOD) and is show if Figure 15D.3-1.

As part of the MEOD analysis by the NSSS vendor (Appendix 15D), operating thermal limits were introduced that are functions of core power and flow. Transient analyses were performed for the limiting events for the initial cycle at different core power and flow conditions to support the MEOD operating limits. These conditions are discussed in the individual subsections of Chapter 15 which deal with the limiting events. The plant was further evaluated for FWHOS and SLO as described in Appendicies 15B and 15C respectively. For the initial cycle, the NSSS vendor evaluated the effects of LOFWH based on the power/flow map shown in Figure 15D.3-2 with a feedwater temperature reduction of 100°F at rated conditions. In addition, the NSSS vendor evaluated the impact on plant transients of FWHOS which would result in decreased feedwater temperature entering the reactor vessel (Appendix 15B).

15.0.3.6.3.4 Results (Initial Cycle)

Analytical evaluations were performed by the NSSS vendor for each event. Critical parameters for each event are shown in Table 15.0-1. The results shown in Table 15.0-1 are based on the NSSS vendor initial core analysis in the operating domain up to 105% initial cycle NBR steam flow/100% core flow. The quantitative information provided in Table 15.0-1 does not reflect operational conditions applicable to MEOD, SLO or reload cores. It provides a comparison of the full spectrum of events that were evaluated by the NSSS vendor for the initial core. MEOD and SLO analyses are discussed in Appendices 15B, 15C, and 15D. From the data in Table 15.0-1 a comparative evaluation of the events for a particular category and parameter can be made. As indicated in subsection 15.0.3, an assessment was made by the reload vendor to identify a subset of limiting events which should be evaluated for each fuel cycle.

Table 15.0-1A provides a summary of applicable accidents for the initial cycle. This table compares the GE calculated amount of failed fuel to that used in worst case Radiological Calculations and provides an assessment of the relative severity of the events.]

Chapter 15 events analyzed by the NSSS vendor do not consider the effect of the Low-Low Set Relief Function. The Low Level Set Relief Function, armed upon relief actuation of any safety/relief valve, will cause a greater magnitude blowdown, in the relief mode, for certain specified safety/relief valves and a subsequent cycling of a single low set valve. The effect of the Low Level Set design on reactor coolant pressure is demonstrated, in Chapter 5, in the MSIV closure with flux scram event. This is considered bounding for all other pressurization events and, therefore, is not simulated in the analysis presented in this chapter.

15.0.3.6.4 Barrier Performance (Initial Cycle)

The barrier performance description for the events for the initial cycle are the same as described for the current cycle. Refer to subsection 15.0.3.4.

15.0.3.6.5 Radiological Consequences (Initial Cycle)

The radiological consequences description for the events for the initial cycle are the same as described for the current cycle. Refer to subsection 15.0.3.5.

15.0.4 Nuclear Safety Operational Analysis (NSOA) Relationship

Appendix 15A is a comprehensive, total plant, system-level, qualitative Failure Modes and Effects Analysis (FMEA), relative to all the Chapter 15 events considered, the protective sequences utilized to accommodate the events and their effects, and the systems involved in the protective actions.

Interdependency of analysis and cross-referral of protective actions is an integral part of this chapter and the appendix. Appendix 15A contains summary tables which classify events by frequency only (i.e., not just within a given category, such as decrease in core coolant temperature).

15.0.5 References

1. United States Nuclear Regulatory Commission Regulatory Guide 1.70 Revision 2 (Preliminary), September 1975, "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants, Light Water Reactor Edition."

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

3. "General Electric Standard Application for Reactor Fuel (GESTAR II),"(NEDE-24011-P-A).

4. General Electric Company Report, "GGNS Single Loop Operating Analysis," February 1986.

5. ECH-NE-20-00009, Rev. 0, Supplemental Reload Licensing Report for Grand Gulf-1 Reload 22 Cycle 23, February 2020.

6. ECH-NE-10-00021, Rev. 5, "GNF2 Fuel Design Cycle-Independent Analyses for Grand Gulf Nuclear Station," Mar. 2020.

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

8. Odar, F., Safety Evaluation for General Electric Topical Report: "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154, NEDE-24154P, Volumes 1, 2, and 3, 1980.

9. General Electric Company Report "GGNS Feedwater Heater(s) out of Service," March 1986.

10. Deleted

11. Deleted

12. Deleted

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14. Deleted

15. Deleted

16. Deleted

17. Deleted

18. Deleted

19. Deleted

20. Deleted

21. General Electric Company Report, "Maximum Extended Operating Domain Analysis", March 1986.

22. US Nuclear Regulatory Commission letter dated August 31, 2015, "Grand Gulf Nuclear Station, Unit 1 - Issuance of Amendment Regarding Maximum Extended Load Line Limit Analysis Plus, (GNRI-2015/00114).

23. NEDC-33292P-A, "GEXL 17 Correlation for GNF2 Fuel, (latest approved revision).

24. NEDC-33880P, GEXL21 Correlation for GNF3 Fuel, (latest approved revision).

25. ECH-NE-20-00006, Rev. 0, GNF3 Fuel Design Cycle-Independent Analyses for Grand Gulf Nuclear Station, February 2020.

TABLE 15.0-1: [HISTORICAL INFORMATION] RELATIVE SEVERITY OF TRANSIENT EVENTS BASED ON GGNS INITIAL CORE++

										Duration of Blowdown	
Sub-section I.D.	Figure I.D.	Description	Maximum Neutron Flux % NBR	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam Line Pressure psig	Maximum Core Average Surface Heat Flux 5 of Initial	Minimum CPR+++ -	Frequency Category*	No. of Valves 1st Blow-down	Duration of Blow-down sec
15.1		DECREASE IN CORE COOLANT TEMPERATURE									
15.1.1	15.1-2	Loss of Feedwater Heater, Manual Flow Control	122	1060	1072	1042	114	1.06	a	0	0
15.1.2	15.1-3	Feedwater Cntl Failure,**,+ Max Demand, with Turbine Bypass	111	1160	1188	1166	105	1.09	a	10	6
15.1.3	15.1-4	Pressure Controller Fail - Open	104	1127	1130	1127	100	>1.13	a	11	3
15.1.4	15.1-5 15.1-6	Inadvertent Opening of Safety or Relief Valve				See Text					
15.1.6		RHR Shutdown Cooling Malfunction Decreasing Temp				See Text					
15.2		INCREASE IN REACTOR PRESSURE				See Text					
15.2.1	15.2-1	Pressure Controller**,+ Downscale Failure	150	1194	1231	1192	102	1.09	a	20	9
15.2.2	15.2-2	Generator Load Rejection,** Bypass-On	105	1165	1193	1163	100	>1.13	a	20	5
15.2.2	15.2-3	Generator Load Rejection,**,+ Bypass-Off	149	1208	1240	1213	101	1.13	a	20	6
15.2.3	15.2-4	Turbine Trip, Bypass-On	111	1154	1161	1148	100	>1.13	a	20	5
15.2.3	15.2-5	Turbine Trip, Bypass-Off**	105	1202	1233	1207	100	>1.13	a	20	6
15.2.4	15.2-6	Inadvertent MSIV Closure**	105	1180	1213	1179	100	>1.13	a	20	5.6
15.2.5	15.2-7	Loss of Condenser Vacuum	104	1190	1217	1194	100	>1.13	a	20	10

TABLE 15.0-1: [HISTORICAL INFORMATION] RELATIVE SEVERITY OF TRANSIENT EVENTS BASED ON GGNS INITIAL CORE++ (Continued)

Sub-section I.D.	Figure I.D.	Description	Maximum Neutron Flux % NBR	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam Line Pressure psig	Maximum Core Average Surface Heat Flux 5 of Initial	Minimum CPR+++ -	Frequency Category*	Duration of Blowdown: No. of Valves 1st Blow-down	Duration of Blowdown: Duration of Blow-down sec
15.2.6	15.2-8	Loss of Auxiliary Power Transformer	104	1134	1147	1131	100	>1.13	a	11	4
15.2.6	15.2-9	Loss of All Grid Connections	121	1156	1163	1149	100	>1.13	a	20	5
15.2.7	15.2-10	Loss of All Feedwater Flow	104	1045	1056	1029	100	>1.13	a	0	5
15.2.8	-	Feedwater Piping Break	See Table 15.0-1A, event 15.6.6								
15.2.9	15.2-13	Failure of RHR Shutdown Cooling				See Text					
15.2.10	-	Loss of Instrument Air				See Text					
15.3		DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE									
15.3.1	15.3-1	Trip of One Recirculation Pump Motor	104	1045	1056	1029	100	>1.13	a	0	0
15.3.1	15.3-2	Trip of Both Recirculation Pump Motors	104	1167	1171	1162	100	>1.13	a	20	7
15.3.2	15.3-3	Fast Closure of One Main Recirc. Valve	104	1045	1056	1029	100	>1.13	a	0	0
15.3.2	15.3-4	Fast Closure of Two Main Recirc. Valves	104	1167	1170	1161	100	>1.13	a	20	7
15.3.3	15.3-5	Seizure of One Recirculation Pump	104	1149	1152	1143	100	>1.13	c	20	8
15.3.4		Recirc. Pump Shaft Break	See Subsection 15.3.3								
15.4		REACTIVITY AND POWER DISTRIBUTION ANOMALIES									

TABLE 15.0-1: [HISTORICAL INFORMATION] RELATIVE SEVERITY OF TRANSIENT EVENTS BASED ON GGNS INITIAL CORE++ (Continued)

										Duration of Blowdown	
Sub-section I.D.	Figure I.D.	Description	Maximum Neutron Flux % NBR	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam Line Pressure psig	Maximum Core Average Surface Heat Flux 5 of Initial	Minimum CPR+++ -	Frequency Category*	No. of Valves 1st Blow-down	Duration of Blow-down sec
15.4.1.1		RWE - Refueling	See Text						b		
15.4.1.2		RWE - Startup	See Text						b		
15.4.2		RWE - At Power	See Text						a		
15.4.3		Control Rod Misoperation	See Subsections 15.4.1 and 15.4.2								
15.4.4	15.4-3	Abnormal Startup of Idle Recirculation Loop	86	985	988	978	135	>1.13	a	0	0
15.4.5	15.4-4	Fast Opening of One Main Recirc. Valve	316	976	994	971	135	>1.13	a	0	0
15.4.5	15.4-5	Fast Opening of Both Main Recirc Valves	256	974	994	969	133	>1.13	a	0	0
15.4.7		Misplaced Bundle Accident	See Text					1.08	b		
15.5		INCREASE IN REACTOR COOLANT INVENTORY									
15.5.1	15.5-1	Inadvertent HPCS Pump Start	104	1045	1056	1029	100	>1.13	a	0	0
15.5.3		BWR Transients	See appropriate Events in Sections 15.1 and 15.2								

* Frequency definition is discussed in subsection 15.0.3.1.
**Transients simulated using ODYN Code (Ref. 2, Section 15.1.8)
a Moderate frequency
b Infrequent
c Unexpected
\+ Results presented here were generated from analyses using updated turbine inlet pressure and steamline volume as reflected in Table 15.0-3.
++ Results of Feedwater Heater(s) Out of Service (FWHOS), Single Loop Operation (SLO), and Maximum Extended Operating Domain (MEOD) analyses are documented in Appendix 15B, 15C, and 15D, respectively.
+++Based on minimum Initial CPR of 1.18.

TABLE 15.0-1A: [HISTORICAL INFORMATION] SUMMARY INITIAL CYCLE OF ACCIDENTS

Paragraph I.D.	Title	Failed Fuel: GE Calculated Value	Failed Fuel: NRC Worst Case Assumption
15.3.3	Seizure of one recirculation pump	None	None
15.3.4	Recirculation pump shaft break	None	None
15.4.9	Rod drop accident	<770	770
15.6.2	Instrument line break	None	None
15.6.4	Steam system pipe break outside containment	None	None
15.6.5	LOCA within RCPB	None	100%
15.6.6	Feedwater line break	None	None
15.7.1	Main condenser gas treatment system failure	N/A	N/A
15.7.3	Liquid radwaste tank failure	N/A	N/A
15.7.4	Fuel handling accident outside containment	101	N/A
15.7.5	Cask drop accident	None	N/A
15.7.6	Fuel handling accident inside containment	130	N/A
15.8	ATWS	None*	N/A

*Special event still under negotiation with the NRC.

TABLE 15.0-1B: DELETED

TABLE 15.0-2: INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS USED BY NSSS VENDOR IN REDY CODE FOR INITIAL CORE (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR SOME OF THE CHAPTER 15 EVENTS)(REF. 7)

1. Thermal power level, MWt	
Warranted value	3833
Analysis value	3993
2. Steam flow, lbs per hr	
Warranted value (NBR)	16.488×10^6
Analysis value	17.312×10^6
3. Core flow, lbs per hr	113.5×10^6
4. Feedwater flow rate, lb per sec	
Warranted value (NBR)	4618
Analysis value	4809
5. Feedwater temperature, F	425
6. Vessel dome pressure, psig	1045
7. Vessel core pressure, psig	1056
8. Turbine bypass capacity, % NBR	35
9. Core coolant inlet enthalpy, Btu per lb	530.2
10. Turbine inlet pressure, psig	960
11. Fuel lattice	8 x 8 R***
12. Core average gap conductance, Btu/hr-ft^2-F	557
13. Core leakage flow, %	10.65
14. Required MCPR operating limit	
First core**	1.18
15. MCPR safety limit for incidents	

TABLE 15.0-2: INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS USED BY NSSS VENDOR IN REDY CODE FOR INITIAL CORE (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR SOME OF THE CHAPTER 15 EVENTS)(REF. 7) (CONTINUED)

of moderate frequency	
First core	1.06
16. Doppler coefficient (-) ¢/F	
Analysis data	0.132
17. Void coefficient (-) ¢/% rated voids	
Analysis data for power	
Increase events	14.0
Analysis data for power	
Decrease events	4.0
18. Core average rated void	
Fraction, %	41.9
19. Scram reactivity, $ k	
Analysis data	Figure 15.0-2
20. Control rod drive speed,	
position versus time	Figure 15.0-3
21. Jet pump ratio, M	2.32
22. Safety/relief valve capacity, % NBR	
@ 1125 psig	100.6
Manufacturer	Dikker
Quantity installed	20
23. Relief function delay, seconds	0.4
24. Relief function response time	
constant, seconds	0.1
25. Set points for safety/relief valves	
Safety function, psig	1175, 1195, 1215
Relief function, psig	1125, 1135, 1145, 1155
26. Number of valve groupings simulated	

TABLE 15.0-2: INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS USED BY NSSS VENDOR IN REDY CODE FOR INITIAL CORE (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR SOME OF THE CHAPTER 15 EVENTS)(REF. 7) (CONTINUED)

Safety function, No.	3
Relief function, No.	4
27. High flux trip, % NBR	
Analysis set point (122 x 1.042), % NBR	127.2
28. High-pressure scram set point, psig	1,095
29. Vessel level trips, feet above separator skirt bottom	
Level 8 - (L8), feet	5.88
Level 4 - (L4), feet	4.03
Level 3 - (L3), feet	2.16
Level 2 - (L2), feet	(-)2.16
30. APRM thermal trip	
Set point, % NBR	118.8
31. Recirculation pump trip delay, Seconds	0.14
32. Recirculation pump trip inertia time constant for analysis, sec*	5

*The inertia time constant is defined by the expression:

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

where t = inertia time constant (sec)

J_o= pump motor inertia (lb-ft^2)

n = rated pump speed (rps)

g = gravitational constant (ft/sec^2)

T_o = pump shaft torque (ft-lb)

**The operating limit MCPR used for the initial core is 1.18.

***Includes two water rods

TABLE 15.0-3: INPUT PARAMETERS AND INTIAL CONDITIONS FOR TRANSIENTS USED BY NSSS VENDOR IN ODYN CODE FOR INTITAL CORE (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR SOME OF THE CHAPTER 15 EVENTS)(REF. 8)****

1.	Thermal power level, MWt	
	Warranted value	3833
	Analysis value	3993
2.	Steam flow, lbs per hr	
	Warranted value (NBR)	16.488×10^6
	Analysis value	17.312×10^6
3.	Core flow, lbs per hr	113.5×10^6
4.	Feedwater flow rate, lb per sec	
	Warranted value (NBR)	4618
	Analysis value	4809
5.	Feedwater temperature, F	425
6.	Vessel dome pressure, psig	1045
7.	Vessel core pressure, psig	1056
8.	Turbine bypass capacity, % NBR	35
9.	Core coolant inlet enthalpy, Btu per lb	530.2
10.	Turbine inlet pressure, psig	960 (1000)+
11.	Fuel lattice	P 8 x 8 R++
12.	Core average gap conductance, $Btu/hr\text{-}ft^2\text{-}F$	681
13.	Core leakage flow, %	10.65
14.	Required MCPR operating limit	
	First core***	1.18

TABLE 15.0-3: INPUT PARAMETERS AND INTIAL CONDITIONS FOR TRANSIENTS USED BY NSSS VENDOR IN ODYN CODE FOR INTITAL CORE (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR SOME OF THE CHAPTER 15 EVENTS)((REF. 8)**(Continued)**

15.	MCPR safety limit for incidents of moderate frequency	
	First core	1.06
16.	Doppler coefficient (-) ¢/F	
	Analysis data	**
17.	Void coefficient (-) ¢/% rated voids	
	Analysis data for power	
	Increase events	**
	Analysis data for power	
	Decrease events	**
18.	Core average rated void	
	Fraction, %	**
19.	Scram reactivity, $ k	
	Analysis data	Figure 15.0-2**
20.	Control rod drive speed,	
	position versus time	Figure 15.0-4
21.	Jet pump ratio, M	2.32
22.	Safety/relief valve capacity, % NBR	
	@ 1145 psig	102.4
	Manufacturer	Dikker
	Quantity installed	20
23.	Relief function delay, seconds	0.4
24.	Relief function response time constant, seconds	0.1
25.	Set points for safety/relief valves	
	Safety function, psig	1175, 1185, 1195, 1205, 1215
	Relief function, psig	1145, 1155, 1165, 1175
26.	Number of valve groupings simulated	
	Safety function, No.	5

TABLE 15.0-3: INPUT PARAMETERS AND INTIAL CONDITIONS FOR TRANSIENTS USED BY NSSS VENDOR IN ODYN CODE FOR INTITAL CORE (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR SOME OF THE CHAPTER 15 EVENTS)((REF. 8)**(Continued)**

	Relief function, No.	4
27.	High flux trip, % NBR Analysis set point (122 x 1.042), % NBR	127.2
28.	High-pressure scram set point, psig	1,095
29.	Vessel level trips, feet above separator skirt bottom	
	Level 8 - (L8), feet	5.88
	Level 4 - (L4), feet	4.03
	Level 3 - (L3), feet	2.16
	Level 2 - (L2), feet(-)	(-)2.182
30.	APRM thermal trip Set point, % NBR	118.8
31.	Recirculation pump trip delay, Seconds	0.190
32.	Recirculation pump trip inertia time constant for analysis, sec*	5
33.	Recirculation pump, high pressure trip set point, psig (nominal)	1135
	Time delay, seconds	0.3
34.	Total steamline volume, ft^3	4358 (6022)+

*The inertia time constant is defined by the expression:

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

where t = inertia time constant (sec)

J_o= pump motor inertia ($lb\text{-}ft^2$)

n= rated pump speed (rps)

g= gravitational constant (ft/sec^2)

T_o= pump shaft torque (ft-lb)

**ODYN values are calculated within the code for the end of cycle 1 condition.

***The operating limit MCPR used for the initial core is 1.18.

****Input parameter(s) for the Feedwater Heater(s) Out of Service (FWHOS), Single Loop Operation (SLO) and Maximum Extended Operating Domain (MEOD) analyses are provided in Appendices 15B, 15C, and 15D, respectively.

+ The value listed in parentheses was used in the analyses of feedwater controller failure, pressure controller downscale failure, and generator load rejection bypass failure events. Results of these analyses are listed in Table 15.0-1.

++ Includes two water rods

TABLE 15.0-4: INPUT PARAMETERS AND INITIAL CONDITIONS USED FOR RELOAD TRANSIENT ANALYSES

Parameter or Condition	Value	
Reactor Thermal Power Level	4408 MWt	
Reactor Pressure - Steam Dome	1040 psia	
Reactor Pressure - Top of Core (100% flow)	1050 psia	
Turbine Pressure	980 psia	
Core Flow Rate	112.5 Mlb/hr	
Feedwater Temperature	420°F	
Feedwater Flow Rate	18.935 Mlb/hr	
Steam Flow Rate	18.968 Mlb/hr	
Control Rod Drive - Rod Travel	144 in	
	Reactor Steam Dome Pressure	
	965 psia	1065 psia
Control Rod Drive - Scram Distance vs Time	10% @ 0.30 sec	0.31 sec
	40% @ 0.78 sec	0.84 sec
	73% @ 1.40 sec	1.53 sec
Recirculation Pump - Flow Rate (rated)	44,600 gpm/pump	
Recirculation Pump - Rotor Speed (rated)	1785 rpm	
Recirculation Pump - Head (rated)	765 ft	
Recirculation Pump - Direction of Rotation	Reverse Rotation Allowed	

TABLE 15.0-4: INPUT PARAMETERS AND INITIAL CONDITIONS USED FOR RELOAD TRANSIENT ANALYSES (CONTINUED)

Parameter or Condition	Value
MSLIV - Closure Time, Lower Limit	3 sec
MSLIV - Closure Time, Upper Limit	5 sec
MSLIV - Delay Time	1 sec
Turbine Control Valve Stroke Time	0.15 sec
Feed Flow Sensor Time Constant	0.25 sec
Steam Flow Sensor Time Constant	1.0 sec
Feedwater Master Controller - Proportional Gain	4.17%/in
Feedwater Master Controller - Deadband	0.0
Feedwater Master Controller - Compensation, Lead	0.7 sec
Feedwater Master Controller - Compensation, Lag	7.0 sec
Feedwater System 100% Mismatch - Gain	48.0 in/100%
Feedwater System 100% Mismatch - Steam Flow Equiv.	18.968 Mlb/hr
Feedwater System 100% Mismatch - Max Demand Flow Rate	130+0.2(1080-p) where P = dome pressure (psia)
Feedwater System 100% Mismatch - Feedwater Turbine Lag	1.0 sec

TABLE 15.0-4: INPUT PARAMETERS AND INITIAL CONDITIONS USED FOR RELOAD TRANSIENT ANALYSES (CONTINUED)

Parameter or Condition	Value
Feedwater System 100% Mismatch - Response Limit, Decrease	25%/sec
Feedwater System 100% Mismatch - Response Limit, Increase	Assumed Instantaneous for Pressure Control Downscale Failure Event
Pressure Regulator - Lead	3.0 sec
Pressure Regulator - Lag 1	7.0 sec
Pressure Regulator - Lag 2	0.445 sec
Pressure Regulator - Gain	3.33%/psi
Feedwater Control Mode	3 element
Relief Valves - Number and Opening Setpoints	6@ 1168 psia 6 available
Relief Valves - Delay Time	0.4 sec
Relief Valves - Stroke Time	0.15 sec
Relief Valves - Flow Capacity	925,000 lb/hr at 103% of 1205 psig
Safety Valves - Number and Opening Setpoints	6 @ 1252 psia 3 @ 1242 psia 9 available
Safety Valves - Delay Time	0.0 sec
Safety Valves - Stroke Time	0.3 sec
Safety Valves - Flow Capacity	925,000 lb/hr

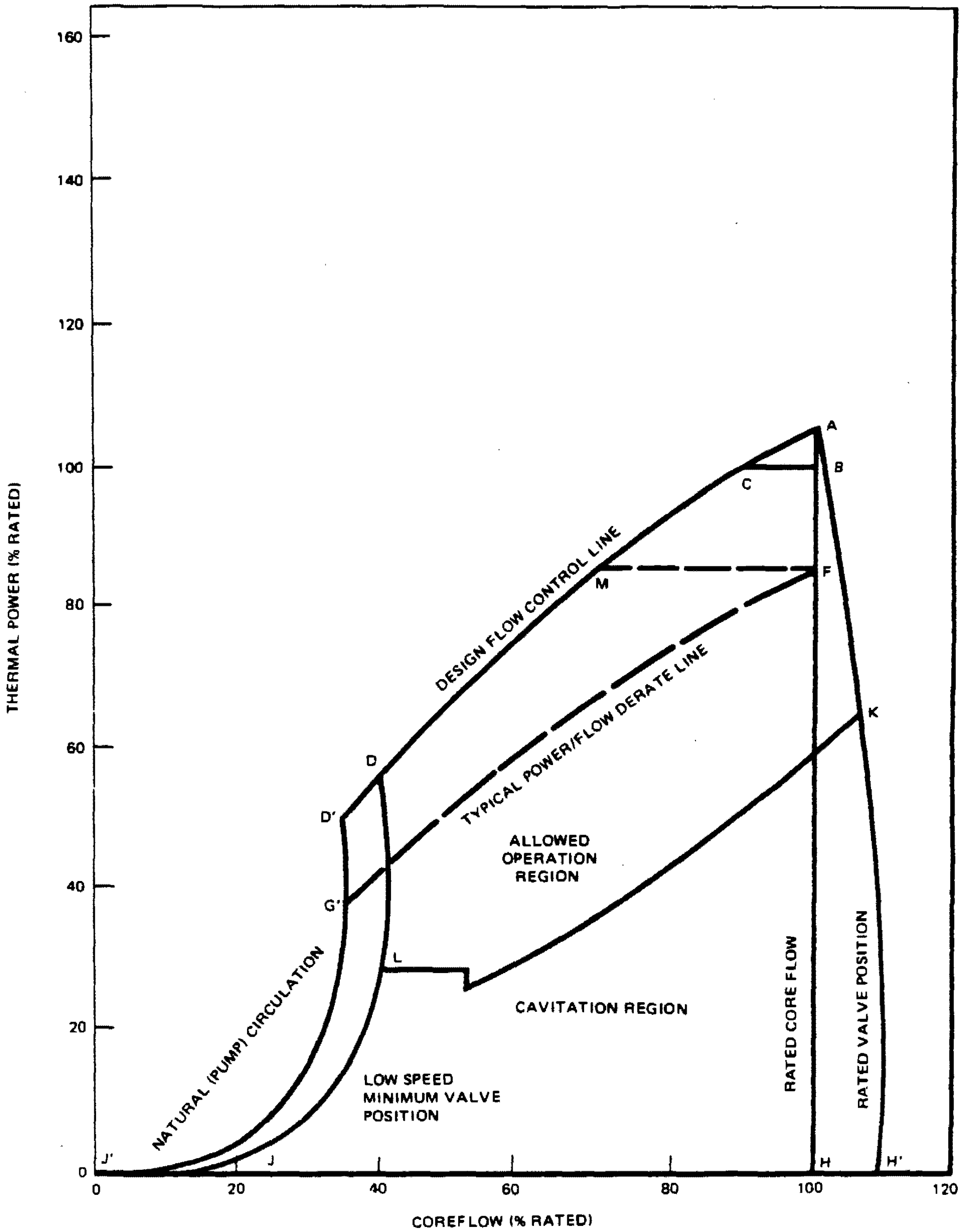
TABLE 15.0-4: INPUT PARAMETERS AND INITIAL CONDITIONS USED FOR RELOAD TRANSIENT ANALYSES (CONTINUED)

Parameter or Condition	Value
	at 103% of 1205 psig
Reactor Protection System - High Flux	122%
Reactor Protection System - Thermal Power	113%
Reactor Protection System - Flow Biased Flux Scram	N/A
Reactor Protection System - Turbine Stop Valve Position	10% Closed
Reactor Protection System - MSLIV Position	10% Closed
Reactor Protection System - High Vessel Pressure	1095 psig
Reactor Protection System - Low Water Level (L3)	543.2 in
Reactor Protection System - High Water Level (L8)	587.7 in
Reactor Protection System - Delay Time, MSLIV Position	0.06 sec
Reactor Protection Sytem - Delay Time, High Vessel Pressure	0.35 sec
Reactor Protection System - Delay Time, Low Water Level	1.05 sec

TABLE 15.0-4: INPUT PARAMETERS AND INITIAL CONDITIONS USED FOR RELOAD TRANSIENT ANALYSES (CONTINUED)

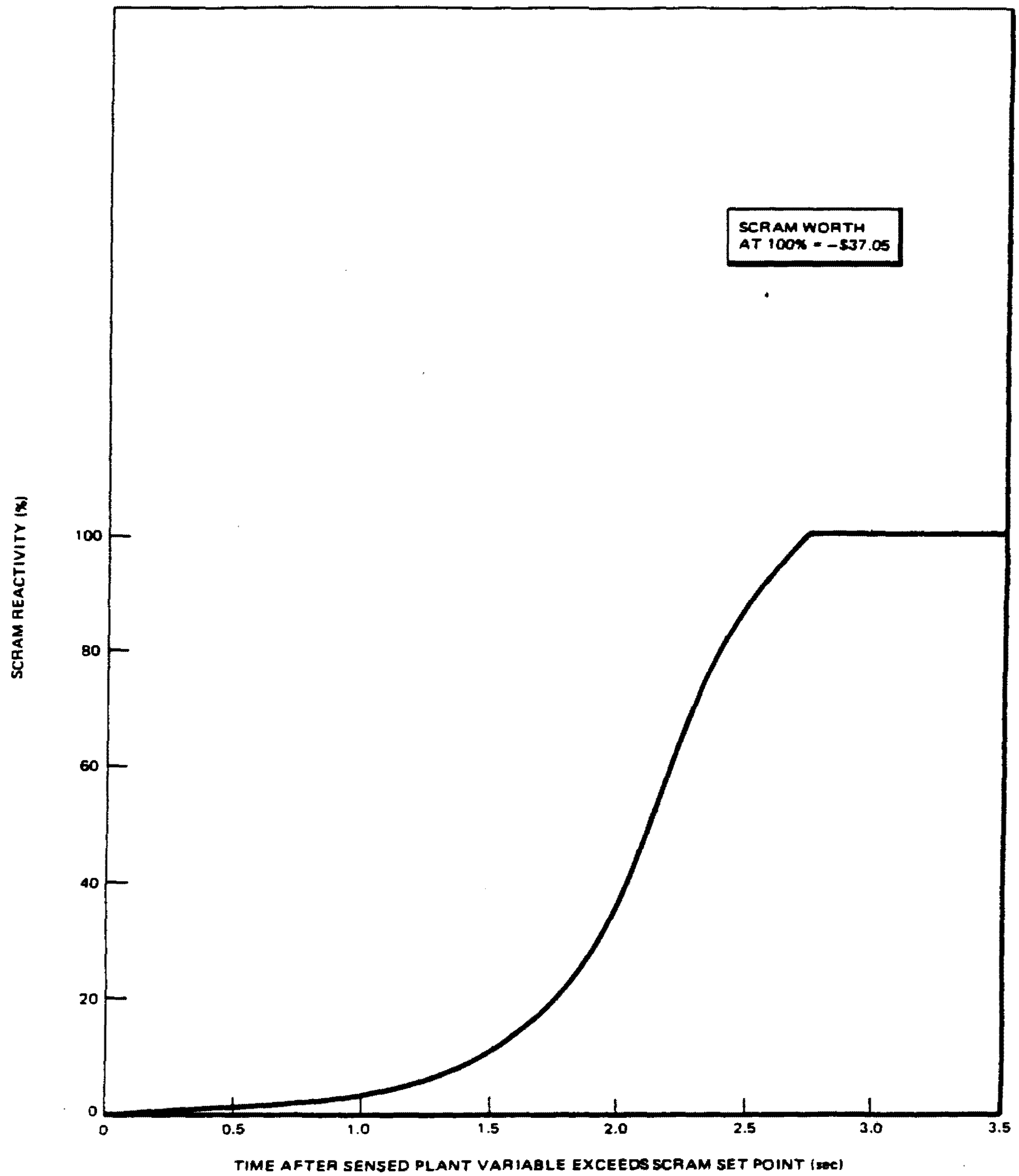
Parameter or Condition	Value
Reactor Protection System - Delay Time, High Water Level	1.05 sec
MSLIV Closure - Low, Low, Low Level	378.3 in
MSLIV Closure - Loss of Power	LOOP
Recirculation Pump Trip - High Vessel Pressure	1150 psig
Recirculation Pump Downshift - Low Water Level	543.2 in
Recirculation Pump Trip - Low Low Water Level	487.0 in
Recirculation Pump Trip - Delay Time	0.20 sec*
Recirculation Pump Trip - Drive Motor	LOOP

* Value includes a 10 msec delay to account for the turbine stop valve position trip (90% open).



[HISTORICAL INFORMATION]

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	TYPICAL POWER/FLOW MAP (Initial Cycle) FIGURE 15.0-1



[HISTORICAL INFORMATION]

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	SCRAM REACTIVITY CHARACTERISTICS (Initial Cycle) FIGURE 15.0-2

15.1 DECREASE IN REACTOR COOLANT TEMPERATURE

15.1.1 Loss of Feedwater Heating

The reload fuel vendor has determined that the Loss of Feedwater Heating (LOFWH) event is a limiting event which requires analysis for the current fuel cycle. This subsection describes the analysis performed by the reload fuel vendor for the current fuel cycle. For a description of the initial fuel cycle analysis of this event by the NSSS vendor, refer to subsection 15.1.1.6. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload fuel vendor for the current cycle, refer to Section 15.0.

15.1.1.1 Identification of Causes and Frequency Classification

15.1.1.1.1 Identification of Causes

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

a. The steam extraction line to the heater is closed.

b. Feedwater is bypassed around the heater.

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

15.1.1.1.2 Frequency Classification

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

This event is analyzed under worst case conditions of a 100˚F loss and full power. A reduction of feedwater temperature of 100˚F at high power has never been reported although smaller decreases have occurred. The probability of occurrence of this event is, therefore, regarded as small.

15.1.1.2 Sequence of Events and Systems Operation

The LOFWH transient is initiated by introducing feedwater whose temperature is lower by 100˚F into the reactor vessel. This results in an increase in core inlet subcooling. This increase in subcooling causes a collapse of voids (steam bubbles) and increases core power due to the associated reactivity insertion. This increase in core power results in a shift of the axial power distribution toward the bottom of the core. The power shift causes void (bubble) formation to increase toward the core bottom. The void formation moderates the core power increase and new higher power level is achieved in several minutes.

The analysis makes use of the BWR Simulator Code, as described in Reference 6. This approach does not account for a reactor scram to mitigate the event even though the flux levels due to the transient may be high enough to initiate a high APRM thermal power trip. The action of other engineered safeguards systems is also not credited.

15.1.1.2.1 Deleted

15.1.1.2.1.1 Deleted

15.1.1.2.2 Deleted

15.1.1.3 Core and System Performance

The response to the LOFWH transient is relatively slow, with the reactor core remaining in a nearly steady state condition throughout the event. The parameters which describe the transient behave smoothly with no sudden increases or decreases. These trends have been verified in start-up tests which show that the time to attain 95% of the parameter changes in response to the feedwater temperature change is greater than 100 seconds. The reload fuel vendor has performed this evaluation with the BWR Simulator Code, (Ref. 6) and the results are available in the SRLR (Ref. 8). The results show that the equilibrium power level is not significantly different from the maximum power level when the transient period exceeds 100 seconds. Start-up tests for other plants confirmed that the end-of-event conditions could be accurately predicted based on a simple heat balance.

Calculations for the LOFWH event make use of the results of the analysis performed by the reload fuel vendor, assuming the following:

a. The reactor is in steady state conditions before and after the event.

b. The xenon distribution does not change during the event.

c. The total core flow during the event is constant. Flow is allowed to be redistributed in order to maintain equal differential pressure across each fuel assembly in the core.

d. Although the flux levels due to the LOFWH event may be sufficient to cause a high APRM thermal power trip, no account is taken of a reactor scram in the evaluation.

Analysis of the LOFWH event reflects reactor operation over the expanded operating domain power flow map and conditions anticipated during actual Grand Gulf operation.

15.1.1.3.1 Deleted

15.1.1.3.2 Deleted

15.1.1.3.3 Results

The LOFWH event was analyzed at 100% rated power using the BWR simulator code Ref. 6 which demonstrates that the MCPR after a LOFWH event can be directly correlated to the MCPR prior to the LOFWH event by the safety limit MCPR and the following plant parameters: core power, rated feedwater flow and the change in feedwater temperature. The analysis assumed a conservative reduction of 100˚F in the feedwater temperature. Analyses were performed for several cycle exposures to ensure that appropriate limits are set. The analysis results are provided in Reference 8 for the current reload fuel cycle.

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The feedwater heating system is designed such that any single equipment failure or operator error would not result in a temperature drop of more than 100˚F. The loss of feedwater heating test performed during the Grand Gulf Unit 1 startup test program resulted in a 82˚F loss. The drop of 100˚F was chosen to be conservative.

15.1.1.3.4 Considerations of Uncertainties

Important factors (such as reactivity coefficient, scram characteristics, magnitude of the feedwater temperature change) are assumed to be at the worst configuration so that any deviations seen in the actual plant operation reduce the severity of the event.

15.1.1.4 Barrier Performance

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

15.1.1.5 Radiological Consequences

Radiological consequences were not evaluated since no fuel failures are associated with the event and no radioactivity is discharged to the suppression pool.

15.1.1.6 Initial Cycle

The reload fuel vendor has determined that the Loss of Feedwater Heating (LOFWH) event is a limiting event which requires analysis for the current fuel cycle. This subsection describes the analysis performed by the NSSS vendor for the initial fuel cycle. For a description of the current fuel cycle analysis of this event, refer to subsection 15.1.1. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload fuel vendor for the current cycle, refer to Section 15.0.

15.1.1.6.1 Identification of Causes and Frequency Classification

The potential causes of this event are the same as for the current cycle. The probability of occurrence of this event is also the same as that for the current cycle. Refer to subsection 15.1.1.1.

15.1.1.6.2 Sequence of Events and Systems Operation

15.1.1.6.2.1 Sequence of Events

[HISTORICAL INFORMATION] [Tables 15.1-1 and 15.1-2 list the sequence of events for this transient, and its effect on various parameters is shown in Figures 15.1-1 and 15.1-2.]

15.1.1.6.2.2 Systems Operation

[HISTORICAL INFORMATION] [In establishing the expected sequence of events and simulating the plant performance for the initial cycle, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems.

The thermal power monitor is the primary protection system trip in mitigating the consequences of this event for the initial cycle.

If there was no high thermal power trip scram design available in the Grand Gulf plant design, reactor scram during the loss of feedwater heating transient would occur when the neutron flux exceeds the high APRM flux scram set point. Usually, the high APRM flux scram set point is higher than the high thermal power scram set point by approximately 6 to 8 percent. therefore, the loss of feedwater heating transient could be more severe without the high thermal power trip scram design.

The high thermal power scram set point for the initial cycle for plant operation up to 100 percent NBR power is shown in Figure 15.1-7.

Since initial cycle transients in Chapter 15 are analyzed at 104.2 percent initial licensed NBR power, the scram set points are increased by the same factor. Therefore, the high thermal power scram set point maximum limit is 114% x 1.042 = 118.8% NBR as shown in Table 15.0-2.

Required operation of engineered safeguard features is not expected for either of the LOFWH transients.]

15.1.1.6.2.3 The Effect of Single Failures and Operator Errors

[HISTORICAL INFORMATION] [These two events generally lead to an increase in reactor power level. The thermal power monitor mentioned in subsection 15.1.1.2.2 is the mitigating system and is designed to be single failure proof. Therefore, single failures are not expected to result in a more severe event than analyzed. See Appendix 15A for a detailed discussion of this subject.]

15.1.1.6.3 Core and System Performance

15.1.1.6.3.1 Mathematical Model

[HISTORICAL INFORMATION] [The predicted dynamic behavior for the initial cycle has been determined using a computer simulated, analytical model of a generic direct-cycle BWR. This model is described in detail in Reference 1. This computer model has been improved and verified through extensive comparison of its predicted results with actual BWR test data.

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

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

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

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

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

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

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

15.1.1.6.3.2 Input Parameters and Initial Conditions

[HISTORICAL INFORMATION] [These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

The plant is assumed to be operating at 105 percent of initial licensed nuclear boiler (NB) rated power and at thermally limited conditions. Both automatic and manual modes of flow control are considered.

The same void reactivity coefficient conservatism used for pressurization transients is applied since a more negative value conservatively increases the severity of the power increase. The values for both the feedwater heater time constant and the feedwater time volume between the heaters and the spargers are adjusted to reduce the time delays since they are not critical to the calculation of this transient. The transient is simulated by programming a change in feedwater enthalpy corresponding to a 100˚F loss in feedwater heating.]

15.1.1.6.3.3 Results

[HISTORICAL INFORMATION] [In the automatic flux/flow control mode, the recirculation flow control system responds to the power increase by reducing core flow so that steam flow from the reactor vessel to the turbine remains essentially constant. In order to maintain the initial steam flow with the reduced inlet temperature, reactor thermal power increases above the initial value and settles at about 110 percent NBR (106 percent of initial power), below the flow-referenced APRM thermal power scram setting and core flow is reduced to approximately 88 percent of rated flow. The MCPR reached in the automatic control mode is greater than for the more limiting manual flow control mode.

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

In manual mode, no compensation is provided by core flow and thus the power increase is greater than in the automatic mode. A scram on high APRM thermal power occurs. Vessel steam flow increases and the initial system pressure increase is slightly larger. Peak heat flux is 114 percent of its initial value and peak fuel center temperature increases 232˚F. The increased core inlet subcooling aids core thermal margins and minimum MCPR is 1.06. Therefore, the design basis is satisfied. The transient responses of the key plant variables for this mode of operation are shown in Figure 15.1-2.

After the reactor scram, water level drops to the low level trip point (L2). This initiates recirculation pump trip as shown in Table 15.1-2.]

15.1.1.6.4 Barrier Performance

[HISTORICAL INFORMATION] [As in the current cycle, the fuel, pressure vessel, and containment barrier design criteria are not exceeded, so these barriers would maintain their integrity as designed.]

15.1.1.6.5 Radiological Consequences

[HISTORICAL INFORMATION] [No fuel failures were associated with this event for the initial cycle, and no radioactivity would be released to the suppression pool.]

15.1.2 Feedwater Controller Failure - Maximum Demand

The reload fuel vendor has determined that the failure of the feedwater controller to maximum demand (FWCF) event is a limiting event which requires analysis for each fuel loading cycle. This subsection describes the analysis performed by the reload fuel vendor for the current fuel cycle. For a description of the initial fuel cycle analysis of this event, refer to subsection 15.1.2.6. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload fuel vendor for the current cycle, refer to Section 15.0.

15.1.2.1 Identification of Causes and Frequency Classification

15.1.2.1.1 Identification of Causes

This event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing the feedwater flow. The most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

15.1.2.1.2 Frequency Classification

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

15.1.2.2 Sequence of Events and Systems Operation

15.1.2.2.1 Sequence of Events

The reload fuel vendor has determined that the FWCF event is the most limiting of the vessel inventory increase transients. Table 15.1-3 lists the sequence of events for this transient. Failure of the feedwater control system to maximum demand would result in an increase in the coolant level in the reactor vessel. Increased feedwater flow results in lower temperatures at the core inlet, which in turn cause an increase in core power level. If the feedwater flow stabilizes at the increased value, the core power will stabilize at a new, higher value. If the flow increase continues, the water level in the downcomer will eventually reach the high level setpoint (L9), at which time the turbine stop and control valves are closed to avoid damage to the turbine from excessive liquid inventory in the steamlines. The high water level trip (L8) initiates a reactor scram, and subsequent turbine trip leads to recirculation pump high to low speed transfer. The core power excursion is terminated by the same mechanisms that end thegenerator load reject W/O bypass transient.

Reference 8 contains the responses of various reactor and plant parameters to the subject transient at 100% power and 105% flow.

15.1.2.2.1.1 Deleted

15.1.2.2.2 Systems Operation

In order to properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. No credit is taken for turbine bypass valves operation. Important system operational actions for this event are tripping of the main turbine and feedwater pumps, recirculation pump trip, scram, and low water level initiation of the reactor core isolation cooling system and the high pressure core spray system to maintain long term water level control following tripping of feedwater pumps.

15.1.2.2.3 The Effect of Single Failures and Operator Errors

In Table 15.1-3 the first sensed event to initiate corrective action to the transient is the vessel high water level (L8) scram. Scram trip signals from Level 8 are designed such that a single failure will neither initiate nor impede a reactor scram trip initiation. Therefore, single failures are not expected to result in a more severe event than analyzed. See Appendix 15A for a detailed discussion of this subject.

15.1.2.3 Core and System Performance

15.1.2.3.1 Mathematical Model

The predicted dynamic behavior has been determined using a computer simulated, analytical model. The computer model is described in detail in Reference 7. Additional, and updated, information on the modeling employed is contained in Reference 6. This computer model has been verified through comparison of its predicted results with actual BWR test data.

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

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

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

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

d. The core average axial water density and pressure distribution is calculated using a single channel to represent the heated active flow and a single channel to represent the bypass flow. A model, representing liquid and vapor mass and energy conservation and mixture momentum conservation, is used to describe the thermal-hydraulic behavior. Changes in the flow split between the bypass and active channel flow are accounted for during transient events.

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

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

15.1.2.3.2 Input Parameters and Initial Conditions

The analyses have been performed, unless otherwise noted, with the plant conditions identified in Section 15.0 for the reload transients (Table 15.0-4). The transient was simulated by programming an upper limit failure in the feedwater control system. The event was analyzed with and without Feedwater Heaters Out of Service (FWHOOS) at the End of Cycle All Control Rods Out condition for a large number of minimum and maximum allowable core flow statepoints for powers ranging from 25% to 100% rated power. These same statepoints were used to analyze the event with and without FWHOOS at MOC.

The safety/relief valve action is conservatively assumed to occur with higher than nominal set points.

15.1.2.3.3 Results

The response of various reactor and plant parameters to the FWCF without bypass event initiated at 100% power/105% core flow are shown in Reference 8. The high water level turbine trip and feedwater pump trips are initiated at approximately 11 seconds. Scram occurs simultaneously, and limits the neutron flux peak and fuel thermal transient so that no fuel damage is sustained. For a given initial power/flow condition, the ΔCPRs increase with exposure. For a given exposure condition, the ΔCPRs are generally higher for a lower value of initial power. For a given power/flow and exposure condition, the ΔCPRs are generally higher for FWHOOS condition. Analyses of the FWCF event at several power/flow combinations were performed to validate the operating power-flow map. The cases of FWCF with bypass and with feedwater heaters out of service were previously analyzed and shown to be bounded by FWCF without bypass case. The turbine bypass system is not assumed to function and the safety/relief valves open to limit pressure in the steam dome.

The level will gradually drop to the low low level trip point (Level 2), activating the RCIC/HPCS systems for long term level control.

15.1.2.3.4 Consideration of Uncertainties

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

Note that, while it is true that there will be a drop in the feedwater temperature with an increase in feedwater flow, the feedwater heater usually has a large time constant (in minutes, not in seconds) so that the feedwater temperature change is very slow. In addition, there is a long transport delay time before the cold feedwater will reach the vessel. Therefore, it is expected that the feedwater temperature change during the first part of the feedwater controller failure (maximum demand) transient is insignificant, and its effect on the transient severity is minimal.

15.1.2.4 Barrier Performance

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

15.1.2.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures; radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in subsection 15.2.4.5. Therefore, the radiological exposures noted in subsection 15.2.4.5 cover the consequences of this event.

15.1.2.6 Initial Cycle

15.1.2.6.1 Identification of Causes and Frequency Classification

The potential event causes and frequency classification did not change from cycle to cycle.

15.1.2.6.2 Sequence of Events and Systems Operation

15.1.2.6.2.1 Sequence of Events (Initial Cycle)

[HISTORICAL INFORMATION] [With excess feedwater flow the water level rises to the high level trip point at which time the feedwater pumps and the main turbine are tripped and a scram is initiated from the turbine trip. Table 15.1-3a lists the sequence of events for Figure 15.1-3. The figure shows the changes in important variables during this transient.]

15.1.2.6.2.2 Systems Operation

[HISTORICAL INFORMATION] [Systems operation assumed is similar for the initial cycle to the current except that in the initial cycle turbine bypass was credited.]

15.1.2.6.2.3 The Effect of Single Failures and Operators Errors

Refer to the current cycle discussion.

15.1.2.6.3 Core and System Performance

15.1.2.6.3.1 Mathematical Model (Initial Cycle)

[HISTORICAL INFORMATION] [The predicted dynamic behavior was determined using a computer simulated, analytical model of a generic direct-cycle BWR. This model is described in detail in Reference 2 (Section 15.1.7). This computer model has been improved and verified through extensive comparison of its predicted results with actual BWR test data.

The nonlinear computer simulated analytical model is similar to the current cycle model. An additional feature of the model used in the initial cycle is that the control systems and reactor protection system models are, for the most part, identical to those employed in the point reactor model, which is described in detail in Reference 1 (Section 15.1.7) and used in analysis for other transients in the initial cycle.]

15.1.2.6.3.2 Input Parameters and Initial Conditions (Initial Cycle)

[HISTORICAL INFORMATION] [These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15.0-3.

End of cycle one, nuclear scram characteristics are assumed. The safety/relief valve action is conservatively assumed to occur with higher than nominal set points. The transient is simulated by programming an upper limit failure in the feedwater system such that 130 percent feedwater flow occurs.]

15.1.2.6.3.3 Results

[HISTORICAL INFORMATION] [The simulated feedwater controller transient for the initial cycle is shown in Figure 15.1-3. The high water level turbine trip and feedwater pump trip are initiated at approximately 12 sec. Scram occurs simultaneously, and limits the neutron flux peak and thermal transient so that no fuel damage occurs. MCPR remains above the safety limit. The turbine bypass system and the safety/relief valves open to limit peak pressure in the steam line near the safety/relief valves to 1166 psig and the pressure at the bottom of the vessel to about 1188 psig.

The level will gradually drop to the low low level trip point (Level 2), activating the RCIC/HPCS systems for long term level control.]

15.1.2.6.3.4 Consideration of Uncertainties

No changes from current cycle. See current cycle discussion.

15.1.2.6.4 Barrier Performance and Radiological Consequences

No changes from current cycle. See current cycle discussion.

15.1.3 Pressure Controller Failure - Open

The reload fuel vendor has determined that the pressure controller failure - open event is not a limiting event for the current reload cycle. Therefore, this subsection describes the original analysis performed by the NSSS vendor for the initial cycle which remains the current licensing basis for GGNS. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.1.3.1 Identification of Causes and Frequency Classification

15.1.3.1.1 Identification of Causes

The total steam flow rate to the main turbine resulting from a pressure control malfunction is limited by a maximum flow limiter imposed at the turbine controls. This Limiter is set to limit maximum steam flow to approximately 115 percent of NB rated.

If the controlling pressure controller fails to the open position, the turbine control valves can be fully opened and the turbine bypass valves can be partially opened until the maximum steam flow is established.

15.1.3.1.2 Frequency Classification

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

15.1.3.2 Sequence of Events and Systems Operation

15.1.3.2.1 Sequence of Events

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

15.1.3.2.1.1 Deleted

15.1.3.2.2 Systems Operation

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

Initiation of HPCS and RCIC system functions will occur when the vessel water level reaches the L2 set point. Normal startup and actuation can take up to 30 seconds before effects are realized.

If these events occur, they will follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred, and are expected to be less severe than those already experienced by the system.

15.1.3.2.3 The Effect of Single Failures and Operator Errors

This transient leads to a loss of pressure control such that the increased steam flow demand causes a depressurization.

Instrumentation for pressure sensing of the turbine inlet pressure is designed to be single failure proof for initiation of MSIV closure.

Reactor scram sensing, originating from limit switches on the main steam line isolation valves, is designed to be single failure proof. It is therefore concluded that the basic phenomenon of pressure decay is adequately terminated. See Appendix 15A for a detailed discussion of this subject.

15.1.3.3 Core and System Performance

15.1.3.3.1 Mathematical Model

The nonlinear dynamic model described briefly in subsection 15.1.1.6.3.1 is used to simulate this event.

15.1.3.3.2 Input Parameters and Initial Conditions

This transient is simulated by setting the pressure controller output to a high value, which causes the turbine control valves to open fully and the turbine bypass valves to open partially. A controller failure with 130 percent steam flow was simulated as a worst case since 115 percent is the normal maximum flow limit.

A 5-second isolation valve closure instead of a 3-second closure is assumed when the turbine pressure decreases below the turbine inlet low pressure set point for main steam line isolation initiation. This is within the specification limits of the valve and tends to aggravate the results of the analysis.

The manual recirculation flow control mode is assumed in the analysis of the pressure regulator failure-open transient. Should the automatic flow control mode be assumed, the recirculation control system would react to the reactor power decrease and initiate an increase of core flow by opening the flow control valves. This would result in a slight increase in reactor power, but the initial power level could not be maintained. When the core flow reached its maximum value, the reactor power would start to fall again. Therefore, with the automatic flow control mode, the initial depressurization rate for this transient will be slightly less than what is analyzed with the assumed manual flow control mode leading to conservative results.

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

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

15.1.3.3.3 Results

Figure 15.1-4 shows graphically how the isolation valve closure stops vessel depressurization and produces a normal shutdown of the isolated reactor.

The main steam line isolation valves automatically close at approximately 6.7 sec when pressure at the turbine decreases below 825 psig. Depressurization results in formation of voids in the reactor coolant and causes a rapid decrease in reactor power almost immediately. The reactor scrams at approximately 7.2 sec as a result of main steam line isolation valve closure. Reactor vessel isolation limits the duration and severity of the depressurization so that no significant thermal stresses are imposed on the reactor coolant pressure boundary. After the rapid portion of the transient is complete and the isolation effective, the nuclear system safety/relief valves operate intermittently to relieve the pressure rise that results from decay heat generation. No significant reductions in fuel thermal margins occur. Because the rapid portion of the transient results in only momentary depressurization of the nuclear system and because the safety/relief valves need operate only to relieve the pressure increase caused by decay heat, the reactor coolant pressure boundary is not threatened by high internal pressure.

The event analyzed assumes that the pressure regulator fails at time zero with steam flow demand of 130 percent. This demand causes turbine control valves to open to their full-open positions and turbine bypass valves to open to such positions that the steam flow demand is satisfied. For Grand Gulf, the turbine bypass valves will not open to their full-open positions due to the high bypass capacity of 30.4 percent NBR. While turbine control valves and bypass valves start to open, the vessel steam flow increases to satisfy the demand. However, the increase of steam flow results in depressurization in the reactor core and increase in void formation inside the reactor core. The void increase reduces the reactor core power due to the negative void reactivity coefficient. Therefore, the reactor core power is not enough to supply steam flow to meet the 130 percent demand, as shown in Figure 15.1-4.

15.1.3.3.3.1 Considerations of Uncertainties

If the maximum flow limiter were set higher or lower than normal, there would result a faster or slower loss in nuclear steam pressure. The rate of depressurization may be limited by the bypass capacity, but it is unlikely.

The turbine valves will open to the valves-wide-open state, admitting slightly more than the rated steam flow, and with the limiter in this analysis set to fail at 115 percent something less than 15 percent bypass would be expected.

This is therefore not a limiting factor on this plant. If the rate of depressurization does change it will be terminated by the low turbine inlet pressure trip set point.

Depressurization rate has a proportional effect upon the voiding action of the core. If it is large enough, the sensed vessel water level trip set point (L9) may be reached initiating turbine and feedwater pump trip early in the transient. Reactor scram will be initiated by turbine trip and will shut down the reactor. Since main turbine is tripped, the depressurization will be terminated.

15.1.3.4 Barrier Performance

The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed. Peak pressure in the bottom of the vessel reaches 1130 psig, which is below the ASME code limit of 1375 psig for the reactor coolant pressure boundary. Vessel dome pressure reaches 1127 psig, just slightly below the set point of the second pressure relief group.

15.1.3.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures; radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in subsection 15.2.4.5. Therefore, the radiological exposures noted in subsection 15.2.4.5 cover the consequences of this event.

15.1.4 Inadvertent Safety/Relief Valve Opening

The reload fuel vendor has determined that the inadvertent SRV opening event is not a limiting event for the current reload cycle. Therefore, this subsection describes the original analysis performed by the NSSS vendor for the initial cycle which remains the current licensing basis for GGNS. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

Inadvertent opening of a safety/relief valve can lead to two possible events. First, the valve may “open” and “reclose.” This event has no significant effect on plant operation. Second, the valve may “open” and stick in the “open” position. This is the more limiting case and results in the plant transient discussed below.

15.1.4.1 Identification of Causes and Frequency Classification

15.1.4.1.1 Identification of Causes

Cause of inadvertent opening is attributed to malfunction of the valve or an operator initiated opening. Opening and closing circuitry at the individual valve level (as opposed to groups of valves) is subject to a single failure impact. It is therefore simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve is provided in Section 5.4.

15.1.4.1.2 Frequency Classification

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

15.1.4.2 Sequence of Events and Systems Operation

15.1.4.2.1 Sequence of Events

Table 15.1-5 lists the sequence of events for Figure 15.1-5.

15.1.4.2.1.1 Identification of Operator Actions

The Technical Specification limit for suppression pool temperature during normal operation is 95˚F. At this temperature the operator must take action to restore pool temperature below this limit. With an initial pool temperature of 80˚F, the operator has 7.5 minutes before the (95˚F) Technical Specification limit is exceeded. Assuming no action is taken at a pool temperature of 95˚F, the operator has an additional 7.5 minutes prior to reaching the Technical Specification limit of 110˚F, which requires the initiation of plant shutdown.

15.1.4.2.2 Systems Operation

In this transient, the analysis assumes normal functioning of plant instrumentation and controls, specifically, the relief valve discharge line temperature sensors and the suppression pool temperature sensors and levels control systems. Additionally, minimum reactor and plant protection systems, ECCS flow and RHR pool cooling, are required.

15.1.4.2.3 The Effect of Single Failures and Operator Errors

In the event of a stuck open safety/relief valve, a single failure or operator error would simply activate the reactor protection system resulting in a plant shutdown. Analysis of such transients has been considered in other sections of Section 15. Therefore a single failure or operator error cannot increase the severity of this event. See Appendix 15A for a detailed discussion.

15.1.4.3 Core and System Performance

15.1.4.3.1 Mathematical Model

The reactor model briefly described in subsection 15.1.1.6.3.1 was previously used to simulate this event in earlier FSARs. This model is discussed in detail in Reference 1. It was determined that this event is not limiting from a core performance standpoint. Therefore a qualitative presentation of results is described below.

15.1.4.3.2 Input Parameters and Initial Conditions

It is assumed that the reactor is operating at an initial power level corresponding to 105 percent of initial licensed rated steamflow conditions when a safety/relief valve is inadvertently opened. Manual recirculation flow control is assumed. Flow through the valve at normal plant operating conditions stated above is approximately 775,000 lb/hr.

15.1.4.3.3 Qualitative Results

The opening of a safety/relief valve allows steam to be discharged into the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient.

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

15.1.4.4 Barrier Performance

As discussed above, the transient resulting from a stuck open relief valve is a mild depressurization which is within the range of normal load following and therefore has no significant effect on RCPB and containment design pressure limits.

15.1.4.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures; radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in subsection 15.2.4.5. Therefore, the radiological exposures noted in subsection 15.2.4.5 cover the consequences of this event.

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

This event is not applicable to BWR plants.

15.1.6 Inadvertent RHR Shutdown Cooling Operation

The reload fuel vendor has determined that the inadvertent RHR shutdown cooling operation event is not a limiting event for the current reload cycle. Therefore, this subsection describes the original analysis performed by the NSSS vendor for the initial cycle which remains the current licensing basis for GGNS. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.1.6.1 Identification of Causes and Frequency Classification

15.1.6.1.1 Identification of Causes

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

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

15.1.6.1.2 Frequency Classification

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

15.1.6.2 Sequence of Events and Systems Operation

15.1.6.2.1 Sequence of Events

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

15.1.6.2.2 System Operation

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

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

15.1.6.2.3 Effect of Single Failures and Operator Action

No single failures can cause this event to be more severe. If the operator takes action, the slow power rise will be controlled in the normal manner. If no operator action is taken, scram will terminate the power increase before thermal limits are reached. (See Appendix 15A for details.)

15.1.6.3 Core and System Performance

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

15.1.6.4 Barrier Performance

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

15.1.6.5 Radiological Consequences

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

15.1.7 References

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

2. Odar, F., "Safety Evaluation for General Electric Topical Report: Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154, NEDE-24154P, Volumes 1, 2, and 3, 1980.

3. "Three Dimensional BWR Core Simulator," NEDO-20953-A, January 1977.

4. Letter, J.S. Charnley (G.E.) to F.J. Miraglia (NRC), "Loss of Feedwater Heating Analysis," July 5, 1983 (MFN-125-83).

5. "Steady-State Nuclear Methods," NEDE-30130-P-A, and NEDO-30130-A, April 1985.

6. "General Electric Standard Application for Reactor Fuel (GESTAR II)," NEDE-24011-P-A, latest approved version.

7. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," Volumes 1, 2, and 3, NEDO-24154-A, and NEDO-24154-P-A, August 1986.

8. ECH-NE-18-00022, Rev. 1, "Supplemental Reload Licensing Report for Grand Gulf Nuclear Station Reload 21 Cycle 22," July 2018.

9. Deleted

10. Deleted

11. Deleted

TABLE 15.1-1: [HISTORICAL INFORMATION] SEQUENCE OF EVENTS FOR LOSS OF FEEDWATER HEATER, AUTO FLOW CONTROL INITIAL CYCLE (FIGURE 15.1-1)

Time-sec	Event
0	Initiate a 100°F temperature reduction in the feedwater system.
5	Initial effect of unheated feedwater starts to raise core power level but feedwater control system automatically reduces core flow to maintain initial steam flow
40+	Reactor variables settle into new steady state.

TABLE 15.1-2: [HISTORICAL INFORMATION] SEQUENCE OF EVENTS FOR LOSS OF FEEDWATER HEATER, MANUAL CONTROL INITIAL CYCLE (FIGURE 15.1-2)

Time-sec	Event
0	Initiate a 100°F temperature reduction in the feedwater system.
5	Initial effect of unheated feedwater starts to raise core power level and steam flow.
6.5	Turbine control valves start to open to regulate pressure.
34	APRM initiates reactor scram on high thermal power.
49	Wide Range (WR) sensed water level reaches Level 2 (L2) set point.
64(est)	Recirculation pump trip initiated due to Level 2 Trip (not included in simulation).
79(est)	HPCS/RCIC flow enters vessel (not simulated).
80(est)	Reactor variables settle into limit cycle.

TABLE 15.1-3: SEQUENCE OF EVENTS FOR FEEDWATER CONTROLLER FAILURE W/O BYPASS (CURRENT CYCLE)

Time-sec (approx.*)	Event
0	Initiate simulated failure of upper limit on feedwater flow.
10.8	L8/L9 vessel level set point trips main turbine and feedwater pumps and initiates reactor scram.
11.0	Recirculation pump trip actuated by stop valve trip fluid pressure transmitters and trip units.
12.0	Safety/relief valves open due to high pressure.
>30.0(est)	Water level dropped to low-low water level set point (not simulated).
60.0(est)	RCIC and HPCS flow into vessel (not simulated).

*Exact timing varies based on power shape and exposure.

TABLE 15.1-3A: [HISTORICAL INFORMATION] SEQUENCE OF EVENTS FOR FEEDWATER CONTROLLER FAILURE WITH TURBINE BYPASS (INITIAL CYCLE) (FIGURE 15.1-3)

Time-sec	Event
0	Initiate simulated failure of 130% upper limit on feedwater flow at the system design pressure of 1065 psig.
11.78	L8 vessel level set point trips main turbine and feedwater pumps and initiates reactor scram.
11.79	Recirculation pump trip actuated by stop valve trip fluid pressure transmitters and trip units.
11.88	Main turbine bypass control valves start to open due to turbine trip.
13.57	Safety/relief valves open due to high pressure.
18.99	Safety/relief valves close.
>30.0(est)	Water level dropped to low-low water level set point (Level 2).
60.0(est)	RCIC and HPCS flow into vessel (not simulated).

TABLE 15.1-4: SEQUENCE OF EVENTS FOR PRESSURE CONTROLLER FAILURE - OPEN(INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT) (FIGURE 15.1-4)

Time-sec	Event
0	Simulate steam flow demand to 130%.
0+	Turbine control valves wide open.
0+	Main turbine bypass control valve opens.
6.7	Low turbine inlet pressure trip initiates main steam line isolation.
7.2	Main steam line isolation valve closure initiates reactor scram.
8.7(est)	Feedwater turbine trip due to main steamline isolation valves closure.
10.0	Vessel water level reaches L4 set point, initiates recirculation flow runback.
14.7	Vessel water level reaches L2 set point.
20.8(est)	Safety/relief valves open.
29.7	Recirculation pump trip due to Level 2 trip.
38.0	Group 1 safety/relief valves open again to relieve decay heat.
41.0	Group 1 safety/relief valves close again.
44.7	HPCS and RCIC flow enters vessel (not simulated).

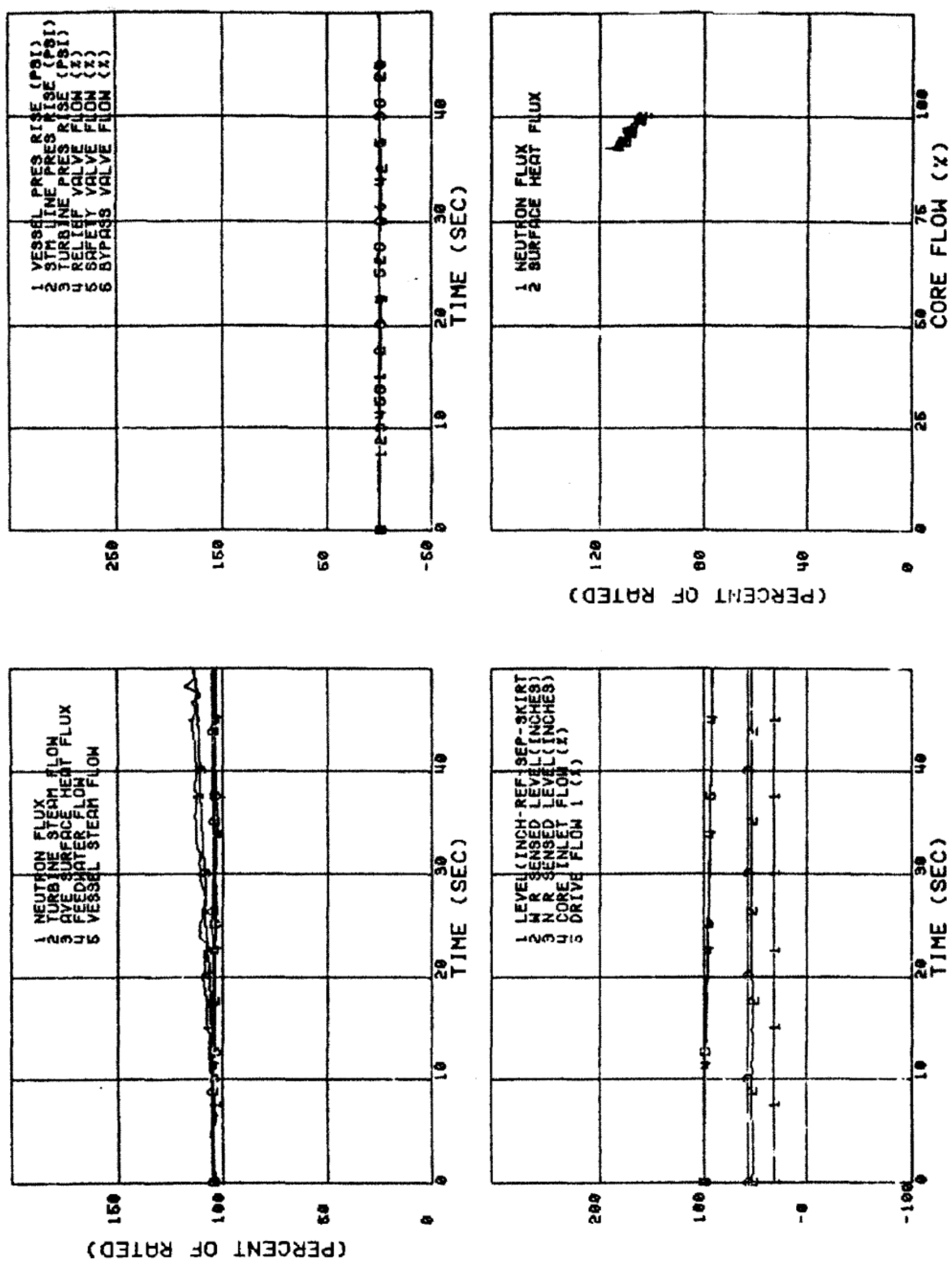
TABLE 15.1-5: SEQUENCE OF EVENTS FOR STUCK OPEN RELIEF VALVE (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT)
(FIGURE 15.1-5)

Time-minutes	Event
0	One of the primary SRVs opens and remains open throughout the event.
0+	Operator receives an alarm from thermocouples on the SRV discharge line of an open or leaking SRV.
5	Operator receives an alarm when suppression pool temperature rises to 90°F.
10	Operator attempts to close the valve unsuccessfully.
20	Operator activates RHR pool cooling.
+	Shutdown and cooldown completed.

TABLE 15.1-6: SEQUENCE OF EVENTS FOR INADVERTENT RHR SHUTDOWN COOLING OPERATION (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT)
(FIGURE 15.1-6)

Approximate Elapsed Time	Event
0	Reactor at states B or D (of Appendix 15A) when RHR shutdown cooling inadvertently activated.
0-10 min	Slow rise in reactor power.
+10 min	Operator may take action to limit power rise. Flux scram will occur if no action is taken.

TABLE 15.1-7: Deleted



Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.

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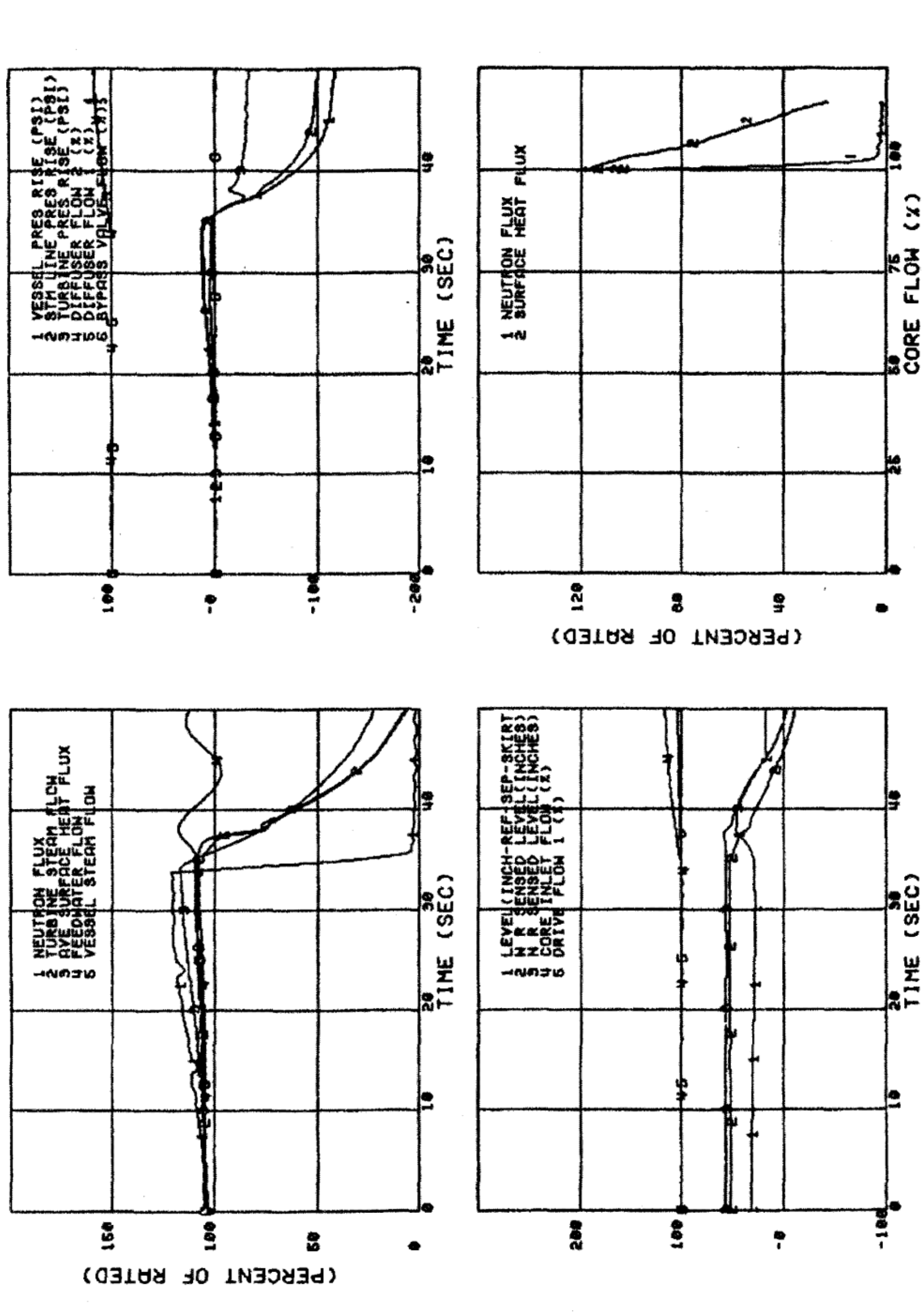
LOSS OF 100°F FEEDWATER HEATING
(AUTOMATIC FLOW CONTROL MODE)
[HISTORICAL INFORMATION]
[INITIAL CYCLE]

FIGURE 15.1-1



Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.

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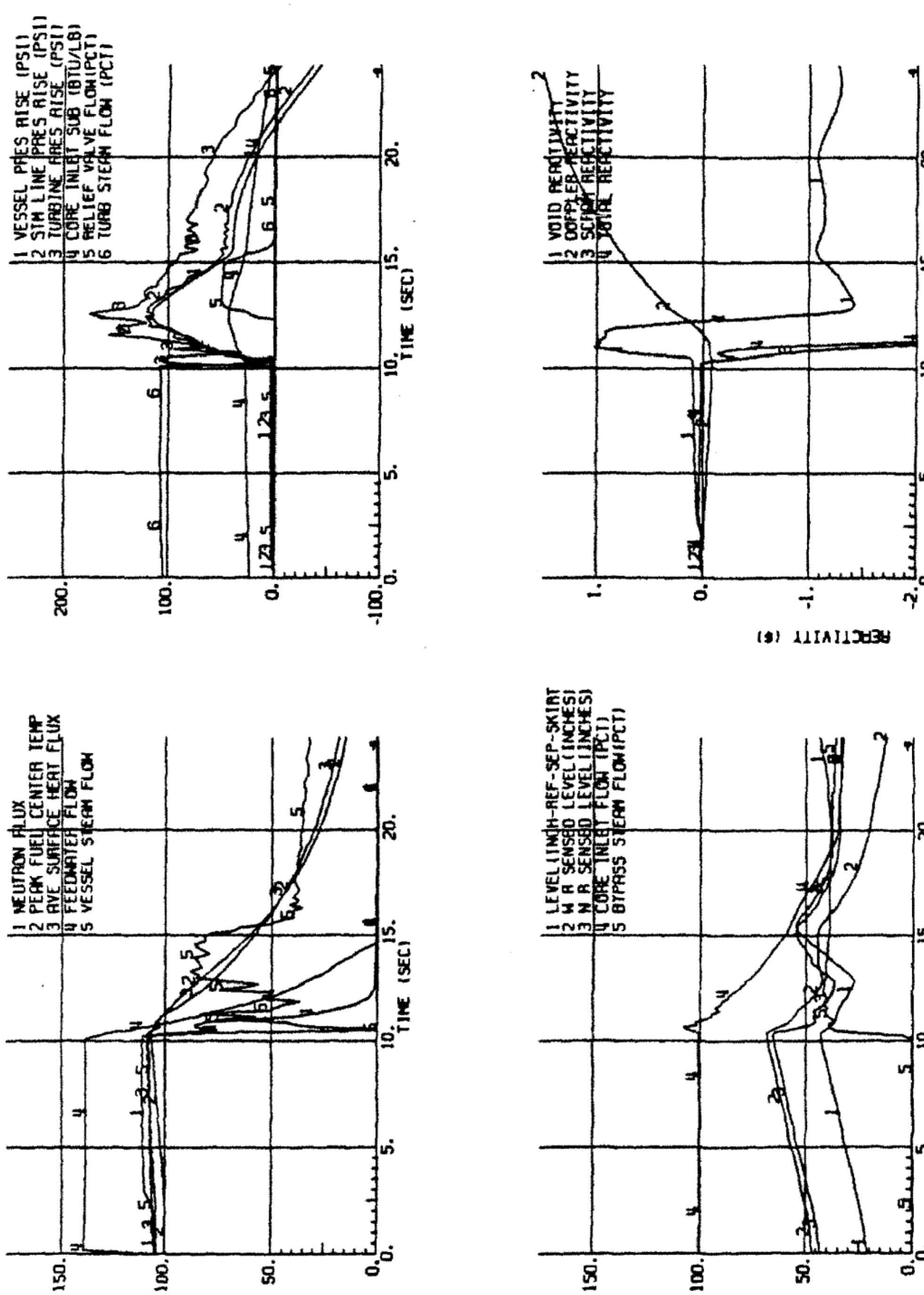
LOSS OF 100°F FEEDWATER HEATING
(MANUAL FLOW CONTROL MODE)
[HISTORICAL INFORMATION]
[INITIAL CYCLE]

FIGURE 15.1-2



Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE, MAXIMUM DEMAND (Initial Cycle) [HISTORICAL INFORMATION]
	FIGURE 15.1-3

Figure 15.1-3A Deleted

Figures 15.1-3b through 15.1-3e

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Figures 15.1-3F through 15.1-3L

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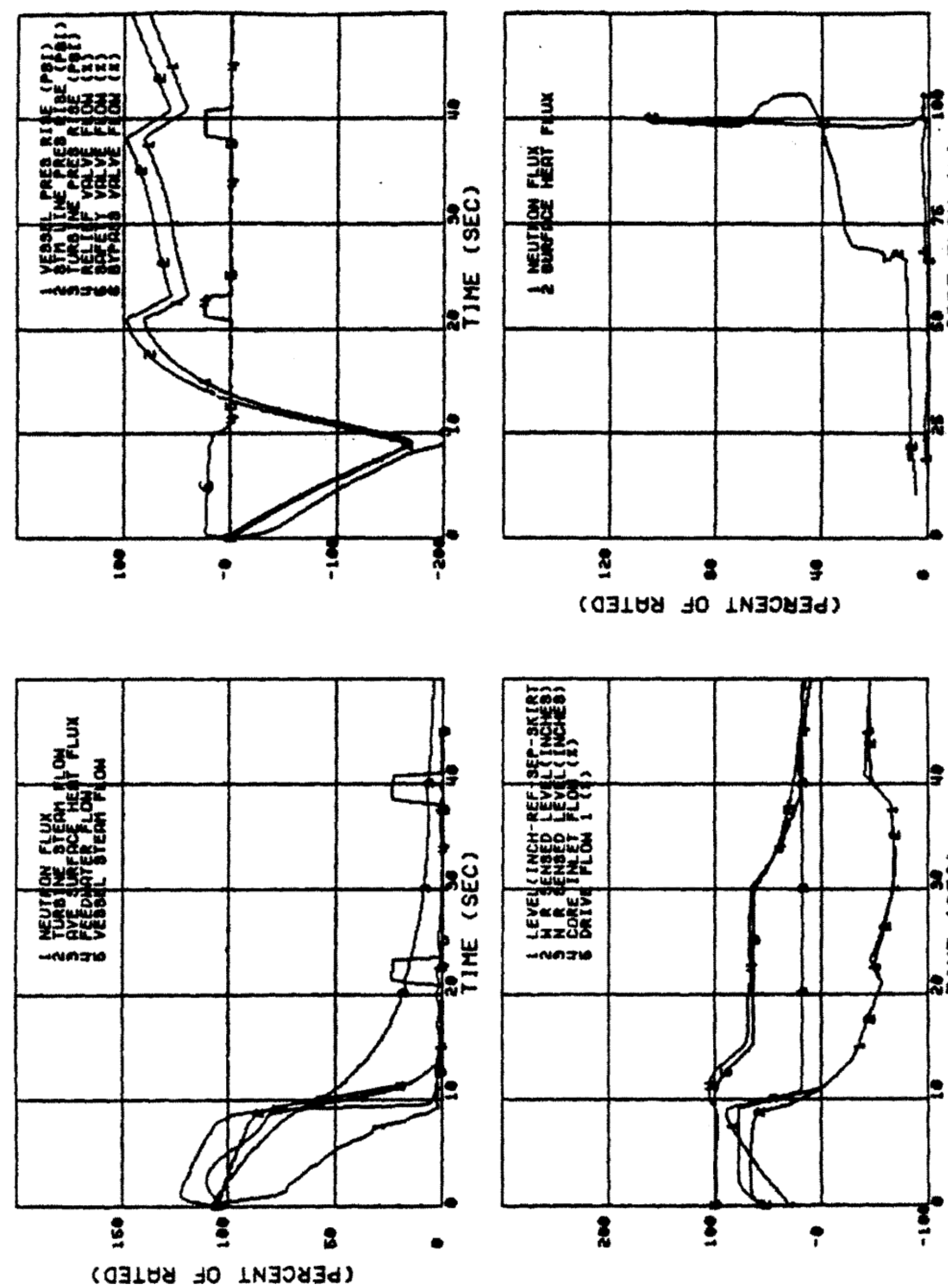

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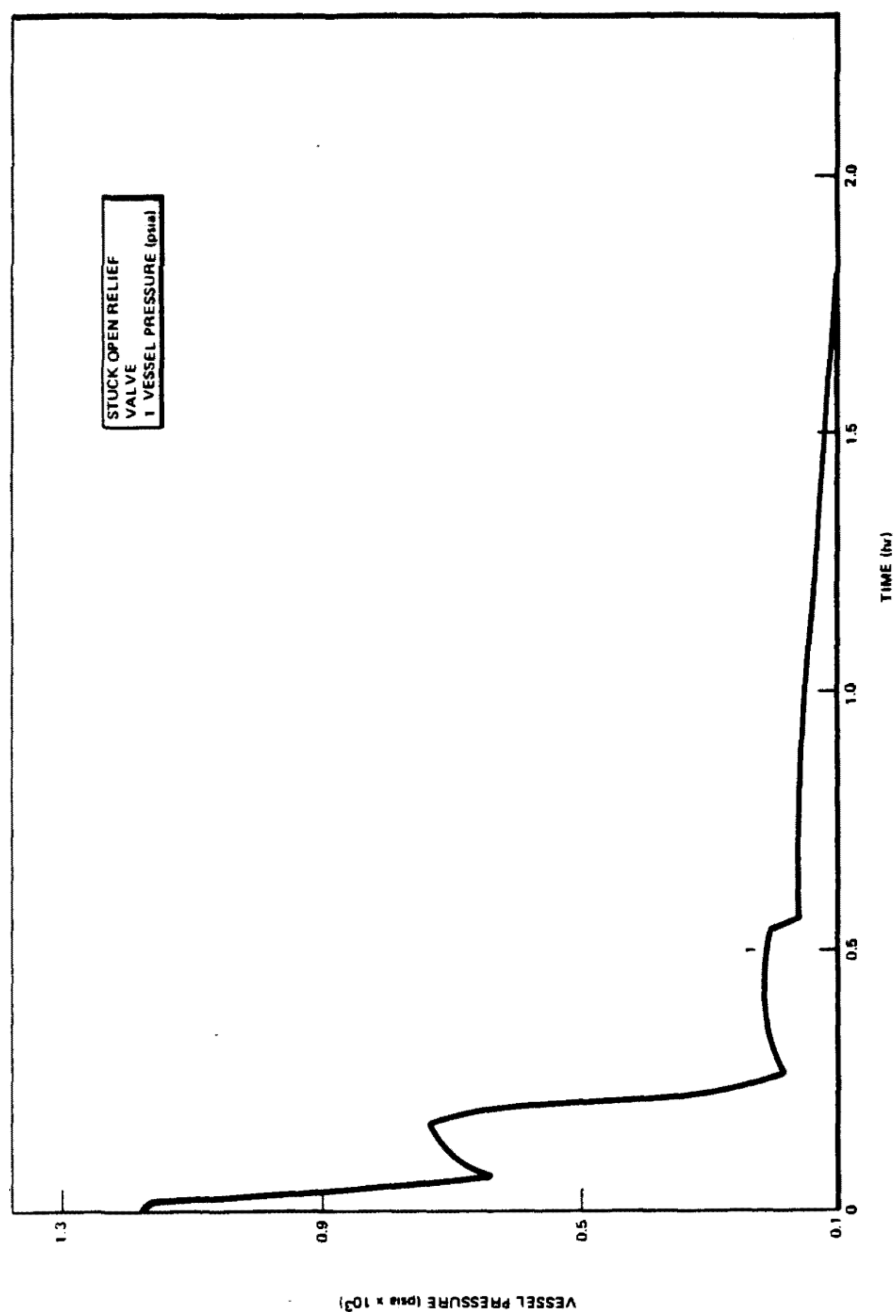
PRESSURE REGULATOR FAILURE - OPEN TO 130%
(INITIAL CYCLE)

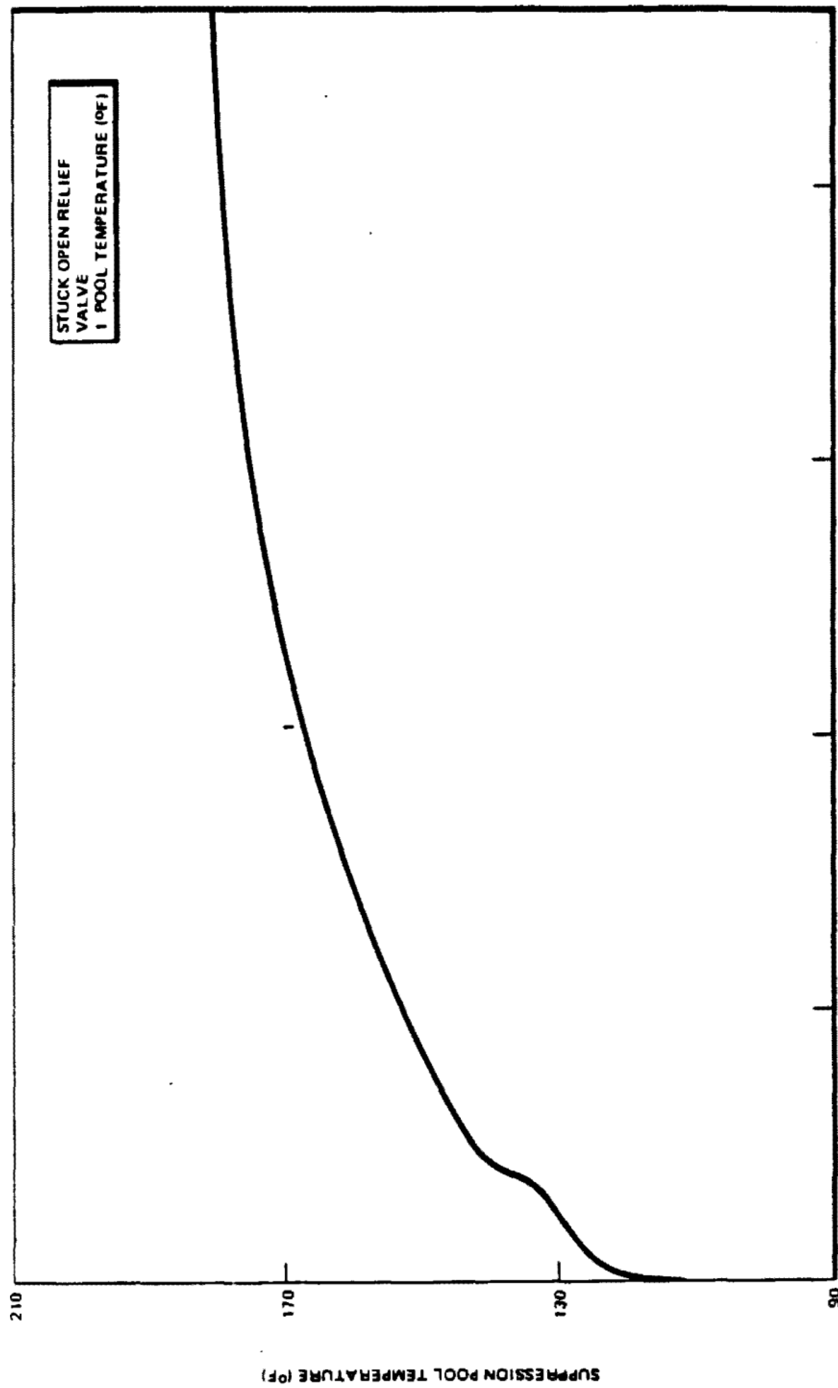
FIGURE 15.1-4

Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW but remains the current analysis of record for this event.



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	REACTOR PRESSURE VERSUS TIME (INITIAL CYCLE) FIGURE 15.1-5

Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW but remains the current analysis of record for this event.



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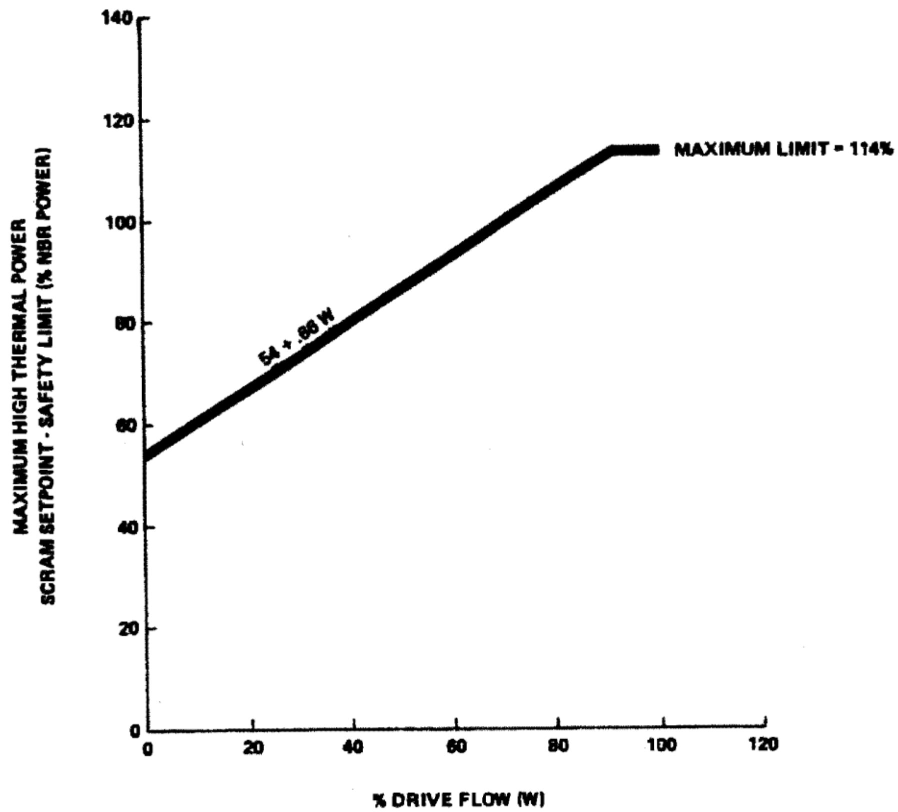
SUPPRESSION POOL TEMPERATURE VERSUS TIME (INITIAL CYCLE)

FIGURE 15.1-6

Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW but remains the current analysis of record for this event.



Note: Initial cycle NBR power was 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	HIGH THERMAL POWER SCRAM SETPOINT FOR PLANT OPERATION UP TO 100% NBR POWER [HISTORICAL INFORMATION] [INITIAL CYCLE] FIGURE 15.1-7

* Figure 15.1-8 is Deleted*

15.2 INCREASE IN REACTOR PRESSURE

15.2.1 Pressure Controller Failure - Closed

The digital Ovation Turbine Control and Protection System (TCPS) is a dual redundant controller system. If one pressure controller fails, an alarm will be generated and a bumpless transfer to the backup controller will occur. The TCPS will continue to regulate pressure and will maintain control of the turbine and bypass valves. The failure of one TCPS pressure controller is an infrequent incident. The failure of the backup controller while the primary controller is out of service is also an infrequent incident. Failure of both pressure controllers in the downscale direction will cause the turbine control valves to close with no bypass valve opening. The failure of both pressure controllers downscale is a limiting fault, however, the analyses in this subsection are retained as licensing bases.

There are three scenarios evaluated for the pressure controller failure - closed event; one where just one channel fails, one where one pressure controller has failed and the second (redundant) controller slowly fails resulting in downscale failure, and the other where the pressure control demand goes to zero (downscale failure). The reload fuel vendor has determined that the pressure control downscale failure event is a limiting event which requires analysis for the current reload cycle. This subsection describes both analyses; 1) the analysis performed by the NSSS vendor for the one pressure controller channel failure event for the initial fuel cycle which remains the current analysis for this scenario, and 2) the analysis performed by the reload fuel vendor for the pressure control downscale failure scenario for the current fuel cycle. For a historical description of the initial fuel cycle analysis of the pressure control downscale failure scenario, refer to subsection 15.2.1.6. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.2.1.1 Identification of Causes and Frequency Classification

15.2.1.1.1 Identification of Causes

A dual-channel digital pressure-control system with an internal supervisory subsystem is used. If one pressure controller fails, an alarm will be generated and a bumpless transfer to the backup controller will occur. The TCPS will continue to regulate pressure and will maintain control of the turbine and bypass valves.

Two separate measurements of actual pressure are made in each of the four main steam lines and passed four averaging algorithms, with each output processed by a four-signal validation algorithm that selects the second highest pressure signals and used as feedback signal to the pressure regulators.

It is assumed for purposes of this transient analysis that a single failure occurs which erroneously causes a pressure controller channel to start closing the main control valves. If this occurs, the monitoring circuit detects the controller failure and transfers to the backup controller giving an alarm in the control room.

It is also assumed for purposes of this transient analysis that a single failure occurs which causes a downscale failure of the pressure control demand to zero (e.g., average and comparison circuit downscale failure). Should this occur, it could cause full closure of turbine control valves in their servo mode (not fast closure) as well as inhibit steam bypass flow and thereby increase reactor power and pressure. When this occurs, a reactor scram will be initiated when either the high neutron flux or high vessel dome pressure scram set point is reached.

15.2.1.1.2 Frequency Classification

15.2.1.1.2.1 One Pressure Controller Channel Failure - Closed

This event is treated as a moderate frequency event.

15.2.1.1.2.2 Pressure Controller Downscale Failure

This event has been licensed as an infrequent event. A probabilistic evaluation has demonstrated that this classification is applicable even with a sub-system out of service.

15.2.1.2 Sequence of Events and System Operation

15.2.1.2.1 Sequence of Events

15.2.1.2.1.1 One Pressure Controller Channel Failure - Closed

Postulating a failure of one channel of the pressure controller in the closed mode as discussed in subsection 15.2.1.1.1 will cause the valves to start closing momentarily. The monitoring circuit switches out the defective channel and the remaining channels will reopen the valves and reestablish steady-state operation to the power level.

15.2.1.2.1.2 Pressure Control Downscale Failure

Table 15.2-1A lists the current cycle sequence of events for the pressure control downscale failure event.

15.2.1.2.1.3 Deleted

15.2.1.2.2 Systems Operation

15.2.1.2.2.1 One Pressure Controller Channel Failure - Closed

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

15.2.1.2.2.2 Pressure Control Downscale Failure

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

Specifically this transient takes credit for high neutron flux scram to shut down the reactor. When the reactor is operating at less than full power, the high neutron flux scram may not be initiated. Under these conditions, the high dome pressure scram is credited. High system pressure is limited by the pressure relief valve system operation.

15.2.1.2.3 The Effect of Single Failures and Operator Errors

15.2.1.2.3.1 One Pressure Controller Channel Failure - Closed

The nature of the first assumed failure produces a slight pressure increase in the reactor until the remaining pressure controller channels gain control, since no other action is significant in restoring normal operation. If the remaining pressure controller channels fail at this time, the second assumed failure, the control valves would start to close, raising reactor pressure to the point where a flux scram or pressure scram trip would be initiated to shut down the reactor. This event is similar to that described in subsection 15.2.1.2.1.1. Detailed discussions on this subject can be found in Appendix 15A.

15.2.1.2.3.2 Pressure Control Downscale Failure

This transient leads to a loss of pressure control such that the zero steam flow demand causes a pressurization. The high neutron flux or high dome pressure scram is the mitigating system and is designed to be single failure proof. Therefore, single failures are not expected to result in a more severe event than analyzed. Detailed discussions on this subject can be found in Appendix 15A.

15.2.1.3 Core and System Performance

15.2.1.3.1 Mathematical Model

The computer model described briefly in subsection 15.1.2.6.3.1 is used to simulate the one pressure controller channel failure event. The nonlinear, dynamic model described briefly in subsection 15.1.2.3.1 is used to simulate the pressure control downscale failure event.

15.2.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-3 for the one pressure controller channel failure event and in Table 15.0-4 for the current cycle analysis of the pressure control downscale failure event.

15.2.1.3.3 Results

15.2.1.3.3.1 One Pressure Controller Channel Failure - Closed

Qualitative evaluation provided only.

Response of the reactor during one pressure controller channel failure is such that there is no significant increase in pressure at the turbine due to the partial closing action of the turbine control valves which reopen when the remaining pressure controller channels gain control.

15.2.1.3.3.2 Pressure Control Downscale Failure

A pressure control downscale failure is simulated at 100% power and 105% flow as shown in Figure 15.2-1A for the current cycle.

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 set point, a reactor scram is initiated. The neutron flux increase is limited to 139% of rated by the reactor scram. Peak fuel surface heat flux does not exceed 105% of its initial value. Those rods calculated to experience boiling transition, excessive cladding strain, or centerline melt are assumed to fail, releasing gap source terms into the reactor coolant.

15.2.1.3.4 Consideration of Uncertainties

All systems utilized for protection in this event were assumed to have the poorest allowable response (e.g., relief set points, scram stroke time, and work characteristics). Expected plant behavior is, therefore, expected to reduce the actual severity of the transient.

15.2.1.4 Barrier Performance

15.2.1.4.1 One Pressure Controller Channel Failure - Closed

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed (see Table 15.0-1); therefore, these barriers maintain their integrity and function as designed.

15.2.1.4.2.1 Pressure Control Downscale Failure (Initial Cycle)

Peak pressure at the safety/relief valves reaches 1192 psig. The peak nuclear system pressure reaches 1231 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1375 psig.

15.2.1.4.2.2 Pressure Control Downscale Failure (Current Cycle)

The peak nuclear system pressure and the peak dome pressure for this event at 100% power and 105% flow do not approach the barrier transient pressure limits of 1375 psig and 1325 psig, respectively.

15.2.1.5 Radiological Consequences

A limited number of fuel failures may result from a pressure control downscale failure event. The resulting offsite doses have been calculated to be no more than a small fraction of the limits in 10CFR50.67.

15.2.1.6 Pressure Control Downscale Failure (Initial Cycle)

The reload fuel vendor has determined that the pressure control downscale failure event is a limiting event which requires analysis for the current reload cycle. This subsection describes the analysis performed by the NSSS vendor for the initial fuel cycle. For a description of the current fuel cycle analysis of this event, refer to subsection 15.2.1. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.2.1.6.1 Identification of Causes and Frequency Classification (Initial Cycle)

The potential causes of this event are the same as for the current cycle. The probability of occurrence of this event is also the same as that for the current cycle. Refer to subsection 15.2.1.1.

15.2.1.6.2 Sequence of Events and Systems Operation (Initial Cycle)

15.2.1.6.2.1 Sequence of Events

[HISTORICAL INFORMATION] [Table 15.2-1 lists the initial cycle sequence of events for Figure 15.2-1.]

15.2.1.6.2.2 Systems Operation

The description of systems operation for this event is the same as for the current cycle. Refer to subsection 15.2.1.2.2.2.

15.2.1.6.2.3 The Effect of Single Failures and Operator Errors

The effect of single failures and operator errors for this event are the same as for the current cycle. Refer to subsection 15.2.1.2.3.2.

15.2.1.6.3 Core and System Performance (Initial Cycle)

15.2.1.6.3.1 Mathematical Model

The computer model described briefly in subsection 15.1.2.6.3.1 was used to simulate this event for the initial cycle.

15.2.1.6.3.2 Input Parameters and initial Conditions

[HISTORICAL INFORMATION] [The analysis has been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0.3 for this event.]

15.2.1.6.3.3 Results (Initial Cycle)

[HISTORICAL INFORMATION] [A pressure control downscale failure was simulated at 105% of initially licensed NB rated steam flow conditions as shown in Figure 15.2-1 for the initial cycle.]

15.2.1.6.3.4 Consideration of Uncertainties

The description of uncertainties for this event is the same as for the current cycle. Refer to subsection 15.2.1.3.4.

15.2.1.6.4 Barrier Performance (Initial cycle)

Peak pressure at the safety/relief valves reaches 1192 psig. The peak nuclear system pressure reaches 1231 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1375 psig.

15.2.1.6.5 Radiological Consequences (Initial Cycle)

The description of radiological consequences for this event is the same as for the current cycle. Refer to subsection 15.2.1.5.

15.2.2 Generator Load Rejection

There are two scenarios evaluated for the Generator Load Rejection event; one with Bypass, and the other without (w/o) Bypass. The reload fuel vendor has determined that the Generator Load Rejection, No Bypass (LRNB) event is a limiting event which requires analysis for each fuel loading cycle. This subsection describes both analyses; 1) the analysis performed by the NSSS vendor for the generator load rejection with bypass event for the initial fuel cycle which remains the current analysis for this event, and 2) the analysis performed by the reload vendor for the LRNB event for the current fuel cycle. For a description of the initial fuel cycle analysis of the LRNB event, refer to subsection 15.2.2.6. For additional information on the relationship between the analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload fuel vendor for the current cycle, refer to Section 15.0.

15.2.2.1 Identification of Causes and Frequency Classification

15.2.2.1.1 Identification of Causes

A turbine trip and fast closure of the turbine control valves (TCVs) is initiated by the Load rejection circuitry whenever electrical grid disturbances occur which result in loss of electrical load on the generator in excess of 86 percent rated load while the turbine is carrying more than 86 percent rated load. The turbine control valves are designed to close as rapidly as possible in response to Load Rejection circuitry actuation to prevent mechanical overspeed trip of the turbine-generator. Fast closure of the turbine control valves will cause a sudden reduction in steam flow, which results in an increase in system pressure and reactor shutdown.

15.2.2.1.2 Frequency Classification

15.2.2.1.2.1 Generator Load Rejection with Bypass

This event is categorized as an incident of moderate frequency.

15.2.2.1.2.2 Generator Load Rejection w/o Bypass

This event is categorized as an incident of moderate frequency. Frequency is expected to be as follows:

Frequency: 0.0036/plant year

MTBE: 278 years

[HISTORICAL INFORMATION] [Frequency Basis: Thorough searches of domestic plant operating records have revealed three instances of bypass failure during 628 bypass system operations. This gives a probability of bypass failure of 0.0048. Combining the actual frequency of a generator load rejection with the failure rate of the bypass yields a frequency of a generator load rejection with bypass failure of 0.0036 event/plant year.]

15.2.2.2 Sequence of Events and System Operation

15.2.2.2.1 Sequence of Events

15.2.2.2.1.1 Generator Load Rejection with Bypass

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

15.2.2.2.1.2 Generator Load Rejection w/o Bypass

The LRNB event is the most limiting of the class of transients characterized by rapid vessel pressurization for Grand Gulf Unit 1. The load rejection causes a turbine trip and fast closure of the TCVs. The resulting compression wave travels through the steam lines into the vessel and creates the rapid pressurization condition. A reactor scram and a recirculation pump transfer from high to low speed are initiated by fast closure of the control valves. Condenser bypass flow, which can mitigate the pressurization effect, is not credited. The excursion of the core power due to void collapse is primarily terminated by the reactor scram and void growth due to the recirculation pump fast speed breaker trips. The sequence of events for this transient are listed in Table 15.2-3.

15.2.2.2.1.3 Deleted

15.2.2.2.2 System Operation

15.2.2.2.2.1 Generator Load Rejection with Bypass

The NSSS vendor's analysis results show that the LRNB results in a more severe response than with bypass available. Table 15.0-1 shows the relative severity of the event both with and without turbine bypass available. The analyses performed by the reload fuel vendor for the current cycle assume that the turbine bypass system is unavailable for all initial power/flow statepoints analyzed.

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

Turbine control valve fast closure initiates a scram trip signal for power levels greater than 40 percent NB rated. In addition recirculation pump trip is initiated. Both of these trip signals satisfy single failure criterion and credit is taken for these protection features.

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

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

15.2.2.2.2.2 Generator Load Rejection w/o Bypass

The assumptions are the same as subsection 15.2.2.2.2.1 except that failure of the main turbine bypass valves is assumed for the entire transient.

The event was analyzed with and without Feedwater Heaters Out of Service (FWHOOS) at the Middle of Cycle (MOC) and End of Cycle (EOC) All Control Rods Out (ARO) condition for a large number of minimum and maximum allowable core flow statepoints for powers ranging from 100% to 25% of rated power. These same statepoints were used to analyze the event with and without Feedwater Heaters Out of Service at MOC. For initial powers below 35.4%, the direct scrams on turbine stop valve closure, turbine control valve fast closure, and recirculation pump downshift are disabled (See Section 15.2.3.3.3.3 for further discussion of low power cases). Six safety/relief valves in the relief mode and nine in the safety mode are assumed to be available. The opening setpoints used in the analyses and other significant input parameters and initial conditions are listed in Table 15.0-4.

15.2.2.2.3 The Effect of Single Failures and Operator Errors

Mitigation of pressure increase, the basic nature of this transient, is accomplished by the reactor protection system functions. Turbine control valve trip scram and recirculation pump trip are designed to satisfy the single failure criterion. An evaluation of the most limiting single failure (i.e., failure of the bypass system) was considered in this event. Details of single failure analysis can be found in Appendix 15A.

15.2.2.3 Core and System Performance

15.2.2.3.1 Mathematical Model

The analyses performed for the LRNB event utilized the computer model described in Section 15.1.2.3.1.

15.2.2.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used by the reload fuel vendor are shown in Table 15.0-4.

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

The closure characteristics of the turbine control valves are assumed that all valves operate together and have a full stroke closure time, from fully open to fully closed, of 0.15 seconds (see subsection 15.2.2.3.4). The expected turbine control valve fast closure characteristics are shown in Table 15.2-17 and Figures 15.2-19a, b, and c. As shown in Table 15.0-4, the reload fuel vendor uses a turbine control valve stroke time of 0.15 seconds. The assumed turbine control valve flow versus position for current cycle analyses is shown in Figure 15.2-19d. Use of this characteristic and linear position versus time during turbine control valve fast closure is conservative relative to the Figure 15.2-19a, b, c characteristics.

Auxiliary power is independent of any turbine generator overspeed effects and is continuously supplied at rated frequency, assuming automatic fast transfer to auxiliary power supplies.

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

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

Events caused by low water level trips, including tripping of recirculation system pumps, and initiation of HPCS and RCIC core cooling system functions are not included in the simulation. Should these events occur, they will follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred, and are expected to be less severe than those already experienced by the system.

15.2.2.3.3 Results

15.2.2.3.3.1 Generator Load Rejection with Bypass

Figure 15.2-2 shows the results of the generator trip from rated power performed by the NSSS vendor. Peak neutron flux rises 105 percent of NB rated conditions.

The average surface heat flux shows no increase from its initial value and MCPR does not significantly decrease below its initial value.

15.2.2.3.3.2 Generator Load Rejection w/o Bypass

Figure 15.2-3B present responses of various reactor and plant parameters to the subject transient at 100% power and 105% flow corresponding to the Middle of Cycle and End of Cycle. For the current cycle, the peak neutron flux is limited to 175% of rated by the reactor scram and the peak fuel surface heat flux does not exceed 103% of its initial value. For a given initial power/flow condition, the ΔCPRs show an increase with exposure. For a given exposure, the ΔCPRs generally are higher for a lower value of initial power. The MCPR operating limit specified in the COLR are exposure dependent.

It is not anticipated that any single active component failure, in addition to failures of the direct trip scram, recirculation pump trip, and the bypass system, would significantly increase the severity of this event due to its brief duration.

15.2.2.3.4 Consideration of Uncertainties

All systems utilized for protection in this event were assumed to have the poorest allowable response (e.g., relief set points, scram stroke time and work characteristics). Expected plant behavior is, therefore, expected to reduce the actual severity of the transient.

15.2.2.4 Barrier Performance

15.2.2.4.1 Generator Load Rejection with Bypass

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

15.2.2.4.2 Generator Load Rejection w/o Bypass

Peak dome pressure reaches 1211 psig for this event at 100% power and 105% flow. The peak nuclear system pressure reaches 1234 psig at the bottom of the vessel. The peak vessel pressure and peak RPV dome pressure do not approach the barrier transient pressure limits of 1375 psig and 1325 psig, respectively.

15.2.2.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in subsection 15.2.4.5. Therefore, the radiological exposures noted in subsection 15.2.4.5 cover the consequences of this event.

15.2.2.6 Initial Cycle Generator Load Rejection w/o Bypass

15.2.2.6.1 Identification of Causes and Frequency Classification

The potential causes and the frequency classification of the load rejection event have not changed from cycle to cycle.

15.2.2.6.2 Sequence of Events and Systems Operation

[HISTORICAL INFORMATION] [For the initial cycle the NSSS vendor found that a Generator Load Rejection w/o Bypass event produces the sequence of events listed in Table 15.2-3b. Systems operation is the same as described in Section 15.2.2.2.]

15.2.2.6.3 Core and System Performance

The computer model described in subsection 15.1.2.6.3.1 was used to simulate this event. The NSSS vendor's analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15.0-3. Systems performance is the same as described for the current cycle.

15.2.2.6.3.1 Results (Initial Cycle)

[HISTORICAL INFORMATION] [For the initial cycle, Figure 15.2-3 shows that, for the case of bypass failure, peak neutron flux reaches about 149 percent of rated, average surface heat flux does not exceed 101 percent of its initial value. MCPR stays above the safety limit for this event.

In response to an NRC question, results were provided from a study performed for a generic BWR/6 without taking credit for non-seismically qualified equipment or any equipment contained in a non-seismic structure. The generator load rejection transient with concurrent failures of direct scram, RPT function, and bypass function was evaluated. The results are as follows and are shown in Figure 15.2-3a.

Maximum vessel pressure (psig)	1264
Time of maximum pressure (seconds)	1.9
Minimum critical power ratio (MCPR)	0.86
Time of MCPR (seconds)	1.2
Rods in boiling transient (%)	7.0
Peak cladding temperature (˚F)	<1220˚F
Peak value of fuel average temperature (˚F)	1599˚F

As these results are generic, the conclusion that no fuel damage will occur is applicable to Grand Gulf and therefore a plant specific analysis would be of little value.

If the above transient were analyzed with a direct trip scram, the results would be bounded by the flux scram trip presented here.

It is not anticipated that any single active component failure, in addition to failures of the direct trip scram, recirculation pump trip, and the bypass system, would significantly increase the severity of this event due to its brief duration.

The NSSS vendor concluded that, by combining the peak clad temperature shown above with the conclusions reached in Reference 6, there will be no calculated fuel failures. This is based on experimental evidence and calculational studies given in the referenced document for conditions similar to those used in the BWR/6 analysis.]

15.2.2.6.3.2 Consideration of Uncertainties

[HISTORICAL INFORMATION] [For the initial cycle, all systems utilized for protection in this event were assumed by the NSSS vendor to have the poorest allowable response (e.g., relief set points, scram stroke time and work characteristics). Anticipated plant behavior is, therefore, expected to reduce the actual severity of the transient.

Sensitivity studies show that the most severe initial condition for this transient occurs when the reactor is operating at 105 percent of the initially licensed NBR steam flow with the assumption of full arc operation, since the pressurization rate is higher at higher initial power level. Other sensitivity studies show that turbine control valve closure times smaller than the assumed 0.15 second do not result in unacceptable increase in CPR and reactor peak pressure. Since this transient is not the most limiting transient in determining the operating CPR limit, the turbine control valve closure time will not reflect the operating CPR limit.]

15.2.2.6.4 Barrier Performance (Initial Cycle)

[HISTORICAL INFORMATION] [For the initial cycle, the generator load rejection w/o bypass event calculated peak vessel pressure and peak RPV dome pressures do not approach the barrier transient limits of 1375 psig and 1325 psig, respectively.]

15.2.2.6.5 Radiological Consequences (Initial Cycle)

The radiological consequences for the turbine trip w/o bypass for the initial cycle are the same as for the current analysis, refer to subsection 15.2.2.5.

15.2.3 Turbine Trip

There are two scenarios evaluated for the turbine trip event; one with bypass, and the other without (w/o) bypass. The reload fuel vendor has determined that the turbine trip no bypass (TTNB) event is a limiting event which requires analysis for the current reload cycle. This subsection describes both analyses; 1) the analysis performed by the NSSS vendor for the turbine trip with bypass event for the initial fuel cycle which remains the current analysis for this event, and 2) the analysis performed by the reload fuel vendor for the TTNB event for the current fuel cycle. For a historical description of the initial fuel cycle analysis of the TTNB event, refer to subsection 15.2.3.6. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.2.3.1 Identification of Causes and Frequency Classification

15.2.3.1.1 Identification of Causes

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

15.2.3.1.2 Frequency Classification

15.2.3.1.2.1 Turbine Trip with Bypass

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

15.2.3.1.2.2 Turbine Trip w/o Bypass

This transient disturbance is categorized as an incident of moderate frequency. Frequency is expected to be as follows:

Frequency:	0.0059/plant year
MTBE:	156 years

Frequency Basis: As discussed in the subsection generator load rejection w/o bypass the failure rate of the bypass is 0.0048. Combining this with the turbine trip frequency of 1.22 events/ plant year yields the frequency of 0.0059/plant year.

15.2.3.2 Sequence of Events and Systems Operation

15.2.3.2.1 Sequence of Events

15.2.3.2.1.1 Turbine Trip with Bypass

Turbine trip at high power produces the sequence of events listed in Table 15.2-4.

15.2.3.2.1.2 Turbine Trip w/o Bypass

Turbine trip at high power w/o bypass produces the sequence of events listed in Table 15.2-5A.

15.2.3.2.1.3 Deleted

15.2.3.2.2 Systems Operation

15.2.3.2.2.1 Turbine Trip with Bypass

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

Turbine stop valve closure initiates a reactor scram trip and recirculation pump trip via turbine stop valve trip fluid pressure signals for power levels greater than 35.4 percent NBR. Credit is taken for successful operation of the reactor protection system.

Turbine stop valve closure initiates EOC - RPT logic and trips the Recirculation Pump from fast speed.

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

15.2.3.2.2.2 Turbine Trip w/o Bypass

Same as subsection 15.2.3.2.2.1 except that failure of the main turbine bypass system is assumed for the entire transient time period analyzed.

15.2.3.2.2.3 Turbine Trip at Low Power w/o Bypass

Same as subsection 15.2.3.2.2.1 except that failure of the main turbine bypass system is assumed.

It should be noted that below 35.4 percent NB rated power level, a main stop valve scram trip inhibit signal derived from the power range neutron monitoring system is assumed to be activated. This is done to eliminate the stop valve scram trip signal from scramming the reactor provided the bypass system functions properly. In other words, the bypass would be sufficient at this low power to accommodate a turbine trip without the necessity of shutting down the reactor. All other protection system functions remain functional as before and credit is taken for those protection system trips.

15.2.3.2.3 The Effect of Single Failures and Operator Errors

15.2.3.2.3.1 Turbine Trips at Power Levels Greater Than 35.4 Percent

Mitigation of pressure increase, the basic nature of this transient, is accomplished by the reactor protection system functions. Main stop valve closure scram trip and RPT are designed to satisfy single failure criterion.

15.2.3.2.3.2 Turbine Trips at Power Levels Less Than 35.4 Percent NBR

Same as subsection 15.2.3.2.3.1 except recirculation pump trip and stop valve closure scram trip is normally inoperative. Since protection is still provided by high flux, high pressure, etc., these will also continue to function and scram the reactor should a single failure occur.

15.2.3.3 Core and System Performance

15.2.3.3.1 Mathematical Model

The computer model for the turbine trip without bypass, described in subsection 15.1.2.3.1, was used to simulate these events.

15.2.3.3.2 Input Parameters and Initial Conditions

The current cycle analyses have been performed with plant conditions tabulated in Table 15.0-4.

Turbine stop valves full stroke closure time is 0.10 second.

The expected turbine stop valve closure characteristics are shown in Table 15.2-17 and Figures 15.2-20a and b. The assumed turbine stop valve flow versus position evaluated in the current cycle analyses is shown in Figure 15.2-20c. Use of this characteristic and linear position versus time during turbine stop valve closure is conservative relative to the expected characteristic curve.

A reactor scram is initiated when the stop valves trip fluid pressure decays, and the signal is present before the stop valves start to close. This signal originates from pressure transmitters and trip units which sense hydraulic trip fluid pressure decay which is indicative of stop valve motion away from fully open.

This stop valve scram trip signal is assumed to be automatically bypassed when the reactor is below 35.4 percent NB rated power level.

Reduction in core recirculation flow is initiated by the trip units associated with the main stop valves, which actuate trip circuitry which trips the recirculation pumps.

15.2.3.3.3 Results

15.2.3.3.3.1 Turbine Trip with Bypass

A turbine trip with the bypass system operating normally is simulated at 105 percent of the initially licensed NB rated steam flow conditions in Figure 15.2-4 for the initial cycle.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited to 111 percent of rated by the stop valve scram and the RPT system. Peak fuel surface heat flux does not exceed its initial value. MCPR for the transient does not change significantly.

15.2.3.3.3.2 Turbine Trip w/o Bypass

The results for a turbine trip w/o bypass at 100% power and 105% rated steam flow are presented in Reference 16. The peak neutron flux is limited to 162% of rated by the reactor scram and the peak fuel surface heat flux does not exceed 101% of its initial value. The MCPR for this transient remains above the safety limit for incidents of moderate frequency and, therefore, the design basis is satisfied.

15.2.3.3.3.3 Turbine Trip w/o Bypass, Low Power

Below 35.4 percent of rated power, the turbine stop valve closure and turbine control valve closure scrams are assumed to be automatically bypassed. At these lower power levels, the power range neutron monitoring system is used to initiate the scram logic bypass. The scram which terminates the transient is initiated by high vessel pressure. The bypass valves are assumed to fail; therefore, system pressure will increase until the pressure relief set points are reached. At this time, because of the relatively low power of this transient event, relatively few relief valves will open to limit reactor pressure. Peak pressures are not expected to greatly exceed the pressure relief valve set points and will be significantly below the RCPB transient limit of 1375 psig. Peak surface heat flux and peak fuel center temperature remain at relatively low values and MCPR is expected to remain well above the GETAB safety limit.

15.2.3.3.4 Considerations of Uncertainties

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

a. Slowest allowable control rod scram motion is assumed.

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

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

d. Set points of the safety/relief valves include errors (high) for all valves.

15.2.3.4 Barrier Performance

15.2.3.4.1 Turbine Trip with Bypass

For the initial cycle, peak pressure in the bottom of the vessel reaches 1161 psig which is below the ASME Code limit of 1375 psig for the reactor cooling pressure boundary. Vessel dome pressure does not exceed 1154 psig. The severity of turbine trips from lower initial power levels decreases to the point where a scram can be avoided if auxiliary power is available from an external source and the power level is within the bypass capability.

15.2.3.4.2 Turbine Trip w/o Bypass

The safety/relief valves open and close sequentially as the stored energy is dissipated and the pressure falls below the set points of the valves. Peak nuclear system pressure reaches 1231 psig for this event at 100% power and 105% flow. Peak dome pressure reaches 1209 psig. The peak vessel pressure and peak RPV dome pressure do not approach the barrier transient pressure limits of 1375 psig and 1325 psig, respectively.

15.2.3.4.2.1 Turbine Trip w/o Bypass at Low Power

Qualitative discussion is provided in subsection 15.2.3.3.3.3.

15.2.3.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in subsection 15.2.4.5. Therefore, the radiological exposures noted in subsection 15.2.4.5 cover the consequences of this event.

15.2.3.6 Turbine Trip w/o Bypass (Initial Cycle)

[HISTORICAL INFORMATION] [The reload fuel vendor has determined that the turbine trip event is a limiting event which requires analysis for the current reload cycle. This subsection describes the analysis performed by the NSSS vendor for the initial fuel cycle. For a description of the current fuel cycle analysis of this event, refer to subsection 15.2.3. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.]

15.2.3.6.1 Identification of Causes and Frequency Classification (Initial Cycle)

The potential causes of this event are the same as for the current cycle. The probability of occurrence of this event is also the same as that for the current cycle. Refer to subsection 15.2.3.1.

15.2.3.6.2 Sequence of Events and Systems Operation (Initial Cycle)

[HISTORICAL INFORMATION] [For the initial cycle the NSSS vendor found that a turbine trip w/o bypass produces a sequence of events listed in Table 15.2-5. Systems operation is the same as "with bypass" except that failure of the main turbine bypass system is assumed for the entire transient time period analyzed.]

15.2.3.6.3 Core and System Performance (Initial Cycle)

[HISTORICAL INFORMATION] [The computer model described in subsection 15.1.1.6.3.1 was used to simulate the turbine trip w/o bypass event. The initial analysis has been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-3. System performance is the same as described for the current cycle.]

15.2.3.6.3.1 Results (Initial Cycle)

[HISTORICAL INFORMATION] [The results for a turbine trip w/o bypass simulated at 105% of the initially licensed NB rated steam flow conditions are shown in Figure 15.2-5 for the initial cycle. Peak neutron fluence reaches 105% of its rated value, and average surface heat flux does not exceed its initial value.]

15.2.3.6.3.2 Consideration of Uncertainties

The consideration of uncertainties for the initial cycle are the same as for the current analysis, refer to subsection 15.2.3.4.

15.2.3.6.4 Barrier Performance (Initial Cycle)

[HISTORICAL INFORMATION] [For the initial cycle, the turbine trip w/o bypass event calculated peak vessel pressure and peak RPV dome pressures did not approach the barrier transient limits of 1375 psig and 1325 psig, respectively.]

15.2.3.6.5 Radiological Consequences (Initial Cycle)

The radiological consequences for the turbine trip w/o bypass for the initial cycle are the same as for the current analysis, refer to subsection 15.2.3.5.

15.2.4 MSIV Closures

The reload fuel vendor has determined that the MSIV closures event is not a limiting event for the current reload cycle. Therefore, this subsection describes the original analysis performed by the NSSS vendor for the initial cycle which remains the current licensing basis for GGNS. The radiological consequences represent the calculation of record following analyses associated with the alternative source term and EPU. The MSIV closure event, with a flux scram only, is discussed in section 5.2.2. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.2.4.1 Identification of Causes and Frequency Classification

15.2.4.1.1 Identification of Causes

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

15.2.4.1.2 Frequency Classification

15.2.4.1.2.1 Closure of All Main Steam Isolation Valves

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

15.2.4.1.2.2 Closure of One Main Steam Isolation Valve

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

15.2.4.2 Sequence of Events and Systems Operation

15.2.4.2.1 Sequence of Events

Table 15.2-6 lists the sequence of events for Figure 15.2-6.

15.2.4.2.1.1 Deleted

15.2.4.2.2 Systems Operation

15.2.4.2.2.1 Closure of All Main Steam Isolation Valves

MSIV closures initiate a reactor scram trip via position signals to the protection system. Credit is taken for successful operation of the protection system. For an MSIV closure when credit is taken only for an indirect derived scram, i.e., flux scram, refer to the discussion in section 5.2.2.

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

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

15.2.4.2.2.2 Closure of One Main Steam Isolation Valve

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

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

15.2.4.2.3 The Effect of Single Failures and Operator Errors

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

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

15.2.4.3 Core and System Performance

15.2.4.3.1 Mathematical Model

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

15.2.4.3.2 Input Parameters and Initial Conditions

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

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

Position switches on the valves initiate a reactor scram when the valves are less than 90 percent open. Closure of these valves inhibits steam flow to the feedwater turbines terminating feedwater flow.

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

15.2.4.3.3 Results

The effects of this event, with respect to core and system performance, are considered to be bounded by the Generator Load Rejection without bypass analysis (Section 15.2.2) which is evaluated each cycle.

15.2.4.3.3.1 Closure of All Main Steam Isolation Valves

Figure 15.2-6 shows the changes in important nuclear system variables for the simultaneous isolation of all main steam lines while the reactor is operating at 105 percent of the initially licensed NB rated steam flow. Neutron flux and fuel surface heat flux show no increase.

15.2.4.3.3.2 Closure of One Main Steam Isolation Valve

Only one isolation valve is permitted to be closed at a time for testing purposes to prevent scram. Normal test procedure requires an initial power reduction to less than 75 percent of design conditions in order to avoid high flux scram, high pressure scram, or full isolation from high steam flow in the "live" lines. With a 3-second closure of one main steam isolation valve during 105 percent of the initially licensed rated power conditions, the steam flow disturbance raises vessel pressure and reactor power enough to initiate a high neutron flux scram. This transient is considerably milder than the full power case. No quantitative analysis is furnished for this event. However, no significant change in thermal margins is experienced and no fuel damage occurs. Peak pressure remains below S/R valve set points.

Inadvertent closure of one or all of the isolation valves while the reactor is shut down (such as operating state C, as defined in Appendix 15A) will produce no significant transient. Closures during plant heatup (operating state D) will be less severe than the maximum power cases (maximum stored and decay heat) discussed in subsection 15.2.4.3.3.1.

15.2.4.3.4 Considerations of Uncertainties

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

a. Slowest allowable control rod scram motion is assumed.

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

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

d. Maximum specified set points of the safety/relief valves are assumed. Usually, they are ~1-2 percent higher than the nominal set points.

15.2.4.4 Barrier Performance

15.2.4.4.1 Closure of All Main Steam Isolation Valves

The nuclear system relief valves begin to open at approximately 3.1 seconds after the start of isolation. The valves close sequentially as the stored heat is dissipated but continue to discharge the decay heat intermittently. Peak pressure at the vessel bottom reaches 1213 psig, clearly below the pressure limits of the reactor coolant pressure boundary. Peak pressure in the main steam line is 1179 psig.

15.2.4.4.2 Closure of One Main Steam Isolation Valve

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

15.2.4.5 Radiological Consequences

15.2.4.5.1 Fission Product Release to the Environment

Although it is assumed that at the time of MSIV closure the containment is being purged at a rate of 6000 cfm. Automatic containment isolation is conservatively neglected. No significant amount of radioactivity is released to the environment as presented in Table 15.2-15.

15.2.4.5.2 Results

Dispersion data and the calculated exposures are presented in Table 15.2-16.

15.2.5 Loss of Condenser Vacuum

The reload fuel vendor has determined that the loss of condenser vacuum event is not a limiting event for the current reload cycle. Therefore, this subsection describes the original analysis performed by the NSSS vendor for the initial cycle which remains the current licensing basis for GGNS. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.2.5.1 Identification of Causes and Frequency Classification

15.2.5.1.1 Identification of Causes

Various system malfunctions which can cause loss of condenser vacuum due to some equipment failure are designated in Table 15.2-7.

15.2.5.1.2 Frequency Classification

This event is categorized as an incident of moderate frequency.

15.2.5.2 Sequence of Events and Systems Operation

15.2.5.2.1 Sequence of Events

Table 15.2-8 lists the sequence of events for Figure 15.2-7.

15.2.5.2.1.1 Deleted

15.2.5.2.2 Systems Operation

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

Tripping functions incurred by sensing main turbine condenser vacuum pressure are designated in Table 15.2-9.

15.2.5.2.3 The Effect of Single Failures and Operator Errors

This event does not lead to a general increase in reactor power level. Mitigation of power increase is accomplished by the protection system initiation of scram.

Failure of the integrity of the condenser offgas treatment system is considered to be an accident situation and is described in subsection 15.7.1.

Single failures will not affect the vacuum monitoring and turbine trip devices which are redundant. The protective sequences of the anticipated operational transient are shown to be single failure proof. See Appendix 15A for details.

15.2.5.3 Core and System Performance

The effects of this event, with respect to core and system performance, are considered to be bounded by the Generator Load Rejection without bypass analysis (Section 15.2.2) which is evaluated each cycle.

15.2.5.3.1 Mathematical Model

The computer model described in subsection 15.1.1.6.3.1 was used to simulate this transient event.

15.2.5.3.2 Input Parameters and Initial Conditions

This analysis was performed with plant conditions tabulated in Table 15.0-2 unless otherwise noted.

Turbine stop valves full stroke closure time is assumed to be 0.1 second for this analysis.

In the plant, the Reactor Protection System detection of turbine stop valve closure is based on electro-hydraulic control system trip fluid pressure. Reactor protection system will receive and process the closure signal before the turbine stop valve is 10% closed.

For modeling purposes, a reactor scram is initiated when the valves are less than 90 percent open. This stop valve scram trip signal is automatically bypassed when the reactor is below 35.4 percent NB rated power level.

15.2.5.3.3 Results

The analysis presented here is a hypothetical case using a conservative assumption that the vacuum decays at an average rate of 10 inches Hg per second. Since the bypass system is signaled to close at a vacuum level of about 10 inches Hg less than the stop valve closure, it is available for only 1 second during this transient event.

The initial part of this transient, therefore, is similar to a normal turbine trip with bypass. From 1 second after initiation of the turbine trip, it is similar to a “turbine trip without bypass” transient. The effect of main steam line isolation valve closure tends to be minimal since the closure of main turbine stop valves and subsequently the bypass valves have already shut off the main steam line flow. Figure 15.2-7 shows the transient expected for this event. It is assumed that the plant is initially operating at 105 percent of the initially licensed nuclear boiler rated steam flow conditions. Peak neutron flux and average fuel surface heat flux do not increase. Safety/relief valves open to limit the pressure rise, then sequentially reclose as the stored energy is dissipated.

15.2.5.3.4 Considerations of Uncertainties

The reduction or loss of vacuum in the main turbine condenser will sequentially trip the main and feedwater turbines and close the main steam line isolation valves and bypass valves. While these are the major events occurring, other resultant actions will include scram (from stop valve closure) and bypass opening with the main turbine trip. Because the protective actions are actuated at various levels of condenser vacuum, the severity of the resulting transient is dependent upon the rate at which the vacuum pressure is lost. Other operational problems which could cause a loss of condenser vacuum, such as failure of the steam jet air ejector, produce a slower rate of vacuum loss (affects take minutes, not seconds) than loss of circulating water flow. See Table 15.2-7. If corrective actions by the reactor operators are not successful, then sequential trips of the main and feedwater turbines will occur, and ultimately complete isolation by closing the bypass valves (opened with the main turbine trip) and the MSIVs will occur.

The faster the rate of loss of the condenser vacuum, the lower the overall effectiveness of the bypass valves since they would be closed more quickly. In these cases, the event is bounded by the turbine trip transient without bypass.

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

a. Slowest allowable control rod scram motion is assumed.

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

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

d. Set points of the safety/relief valves are assumed to be at the upper limit of Technical Specifications for all valves.

15.2.5.4 Barrier Performance

As shown in Figure 15.2-7, the consequences of this event do not result in any temperature, or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.5.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of safety relief valve actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in subsection 15.2.4.5. Therefore, the radiological exposures noted in subsection 15.2.4.5 cover the consequences of this event.

15.2.6 Loss of AC Power

The reload fuel vendor has determined that the loss of AC power event is not a limiting event for the current reload cycle. Therefore, this subsection describes the original analysis performed by the NSSS vendor for the initial cycle which remains the current licensing basis for GGNS. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle,refer to Section 15.0.

15.2.6.1 Identification of Causes and Frequency Classification

15.2.6.1.1 Identification of Causes

15.2.6.1.1.1 Loss of Service Transformer

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

15.2.6.1.1.2 Loss of All Grid Connections

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

15.2.6.1.2 Frequency Classification

15.2.6.1.2.1 Loss of Service Transformer

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

15.2.6.1.2.2 Loss of All Grid Connections

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

15.2.6.2 Sequence of Events and Systems Operation

15.2.6.2.1 Sequence of Events

15.2.6.2.1.1 Loss of Service Transformer

Table 15.2.10 lists the sequence of events for Figure 15.2-8.

15.2.6.2.1.2 Loss of All Grid Connections

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

15.2.6.2.1.3 Deleted

15.2.6.2.2 Systems Operation

15.2.6.2.2.1 Loss of Service Transformer

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

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

a. All pumps are tripped at a reference time, t=0, with normal coastdown times for the recirculation pumps.

b. Within 2 seconds, the loss of power to the scram solenoid valves and the MSIVs causes a reactor scram and MSIV closure. The feedwater turbine trip occurs in about 4 seconds due to MSIV closure.

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

15.2.6.2.2.2 Loss of All Grid Connections

Same as subsection 15.2.6.2.2.1 with the following additional concern.

The loss of all grid connections is another feasible, although improbable, way to lose all auxiliary power. This event would add a generator load rejection to the above sequence at time, t=0. The load rejection immediately forces the turbine control valves closed, causes a turbine trip, causes a scram, and initiates recirculation pump trip (RPT) (already tripped at reference time t=0).

15.2.6.2.3 The Effect of Single Failures and Operator Errors

Loss of the service transformer in general leads to a reduction in power level due to rapid pump coastdown with pressurization effects due to turbine trip occurrence. Additional failures of the other systems assumed to protect the reactor would not result in an effect different from those reported. Failures of the protection systems have been considered and satisfy single failure criteria and as such no change in analyzed consequences is expected. See Appendix 15A for details on single failure analysis.

15.2.6.3 Core and System Performance

The effects of this event, with respect to core and system performance, are considered to be bounded by the Generator Load Rejection without bypass analysis (Section 15.2.2) which is evaluated each cycle.

15.2.6.3.1 Mathematical Model

The computer model described in subsection 15.1.1.6.3.1 was used to simulate this event.

Operation of the RCIC or HPCS systems is not included in the simulation of this transient, since startup of these pumps does not permit flow in the time period of this simulation.

15.2.6.3.2 Input Parameters and Initial Conditions

15.2.6.3.2.1 Loss of Service Transformer

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2 and under the assumed systems constraints described in subsection 15.2.6.2.2.

15.2.6.3.2.2 Loss of All Grid Connections

Same as subsection 15.2.6.3.2.1

15.2.6.3.3 Results

15.2.6.3.3.1 Loss of Service Transformer

Figure 15.2-8 shows graphically the simulated transient. The initial portion of the transient is similar to the two pump trip transient. Within two seconds, reactor scram and main steam line isolation valve closure occurs.

Sensed level drops to the RCIC and HPCS initiation set point at approximately 22 seconds after loss of auxiliary power.

There is no significant increase in fuel temperature or decrease in the operating MCPR value of 1.18. Fuel thermal margins are not threatened and the design basis is satisfied.

15.2.6.3.3.2 Loss of All Grid Connections

Loss of all grid connections is a more general form of loss of auxiliary power. It essentially takes on the characteristic response of the standard full load rejection discussed in subsection 15.2.2. Figure 15.2-9 shows graphically the simulated event.

15.2.6.3.4 Consideration of Uncertainties

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

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

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

Following main steam line isolation and RHR initiation the reactor pressure is expected to increase until the safety/ relief valve set point is reached. At this time the valves operate in a cyclic manner to discharge the decay heat to the suppression pool.

15.2.6.4 Barrier Performance

15.2.6.4.1 Loss of Service Transformer

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

15.2.6.4.2 Loss of All Grid Connections

Safety/relief valves open in the pressure relief mode of operation as the pressure increases beyond their set points. The pressure in the dome is limited to a maximum value of 1156 psig, well below the vessel pressure limit of 1375 psig.

15.2.6.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of safety relief valve actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in subsection 15.2.4.5. Therefore, the radiological exposures noted in subsection 15.2.4.5 cover the consequences of this event.

15.2.7 Loss of Feedwater Flow

The reload fuel vendor has determined that the loss of feedwater flow event is not a limiting event for the current reload cycle. However, this event has been re-analyzed by GEH due to the higher decay heat associated with EPU. Therefore, this subsection describes the current analysis performed by GEH for EPU. For a historical description of the initial fuel cycle analysis of the loss of feedwater flow event by the NSSS vendor, refer to subsection 15.2.7.6. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.2.7.1 Identification of Causes and Frequency Classification

15.2.7.1.1 Identification of Causes

A loss of feedwater flow could occur from pump failures, valve malfunction, or a loss of off-site power.

15.2.7.1.2 Frequency Classification

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

15.2.7.2 Sequence of Events and Systems Operation

15.2.7.2.1 Sequence of Events

The following is the general sequence of events in the analysis. The reactor is assumed to be at 102% of the EPU power level when the loss of feedwater occurs. The initial level in the model is conservatively set at the low-level scram setpoint and reactor feedwater is instantaneously isolated at event initiation. Scram is initiated at the start of the event. The RCIC system is initiated when the level decreases to the low-low level. The MSIV initiates when the level decreases to low-low-low level. The RCIC flow to the vessel begins at 60 seconds into the event, minimum level is reached at 622 seconds and level is recovered after that point.

15.2.7.2.2 Systems Operation

Extra decay heat due to EPU resulted in slightly more time being required for the automatic systems to restore water level than was in the original analysis by the NSSS vendor for the initial core. Operator action is only needed for a long-term plant shutdown. The results of the analysis shows that the minimum water level inside the shroud is 50 inches above the top of active fuel region at EPU conditions. After the water level is restored, the operator manually controls the water level, reduces reactor pressure, and initiates RHR shutdown cooling. This sequence of events does not require any new operator actions or shorter response times than that assumed in the original analysis by the NSSS vendor for the initial core. See additional detail in subsection 15.2.7.6.2.1.

One other operational requirement is that the RCIC system restores the reactor water level while avoiding automatic depressurization system timer initiation and MSIV activation functions associated with the low-low-low reactor water level setpoint (Level 1). This requirement is intended to avoid unnecessary initiations of safety systems. The requirement is not a safety-related function. The analysis results show the nominal Level 1 setpoint trip is avoided.

15.2.7.2.3 The Effect of Single Failures and Operator Errors

The analysis assumed failure of the HPCS system and used only RCIC system to recover the reactor water level. There are no additional failures assumed.

15.2.7.3 Core and System Performance

15.2.7.3.1 Mathematical Model

The computer code used for the loss of feedwater flow event is the SAFER code as described in the PUSAR which is the same code as used in the DBA LOCA analysis for the ECCS.

15.2.7.3.2 Input Parameters and Initial Conditions

Input parameters and initial conditions are described in the PUSAR discussions on the DBA LOCA analysis. One other key assumption for this analysis is the assumed decay heat level of ANSI 5.1-1979 with a two-sigma uncertainty. The assumed decay heat level for the EPU analysis was ANSI 5.1-1979 decay heat +10%, which bounds the + two sigma. Thus, the key analytical assumptions were the same or conservative relative to the previous licensing basis.

15.2.7.3.3 Results

The loss of feedwater flow event was analyzed using the SAFER code to demonstrate acceptable RCIC performance. The design basis criterion for GGNS was confirmed by demonstrating that the RCIC is capable of maintaining the water level inside the shroud above the top of active fuel during the entire transient. As shown in Figure 15.2-10a, the minimum level is maintained at least 50 inches above the top of active fuel, thereby demonstrating acceptable RCIC system performance. There were no applicable equipment out of service assumptions for this transient. The results of the analysis demonstrate that the reactor protection and safety systems will continue to ensure that the SAFDLs and the reactor coolant pressure boundary pressure limits will not be exceeded as a result of this event.

The loss of one feedwater pump event was also looked at as part of EPU implementation. This analysis of this event only addressed operational considerations to avoid reactor scram on low reactor water level (Level 3). This requirement is intended to avoid unnecessary reactor shutdowns. Because the MELLLA region is extended along the existing upper boundary to the EPU rated thermal power, there is no increase in the highest flow control line for GGNS.

15.2.7.3.4 Considerations of Uncertainties

End-of-cycle scram characteristics are assumed.

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

Operation of the RCIC or HPCS systems is not included in the simulation of the first 50 seconds of this transient since startup of these pumps occurs in the latter part of this time period and therefore these systems have no significant effects on the results of this transient except perhaps as discussed in subsection 15.2.7.2.3.

15.2.7.4 Barrier Performance

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

15.2.7.5 Radiological Consequences

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

15.2.7.6 Loss of Feedwater Flow (Initial Cycle)

[HISTORICAL INFORMATION] [The reload fuel vendor has determined that the loss of feedwater flow event is not a limiting event for the current reload cycle. However, this event has been re-analyzed by GEH due to the higher decay heat associated with EPU. This subsection describes the analysis performed by the NSSS vendor for the initial fuel cycle. For a description of the current fuel cycle analysis of this event, refer to subsection 15.2.7. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.2.7.6.1 Identification of Causes and Frequency Classification

15.2.7.6.1.1 Identification of Causes

[HISTORICAL INFORMATION] [A loss of feedwater flow could occur from pump failures, feedwater controller failures, operator errors, or reactor system variables such as high vessel water level (L9) trip signal.]

15.2.7.6.1.2 Frequency Classification

[HISTORICAL INFORMATION] [This transient disturbance is categorized as an incident of moderate frequency.]

15.2.7.6.2 Sequence of Events and Systems Operation (Initial Cycle)

15.2.7.6.2.1 Sequence of Events

[HISTORICAL INFORMATION] [Table 15.2-12 lists the sequence of events for Figure 15.2-10.]

15.2.7.6.2.1.1 Deleted

15.2.7.6.2.2 Systems Operation

[HISTORICAL INFORMATION] [Loss of feedwater flow results in a proportional reduction of vessel inventory causing the vessel water level to drop. The first corrective action is the low level (L3) scram trip actuation. Reactor protection system responds within 1 second after this trip to scram the reactor. The low level (L3) scram trip function meets single failure criterion.

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

15.2.7.6.2.3 The Effect of Single Failures and Operator Errors

[HISTORICAL INFORMATION] [The nature of this event, as explained above, results in a lowering of vessel water level. Key corrective efforts to shut down the reactor are automatic and designed to satisfy single failure criterion; therefore, any additional failure in these shutdown methods would not aggravate or change the simulated transient. See Appendix 15A for details.]

15.2.7.6.3 Core and System Performance

15.2.7.6.3.1 Mathematical Model

[HISTORICAL INFORMATION] [The computer model described in subsection 15.1.1.6.3.1 was used to simulate this event.]

15.2.7.6.3.2 Input Parameters and Initial Conditions

[HISTORICAL INFORMATION] [These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

The loss of the feedwater flow transient was analyzed at the end of equilibrium cycle with a very conservative dynamic void reactivity coefficient. Since the dynamic void reactivity coefficient is the major factor which determines the severity of the loss of feedwater flow transient, it is more conservative to analyze the transient at the end of the equilibrium cycle than at the beginning-of-cycle when a less negative void coefficient occurs.]

15.2.7.6.3.3 Results

[HISTORICAL INFORMATION] [The results of this transient simulation are shown in Figure 15.2-10. Feedwater flow terminates at approximately 5 seconds. Subcooling decreases causing a reduction in core power level and pressure. As power level is lowered, the turbine steam flow starts to drop off because the pressure regulator is attempting to maintain pressure for the first 5 seconds or so. Water level continues to drop until the vessel level (L3) scram trip set point is reached whereupon the reactor is shut down. Vessel water level continues to drop to the L2 trip. At this time, the recirculation system is tripped and HPCS and RCIC operation is initiated. MCPR remains considerably above the safety limit since increases in heat flux are not experienced.]

15.2.7.6.3.4 Considerations of Uncertainties

[HISTORICAL INFORMATION] [End-of-cycle scram characteristics are assumed.

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

Operation of the RCIC or HPCS systems is not included in the simulation of the first 50 seconds of this transient since startup of these pumps occurs in the latter part of this time period and therefore these systems have no significant effects on the results of this transient except perhaps as discussed in subsection 15.2.7.6.2.3.]

15.2.7.6.4 Barrier Performance

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

15.2.7.6.5 Radiological Consequences

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

15.2.8 Feedwater Line Break

(Refer to subsection 15.6.6.)

15.2.9 Failure of RHR Shutdown Cooling

The reload fuel vendor has determined that the failure of RHR shutdown cooling event is not a limiting event for the current reload cycle. Therefore, this subsection describes, in general, the original analysis performed by the NSSS vendor for the initial cycle which includes Activities A and B (Figure 15.2-12) for plant cooldown from full power to a RPV pressure of approximately 100 psig, as well as, several scenarios of Activity C for plant cooldown from 100 psig to cold shutdown. These descriptions of the initial cycle analysis remain part of the current licensing basis for GGNS. However, the long term DBA LOCA event was re-analyzed during EPU implementation and as a result of the concern discussed in GEH Safety Communication 06-01 (Ref. 14). In the conclusions of the re-analysis, GEH determined that the alternate shutdown cooling (ASDC) event, specifically Activity C2(a), is the limiting event for determination of the peak temperatures reached in the suppression pool. Therefore, this subsection includes a description of the current re-analysis for the ASDC (Activity C2(a)) event performed by GEH to support plant operation at EPU conditions in applicable subsections. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0. Normally, in evaluating component failure considerations associated with the RHRS - shutdown cooling mode operation, active pumps or instrumentation (all of which are redundant for safety system portions of the RHRS aspects) would be assumed to be the likely errant equipment. For purposes of worst case analysis, the single recirculation loop suction valve to the redundant RHRS loops is assumed to fail. This failure would, of course, still leave two complete RHRS loops for LPCI, pool, and containment cooling minus the normal RHRS - shutdown cooling loop connection. Although the errant valve could be manually manipulated open, it is assumed failed indefinitely. If it is now assumed that the SACF criteria is applied, the plant operator has one complete RHRS loop available with the further selective worst case assumption that the other RHRS loop is lost.

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

a. Loss of all offsite ac power.

b. Utilization of safe shutdown equipment only.

c. Operator involvement only after 10 minutes after coincident assumptions.

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

15.2.9.1 Identification of Causes and Frequency Classification

15.2.9.1.1 Identification of Causes

The plant is operating at 105 percent of the rated steam flow at licensed conditions (i.e. initially licensed NB rated flow for Activities A, B, C1 and C2(b) and EPU rated flow for ASDC Activity C2(a)) when a long-term loss of offsite power occurs, causing multiple safety/relief valve actuation (see subsection 15.2.6) and subsequent heatup of the suppression pool. Reactor vessel depressurization is initiated to bring the reactor pressure to approximately 100 psig. Concurrent with the loss of offsite power an additional (divisional) single failure occurs which prevents the operator from establishing the normal shutdown cooling path through the RHR shutdown cooling lines. He then establishes a shutdown cooling path for the vessel through the safety relief valves.

15.2.9.1.2 Frequency Classification

This event is evaluated as a moderate frequency event.

15.2.9.2 Sequence of Events and System Operation

15.2.9.2.1 Sequence of Events

The sequence of events for Activities A, B, C1 and C2(b) of this event is shown in Table 15.2-13.

The sequence of events for Activity C2(a) follows the same path and similarly, at 10 minutes into the event, the operators initiate supression pool cooling using one RHR heat exchanger on one loop utilizing one LPCI pump. Further assumptions for Activity C2(a) include the following (Ref. 15):

a. Operators initiate a controlled reactor cooldown at the rate of 100°F/hour when the suppression pool temperature gets about 110°F.

b. Operators manually shut down the suppression pool cooling mode at 35 hours in preparation for establishing ASDC.

c. RHR in LPCI mode is initiated at 35 hours and 35 minutes to flood the vessel above the main steam line elevation in order to provide liquid recirculation flow to the suppression pool via the automatic depressurization system valves.

15.2.9.2.1.1 Deleted

15.2.9.2.2 System Operation

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

15.2.9.2.3 The Effect of Single Failures and Operator Errors

The worst case single failure (loss of division power) has already been analyzed in this event. Therefore, no single failure or operator error can make the consequences of this event any worse. See Appendix 15A for a discussion of this subject.

15.2.9.3 Core and System Performance

15.2.9.3.1 Methods, Assumptions, and Conditions

An event that can directly cause reactor vessel water temperature increase is one in which the energy removal rate is less than the decay heat rate. The applicable event is loss of RHR shutdown cooling. For Activities A, B, C1 and C2(b) this event can occur during the low pressure portion of a normal reactor shutdown and cooldown, when the RHR system is operating in the shutdown cooling mode. For Activity C2(a), the event is initiated with a loss of off-site power which triggers reactor isolation and scram with vessel depressurization and steam release through the safety release valves. The earliest time the shutdown system can be actuated is ~1 hour after shutdown is initiated. During this time MCPR remains high and nucleate boiling heat transfer is not exceeded at any time. Therefore, the core thermal safety margin remains essentially unchanged. The 10-minute time period approximated for operator action is an estimate of how long it would take the operator to initiate the necessary actions; it is not a time by which he must initiate action.

15.2.9.3.2 Mathematical Model

In evaluating this event, the important parameters to consider are reactor blowdown rate and suppression pool temperature. For Activities A, B, C1 and C2(b), the models used for this evaluation are described in References 4 and 5.

For the ASDC Activity C2(a), the GEH computer code SHEX06A is used to perform the analysis of the longterm containment pressure and temperature responses. The code calculates the suppression pool bulk temperature, and the pressures and temperatures in the drywell and wetwell airspace. The use of the SHEX code has been accepted by the NRC for calculating the response of the containment during an accident or transient event and has been applied to the evaluation of containment responses for many BWR plants.

15.2.9.3.3 Input Parameters and Initial Conditions

Table 15.2-14 shows the input parameters and initial conditions used in evaluation of Activities A, B, C1 and C2(b) for this event except as noted below.

For the ASDC Activity C2(a), the input parameters and initial conditions for the long-term containment pressure and temperature responses to LOCAs are described in Reference 15 which includes the DBA LOCA as well as the ASDC analysis.

15.2.9.3.4 Results

For most single failures that could result in loss of shutdown cooling, no unique safety actions are required. In these cases, shutdown cooling is simply reestablished using other, normal shutdown cooling equipment. In cases where both of the RHRS shutdown cooling suction valves cannot be opened, alternate paths are available to accomplish the shutdown cooling function (Figure 15.2-11). An evaluation has been performed assuming the worst single failure that could disable the RHRS shutdown cooling valves.

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

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

The alternate shutdown systems are capable of performing the function of transferring heat from the reactor to the environment using only safety grade systems. The systems are capable of bringing the reactor to a cold shutdown in less than 3 hours after the transient occurs. Even if it is additionally postulated that all of the ADS or relief valves discharge piping also fails, the shutdown cooling function would eventually be accomplished as the cooling water would run directly out of the ADS or safety/relief valves, flooding into the drywell.

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

The design evaluation is divided into two phases: (1) full power operation to approximately 100 psig vessel pressure, and (2) approximately 100 psig vessel pressure to cold shutdown (200 F) conditions.

15.2.9.3.4.1 Full Power to Approximately 100 psig

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

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

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

15.2.9.3.4.2 Approximately 100 psig to Cold Shutdown

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

a. The vessel is at 100 psig and saturated conditions

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

c. There is no offsite power available

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

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

ADS (DC Division 1 and DC Division 2)

RHR Loop (A) (Division 1)

HPCS (Division 3)

RCIC (DC Division 1)

LPCS (Division 1)

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

Division 1 Fails, Divisions 2 and 3 Functional:

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

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

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

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

Assuming the single failure is the failure of Division 2, the safety function is accomplished by establishing one of the cooling loops described in Activity C2 of Figure 15.2-12. Figures 15.2-15, 16, and 17 show RHR loops A, B, and/or C (simplified).

For Activities A, B, C1 and C2(b), using the above assumptions and following the depressurization transient shown in Figures 15.2-13a and 15.2-13b, the suppression pool temperature is shown in Figures 15.2-14a and 15.2-14b. For the ASDC Activity C2(a), using the above assumptions and following the depressuization transient shown in Figure 15.2-13c, the suppression pool and wetwell temperature response is shown in Figure 15.2-14c. Note that it stays below the technical specification limit and therefore, even under worst-case conditions (failure of an emergency power division), a cooling path is available to remove decay heat and thus fully comply with GDC 34.

15.2.9.4 Barrier Performance

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

RPV pressures and suppression pool temperature are given in subsection 15.2.9.3 above. Release of coolant to the containment occurs via SRV actuation. Release of radiation to the environment is described below.

15.2.9.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in subsection 15.2.4.5. Therefore, the radiological exposures noted in subsection 15.2.4.5 cover the consequences of this event.

15.2.10 Loss of Instrument Air System

The reload fuel vendor has determined that the loss of instrument air system event is not a limiting event for the current reload cycle. Therefore, this subsection describes the original analysis performed by the NSSS vendor for the initial cycle which remains the current licensing basis for GGNS. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.2.10.1 Identification of Causes and Frequency Classification

15.2.10.1.1 Identification of Causes

Loss of the instrument air system for the plant during normal plant operation could occur as the result of a major line break in the system or as a result of mechanical or electrical failure of the normal instrument air supply and the backup service air source.

15.2.10.1.2 Frequency Classification

Due to a lack of a comprehensive data base this transient disturbance is being evaluated as an incident of moderate frequency.

15.2.10.2 Sequence of Events and System Operation

15.2.10.2.1 Sequence of Events

The following events will occur on a time schedule which depends on the location and type of failure, because the failure determines the depressurization rate of the system.

a. Control rod drive system - The scram inlet and outlet valves will open, shutting down the reactor. The control rod drive flow control valve will close to approximately 2 percent open. The drain and vent valves for the scram discharge volume will close.

   The main turbine pressure control system will maintain reactor pressure after the reactor is shut down until the turbine control valves are closed. If the reactor mode switch is still in the "run" mode, the main steam isolation valves will close and produce a scram signal as the reactor pressure decreases below 850 psi.

b. Reactor cleanup system - The cleanup filter demineralizer valves and the reject valve to radwaste or the main condensers will close.

c. Standby liquid control - The level indication for the storage tank will decrease to zero.

d. Main steam line isolation valves will close.

e. Main steam safety/relief valves will remain available.

f. Drywell cooling system dampers and containment cooling system isolation valves will close or remain closed with or without an isolation signal.

g. Containment cooling system cooling water valves will close. Drywell cooling system water valves will remain open.

h. Fuel pool and closed cooling water system makeup water valves will close.

i. The ventilation exhaust isolation dampers from the ECCS pump rooms and the fuel handling area will close with or without an isolation signal.

j. The control room ventilation system outside air supply dampers will close. Also, the control room utility exhaust dampers will close.

k. The RCIC steam line drain and RHR heat exchanger steam supply valves will receive a close signal.

l. The LPCI minimum flow valve will open.

m. All testable check valves in the systems will remain in their original positions.

15.2.10.2.1.1 Deleted

15.2.10.2.2 System Operation

This event assumes normal functioning of normal plant instrumentation and controls.

15.2.10.2.3 The Effect of Single Failure and Operator Errors

Failure of additional components (e.g., pressure regulator, feedwater flow controller) is discussed elsewhere in Chapter 15.

15.2.10.3 Core and System Performance

15.2.10.3.1 Mathematical Model

Qualitative evaluation provided only

15.2.10.3.2 Input Parameters and Initial Conditions

Qualitative evaluation provided only

15.2.10.3.3 Qualitative Results

Loss of the instrument air system will result in the shutdown of the reactor due to the opening of the control rod scram valves and/or the closing of the main steam line isolation valves. The failure of instrument air will not interfere with the safe shutdown of the reactor since all equipment using instrument air is designed to fail to a position that is consistent with the safe shutdown of the plant.

Air-operated equipment that must be available for use in the event of a failure of the instrument air system must be provided with a backup source to provide the required air supply.

15.2.10.4 Barrier Performance

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

15.2.10.5 Radiological Consequences

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

15.2.11 References

1. Brutschy F. G., et al, "Behavior of Iodine in Reactor Water During Plant Shutdown and Startup."

2. Nguyen, D., "Realistic Accident Analysis for General Electric Boiling Water Reactor - The RELAC Code and User's Guide," NEDO-21142, to be issued (December 1977).

3. Letter - R.S. Boyd to I. F. Stuart; dated November 12, 1975, Subject: Requirements Delineated for RHRS-Shutdown Cooling System-Single Failure Analysis.

4. Fukushima, T. Y., "Hex 01 User Manual," NEDE-23014, July 1976.

5. General Electric Topical Report NEDO-20533 Supplement 1, September 1975, "General Electric Mark III Pressure Suppression Containment System Analytical Model."

6. Letter MFN 341-76 by E. A. Hughes to Director of Nuclear Reactor Regulations Attn: R. C. DeYoung, "Turbine Trip Without Bypass Analyzed as an Infrequent Event," October 5, 1976.

7. Deleted

8. ECH-NE-10-00021, Rev. 5, "GNF2 Fuel Design Cycle-Independent Analyses for Grand Gulf Nuclear Station," Mar. 2020.

9. ECH-NE-10-00075, Rev. 1, “GGNS EPU Containment Systems Response,” GEH-0000-0106-8269-R1, May 2011.

10. Deleted

11. Deleted

12. Deleted

13. Deleted

14. GEH Safety Communication SC06-01, "Worst Single Failure for Suppression Pool Temperature Analysis," January 19, 2006.

15. GGNS-NE-10-00075, GGNS EPU T0400 Containment System Response, Revision 1, May 2011.

16. ECH-NE-20-00009, Rev. 0, Supplemental Reload Licensing Report for Grand Gulf-1 Reload 22 Cycle 23, February 2020.

17. GNRI-209/00030, “Grand Gulf Nuclear Station, Unit 1 – Issuance of Amendment to Modify the Updated Safety Analysis Report to Replace First Stage Pressure Signals with Power Range Neutron Monitoring System Signals (EPID L-2018-LLA-0072),” March 12, 2019. [Amendment No. 217]

18. ECH-NE-20-00006, Rev. 0, GNF3 Fuel Design Cycle-Independent Analyses for Grand Gulf Nuclear Station, February 2020.

TABLE 15.2-1: SEQUENCE OF EVENTS FOR PRESSURE CONTROLLER DOWNSCALE FAILURE (INITIAL CYCLE) (FIGURE 15.2-1) [HISTORICAL INFORMATION]

Time-sec	Event
0	Simulate zero steam flow demand to main turbine and bypass valves.
0	Turbine control valves start to close.
1.09	Neutron flux reaches high flux scram set point and initiates a reactor scram.
2.76	Safety/relief valves open due to high pressure.
2.76	High-pressure recirculation pump trip initiated.
9.8	Safety/relief valves close.
9.86	Group 1 safety/relief valves open again to relieve decay heat.
>15.0(est.)	Group 1 safety/relief valves close

TABLE 15.2-1A: SEQUENCE OF EVENTS FOR PRESSURE CONTROLLER DOWNSCALE FAILURE (CURRENT CYCLE) 100% Power, 105% Flow

Time-sec (approx.**)	Events
0	Simulate zero steam flow demand to main turbine and bypass valves.
0	Turbine control valves start to close.
1.06	Neutron flux reaches high flux scram set point and initiates a reactor scram
3.40	Safety/relief valves open due to high pressure.
*	High-pressure recirculation pump trip initiated.

* The high pressure recirculation pump trip was not simulated which maximizes the calculated peak vessel pressure.

** Exact timing varies based on power shape and exposure.

TABLE 15.2-2: SEQUENCE OF EVENTS FOR GENERATOR LOAD REJECTION WITH BYPASS (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT)
(FIGURE 15.2-2)

Time-sec	Event
(-)0.015(approx)	Turbine-generator detection of loss of electrical load.
0	Turbine-generator load rejection sensing devices trip to initiate turbine control valve fast closure and main turbine bypass system operation.
0	Fast control valve closure (FCV) initiates scram trip and recirculation pump trip (RPT).
0.07	Turbine control valves closed.*
0.10	Turbine bypass valves start to open.
2.02	Safety/relief valves open due to high pressure.
7.39	Safety/relief valves close.
25.3(est.)	Turbine bypass valves start to close.
26.8(est.)	Turbine bypass closed.
38.1(est.)	Turbine bypass valves reopen to regulate pressure.

*Partially open when the transient occurs.

TABLE 15.2-3: SEQUENCE OF EVENTS FOR GENERATOR LOAD REJECTION W/O BYPASS CURRENT CYCLE

Time-sec (approx.*)	Event
(-)0.015	Turbine-generator detection of loss of electrical load.
0	Turbine-generator load rejection sensing devices trip to initiate turbine control valve fast closure.
0	Turbine bypass valves fail to operate.
0.03	Fast control valve closure (FCV) initiates scram trip and recirculation pump trip (RPT).
0.063	Turbine control valves closed.
0.20	Begin recirculation pump trip (RPT).
0.23	Start of control blade motion.
1.90	Safety/relief valves open due to high pressure.
>5.00(est.)	Safety/relief valves cycle (not simulated).

*Exact timing varies based on power shape and exposure.

TABLE 15.2-3A: Deleted

TABLE 15.2-3B: SEQUENCE OF EVENTS FOR GENERATOR LOAD REJECTION, BYPASS OFF (INITIAL CYCLE)
(FIGURE 15.2-3)

Time-sec (approx.)	Event
(-)0.015	Turbine-generator detection of loss of electrical load.
0	Turbine-generator load rejection sensing devices trip to initiate turbine control valve fast closure.
0	Turbine bypass valves fail to operate.
0	Fast control valve closure (FCV) initiates scram trip and recirculation pump trip (RPT).
0.07	Turbine control valves closed.
1.35	Safety/relief valves open due to high pressure.
7.60	Safety/relief valves close.
7.74	Group 1 safety/relief valves open again to relieve decay heat.
10.0	Group 1 safety/relief valves close again
32.0 (est.)	Group 1 safety/relief valves open again to relieve decay heat.
35.0 (est.)	Group 1 safety/relief valves close again.
36.0 (est.)	L8 trip initiates a feedwater trip.

TABLE 15.2-4: SEQUENCE OF EVENTS FOR TURBINE TRIP WITH BYPASS(INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT)
(FIGURE 15.2-4)

Time-sec	Event
0	Turbine trip initiates closure of main stop valves and bypass system operation.
0	Main turbine stop valves trip fluid pressure initiates reactor scram trip and recirculation pump trip (RPT).
0.1	Turbine stop valves close.
0.1	Turbine bypass valves start to open to regulate pressure.
1.6	Safety/relief valves open due to high pressure.
5.7	Safety/relief valves close.
24.5 (est.)	Turbine bypass valve starts to close
25.6 (est.)	Turbine bypass valve closed.
36.0 (est.)	Turbine bypass valve reopens on pressure signal.

TABLE 15.2-5: [HISTORICAL INFORMATION] SEQUENCE OF EVENTS FOR TURBINE TRIP W/O BYPASS (INITIAL CYCLE) (FIGURE 15.2-5)

Time-sec(approx.)	Event
0	Turbine trip initiates closure of main stop valves.
0	Turbine bypass valves fail to operate.
0.01	Main turbine stop valves trip fluid pressure initiates reactor scram trip, and recirculation pump trip (RPT).
0.1	Turbine stop valves close.
1.38	Safety/relief valves open due to high pressure.
7.56	Safety/relief valves close.
7.59	Group 1 safety/relief valves open again to relieve decay heat.
>10.0(est.)	Group 1 safety/relief valves close again.
>20.0(est.)	Group 1 safety/relief valves open again to relieve decay heat.
>23.0(est.)	Group 1 safety/relief valves close again.

TABLE 15.2-5A: SEQUENCE OF EVENTS FOR TURBINE TRIP W/O BYPASS (CURRENT CYCLE)(100% POWER, 105% FLOW RATED FW TEMP)

Time-sec (approx.*)	Event
0	Turbine trip initiates closure of main stop valves.
0	Turbine bypass valves fail to operate.
0.004	Main turbine stop valves trip fluid pressure initiates reactor scram trip, and recirculation pump trip (RPT).
0.10	Turbine stop valves close.
0.20	Begin recirculation pump trip (RPT).
2.96	Safety/relief valves open due to high pressure.
0.24	Start of control blade motion.

*Exact timing varies based on power shape and exposure.

TABLE 15.2-6: SEQUENCE OF EVENTS FOR INADVERTENT MSIV CLOSURE (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT)
(FIGURE 15.2-6)

Time-sec	Event
0	Initiate closure of all main steam line isolation valve (MSIV).
0.3	MSIVs reach 90 percent open.
0.3	MSIV position trip scram initiated.
2.95	Recirculation pump motor trip due to high vessel dome pressure.
3.10	Safety/relief valves actuate due to high pressure.
8.63	Safety/relief valves close.
8.72	Group 1 safety/relief valves actuate again to relieve decay heat.
8.91	Group 2 safety/relief valves actuate again to relieve decay heat.
>10(est)	Group 1 safety/relief valves close again

TABLE 15.2-7: TYPICAL RATES OF DECAY FOR CONDENSER VACUUM (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT)

Cause	Estimated Vacuum Decay Rate
Failure of Isolation of Steam Jet Air Ejectors	<1 inch Hg/minute
Loss of Sealing Steam to Shaft Gland Seals	~1 to 2 inches Hg/minute
Opening of Vacuum Breaker Valves	~2 to 12 inches Hg/minute
Loss of One or More Circulating Water Pumps	~4 to 24 inches Hg/minute initial, increasing to ~10 in. Hg/sec at about 10 inches Hg vacuum.

TABLE 15.2-8: SEQUENCE OF EVENTS FOR LOSS OF CONDENSER VACUUM (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT) (FIGURE 15.2-7)

Time-sec	Event
-3.0 (est)	Initiate simulated loss of condenser vacuum.
0.0 (est)	Low condenser vacuum main turbine trip actuated. Assume a conservative average rate of decay of condenser vacuum of 10 inches Hg per second.
0.0 (est)	Low condenser vacuum feedwater trip actuated.
0.01	Main turbine trip initiates recirculation pump trip (RPT) and scram.
1.0	Low condenser vacuum initiates main steam line isolation valve closure.
1.0	Low condenser vacuum initiates bypass valve closure.
2.0	Safety/relief valves open due to high pressure.
7.2	Safety/relief valves close.
7.8	Safety/relief valves open again to relieve decay heat.
13.0	Safety/relief valves close again.
16.0	Safety/relief valves open again to relieve decay heat.
16.5	Water level reaches Level 2 set point initiates HPCS and RCIC (not simulated).
21.0	Safety/relief valves close again.
46.5(est)	HPCS and RCIC flow enters vessel (not stimulated).

TABLE 15.2-9: TRIP SIGNALS ASSOCIATED WITH LOSS OF CONDENSER VACUUM ANALYSIS (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT)

Vacuum (inches or Hg)	Protective Action Initiated
27 to 28	Normal Vacuum Range
20	Main Turbine Trip and Feedwater Turbine Trip (Stop Valve Closures)
10	Main Steam Isolation Valve (MSIV) Closure and Bypass Valve Closure

TABLE 15.2-10: SEQUENCE OF EVENTS FOR LOSS OF SERVICE TRANSFORMER (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT))
(FIGURE 15.2-8)

Time-sec	Event
0	Loss of service transformer occurs.
0	Recirculation system pump motors are tripped.
0	Condensate booster pumps are tripped.
0	Condenser circulating water pumps are tripped.
2.0	Scram and MSIV closure occur due to loss of power to the solenoids.
4.0	Feedwater turbine trip due to MSIV closure.
5.0	Safety/relief valves actuate due to high pressure.
10.3	Safety relief valves close.
22.2	Vessel water level reaches Level 2 set point.
52.2(est)	HPCS AND RCIC flow enters vessel (not simulated).

TABLE 15.2-11: SEQUENCE OF EVENTS FOR LOSS OF GRID CONNECTIONS (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT) (FIGURE 15.2-9)

Time-Sec	Event
(-)0.015 (approx.)	Loss of Grid causes turbine-generator to detect a loss of electrical load.
0	Turbine control valve fast closure is initiated.
0	Turbine-generator PLU trip initiates main turbine bypass system operation.
0	Recirculation system pump motors are tripped.
0	Fast control valve closure (FCV) initiates a reactor scram trip.
0.08	Turbine control valves closed.
0.14	Turbine bypass valves open.
1.7	Safety/relief valves actuate due to high pressure.
2.0	Closure of MSIV due to loss of power.
4.0	Feedwater pumps trip due to MSIV closure.
18.6	Safety/relief valves close.
21.7	Vessel water level reaches Level 2 set point.
51.7(est)	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.2-12: [HISTORICAL INFORMATION] SEQUENCE OF EVENTS FOR LOSS OF ALL FEEDWATER FLOW (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT) (FIGURE 15.2-10)

Time-Sec	Event
0	Trip of all feedwater pumps initiated.
3.48	Vessel water level reaches Level 4 and initiates recirculation flow runback.
6.71	Vessel water level (L3) trip initiates scram trip.
4.7	Feedwater flow decays to zero.
13.67	Vessel water level reaches Level 2.
13.87	Recirculation pumps trip due to Level 2 trip.
43.67(est)	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.2-13: SEQUENCE OF EVENTS FOR FAILURE OF RHR SHUTDOWN COOLING
(Activities A, B, C1 and C2(b))

Approximate Elapsed Time	Events
0	Reactor is operating at 105 percent initially licensed NBR steam flow when LOP transient occurs initiating plant shutdown.
0	Concurrently loss of Division power (i.e., loss of one diesel generator) occurs.
0	Suppression pool temperature alarm has been activated (alarm at 90°F).*
7.5 min	Suppression pool temperature reaches 110°F. This technical specification limit requires the operator to scram the plant if not already scrammed.
10 min	Suppression pool cooling initiated to prevent overheating from SRV actuation.**
42 min	Suppression pool temperature is held to a maximum of 120°F. Full blowdown initiated.
68 min	Blowdown to 100 psi completed.
98 min	Personnel are sent in to open RHR shutdown cooling suction valve; this fails.
	Actuate ADS and complete blowdown to suppression pool.
103 min	Redirect RHR pump discharge from pool to vessel via LPCI line. Alternate path now established.

*Initial suppression pool temperature is conservatively assumed to be 95°F for this transient.

**See Table 15.2-10 for detailed sequence of events for loss of ac power transient.

TABLE 15.2-14: INPUT PARAMETERS FOR EVALUATION OF FAILURE OF RHR SHUTDOWN COOLING

(Activities A, B, C1 and C2(b))

Initial Power (% of 3833 MW)	105
Suppression Pool Mass (lbm)	8.66×10^{6} (1)
RHR (KHX value) (Btu/Sec/˚F)	
Pool cooling	540
Cooled water to vessel	511
Initial vessel condition	
Pressure (psia)	1060
Temperature (˚F)	552
Initial primary fluid inventory (lbm)	6.71×10^{5}
Initial pool temperature, (˚F)	95
Service water temperature, (˚F)	90
Vessel heat capacity (Btu/lbm/˚F)	0.123
HPCS on - water level (ft)	
On	40.93
Off	49

TABLE 15.2-14: INPUT PARAMETERS FOR EVALUATION OF FAILURE OF RHR SHUTDOWN COOLING (Continued)

HPCS flow rate, (gpm)	7450

Note:

1. In response to NRC Bulletin 96-03, ER 97/0089-00-00 installed a new ECCS/RCIC suction strainer, which rests on the floor of the suppression pool, to replace one of the conical basket strainers on each of the ECCS and RCIC system suction strainers. The ECCS/RCIC suction strainer displaces ~500 ft^3 of suppression pool water. Analysis has shown that the displacement of the water has a negligible effect on the existing analyses.

TABLE 15.2-15: CLOSURE OF ALL MAIN STEAM ISOLATION VALVES ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

I-131	6.41E-02
I-132	2.65E-01
I-133	3.72E-01
I-134	2.53E-01
I-135	4.33E-01
Kr-83m	3.68E-02
Kr-85	6.16E-04
Kr-85m	9.65E-02
Kr-87	1.63E-01
Kr-88	2.62E-01
Kr-89	6.18E-02
Rb-88	2.36E-03
Xe-131m	1.17E-03
Xe-133	9.71E-01
Xe-133m	5.99E-02
Xe-135	1.02E+01
Xe-135m	6.16E+00
Xe-137	9.26E-02
Xe-138	1.86E-01

TABLE 15.2-16:CLOSURE OF ALL MAIN STEAM ISOLATION VALVES RADIOLOGICAL EFFECTS

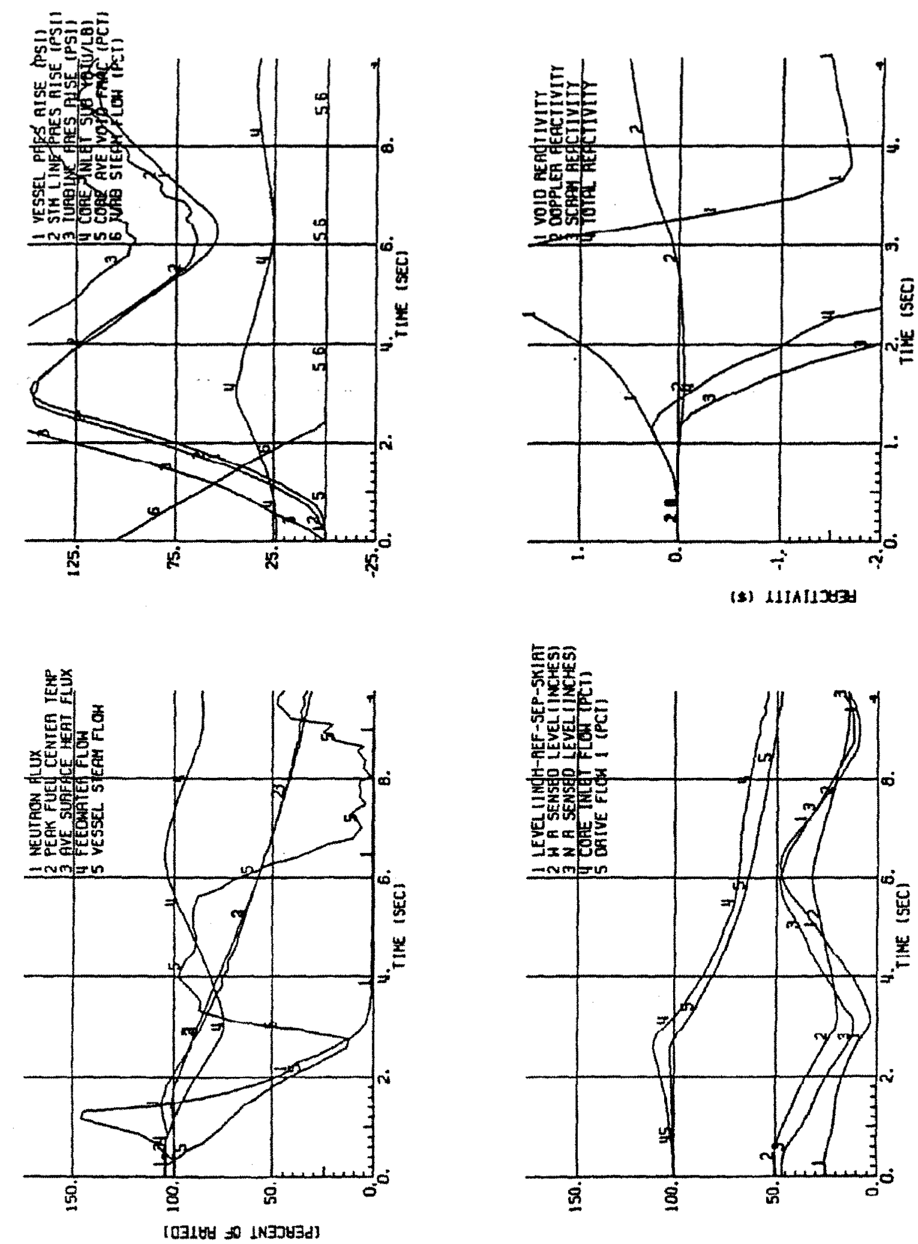
Site boundary:

Annual average χ/Q	= 2.0E-5 sec/m^3
Total Effective Dose Equivalent	< 0.083 mrem/event

TABLE 15.2-17: TURBINE STOP & CONTROL VALVE CLOSURE DATA
(Figures 15.2-19a, b & c, and 15.2-20a & b)

		Control Valves at TUTR [1]) = LORE [2])				Stop Valves at TUTR				
Load	**Steam Flow [9])**	**Stroke[3] %)**	**Dead Time[4] ms)**	**Travel Time[5] ms)**	**Characteristic**	**Stroke %**	**Dead Time[6] ms)**	**Dead Time[7] ms)**	**Travel Time[8] ms)**	**Characteristic**
100	99	V1=72 V2&4=65	100	150	[10])	100	80	140	100	[10])
75	74	V1=37 V2&4=30	140	110	about linear	100	80	200	80	about linear
50	49	V1=27 V2&4=20	160	90	about linear	100	80	220	60	about linear

1) TUTR = Turbine Trip
2) LORE = Load Rejection
3) Valves 2 & 4 open together at about 7% - position of valve 1
4) Time difference between beginning of the failure (TUTR, LORE) and beginning of valve movement beginning of throttling of the steam flow
5) Time difference in which the valve stroke or steam flow will go down to zero
6) Time difference between starting of the failure (TUTR) and starting of valve movement
7) Time difference between starting of the failure (TUTR) and starting of steam flow throttling
8) Time difference in which the steam flow goes down to zero
9) 100% - steam flow at fully opened valves
10) See Figures 15.2-19a, b & c for control valve and Figure 15.2-20a & b for stop valve.



GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

PRESSURE REGULATOR DOWNSCALE FAILURE
(INITIAL CYCLE) [HISTORICAL INFORMATION]
FIGURE 15.2-1

Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.

Current cycle results are presented in reference 8.

Pressure Regulator Downscale Failure (Current Cycle) Key Parameters
Figure 15.2-1A

Figures 15.2-1B through 15.2-1E
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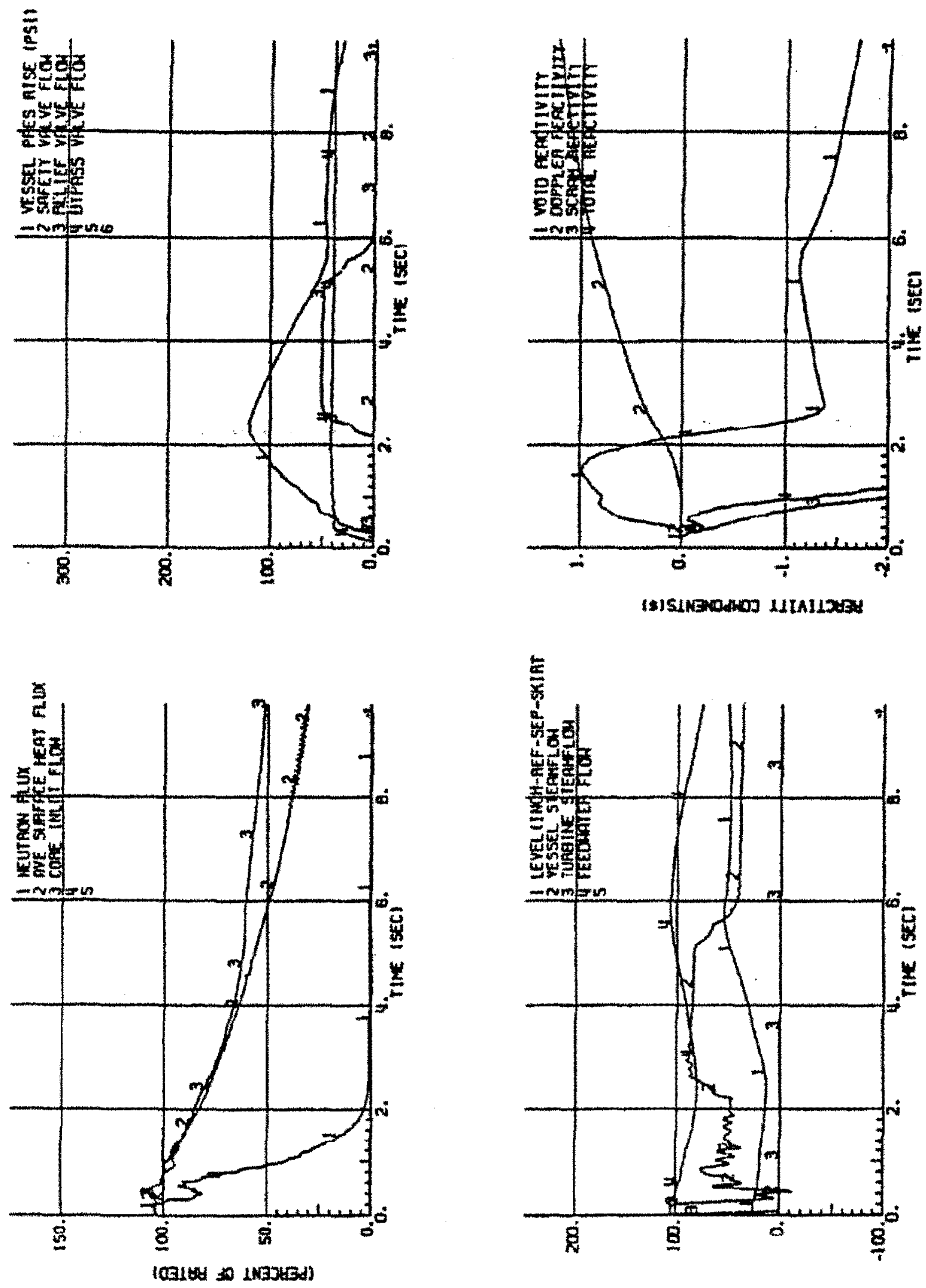
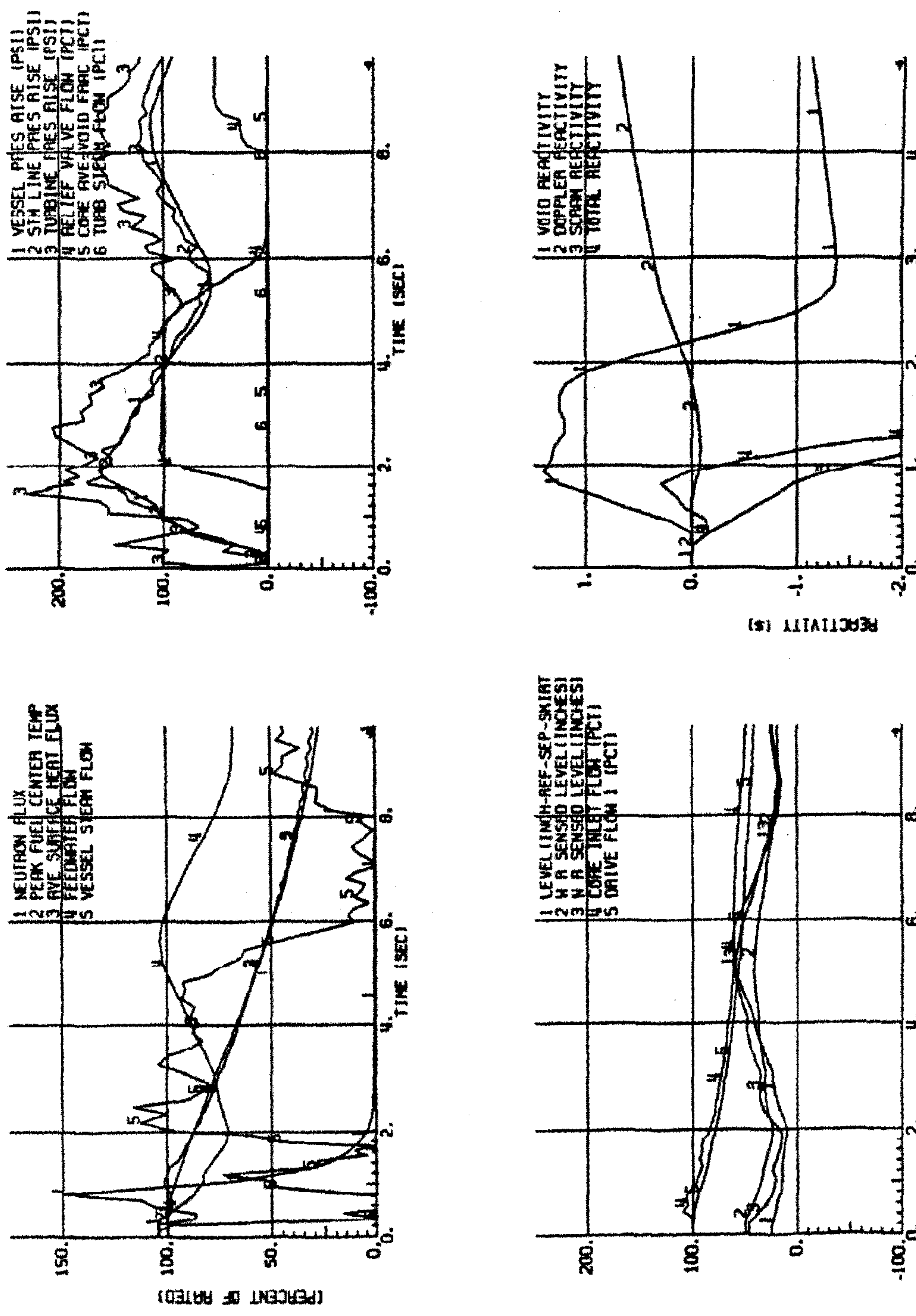


Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW, but remains the current analysis of record for this event.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	GENERATOR LOAD REJECTION, TRIP SCRAM, BYPASS-ON (INITIAL CYCLE) FIGURE 15.2-2



GRAND GULF NUCLEAR STATION
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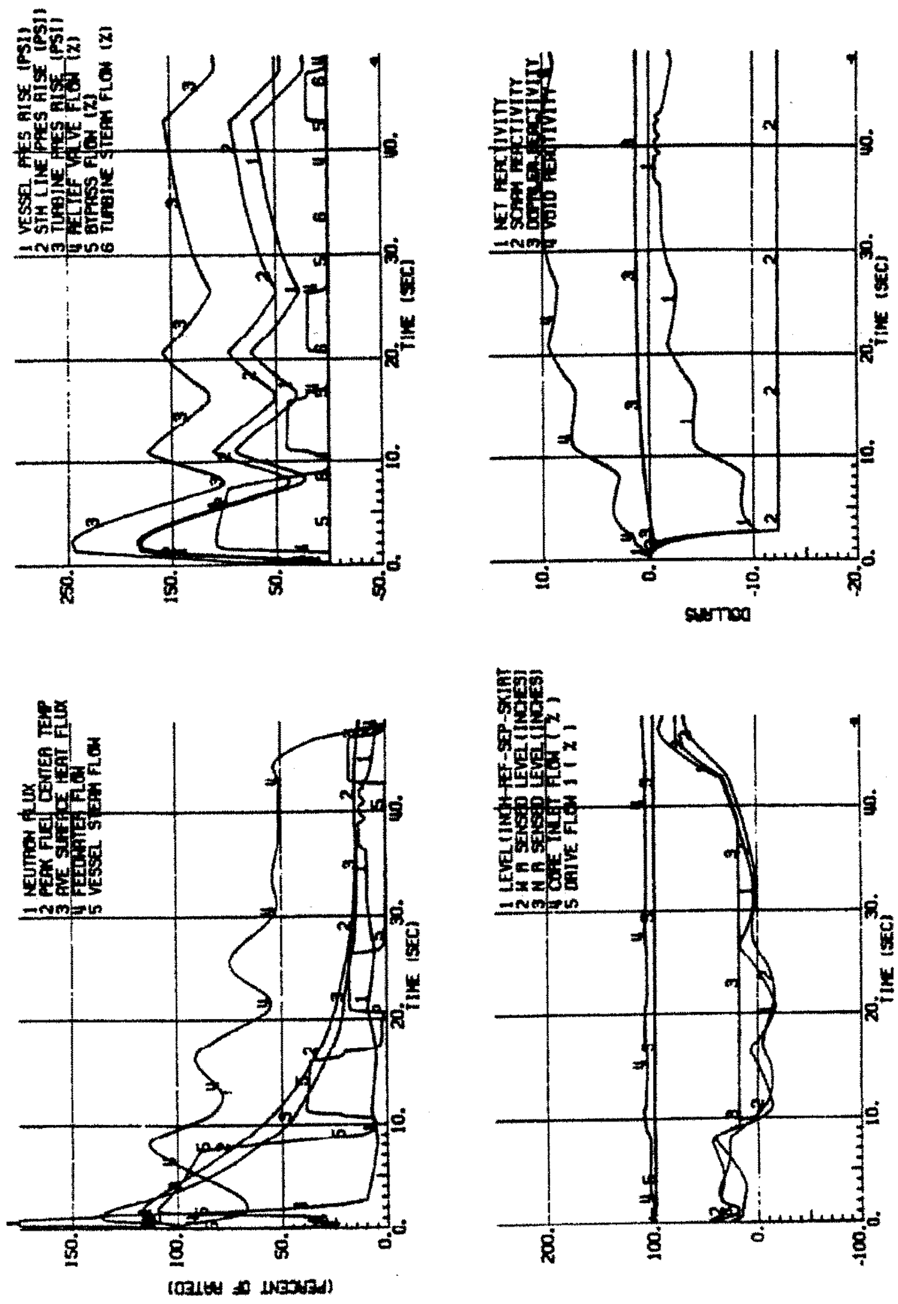
UPDATED FINAL SAFETY ANALYSIS REPORT

GENERATOR LOAD REJECTION,
BYPASS FAILURE
[HISTORICAL INFORMATION]
[INITIAL CYCLE]
FIGURE 15.2-3

Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.



Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FULL LOAD REJECTION WITHOUT BYPASS W/O 2 PUMP TRIP, FLUX SCRAM FULL POWER [HISTORICAL INFORMATION] [INITIAL CYCLE] FIGURE 15.2-3A

Figure 15.2-3B Deleted

Figures 15.2-3c through 15.2-3f
Deleted

Figures 15.2-3G through 15.2-3M

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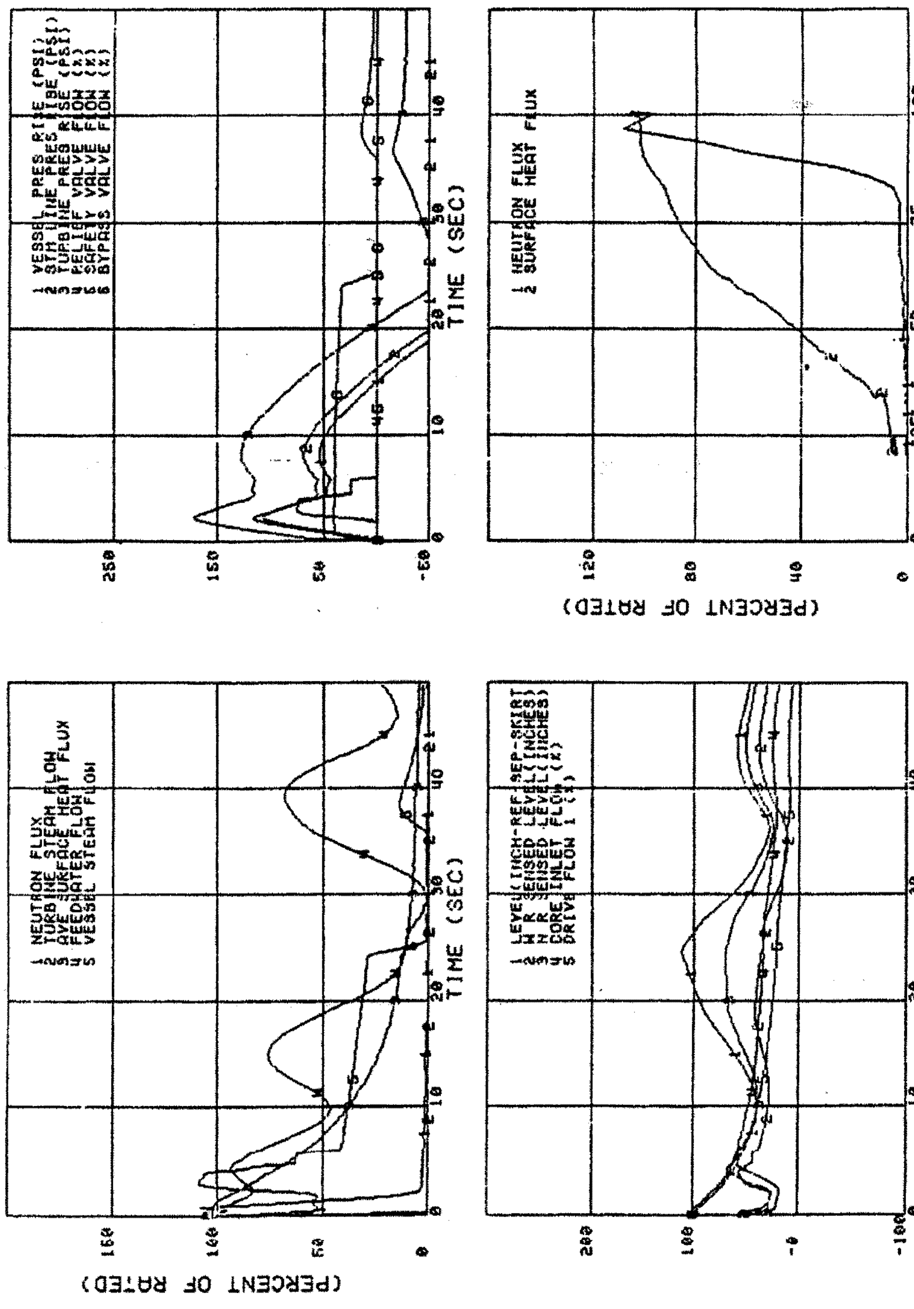

GRAND GULF NUCLEAR STATION
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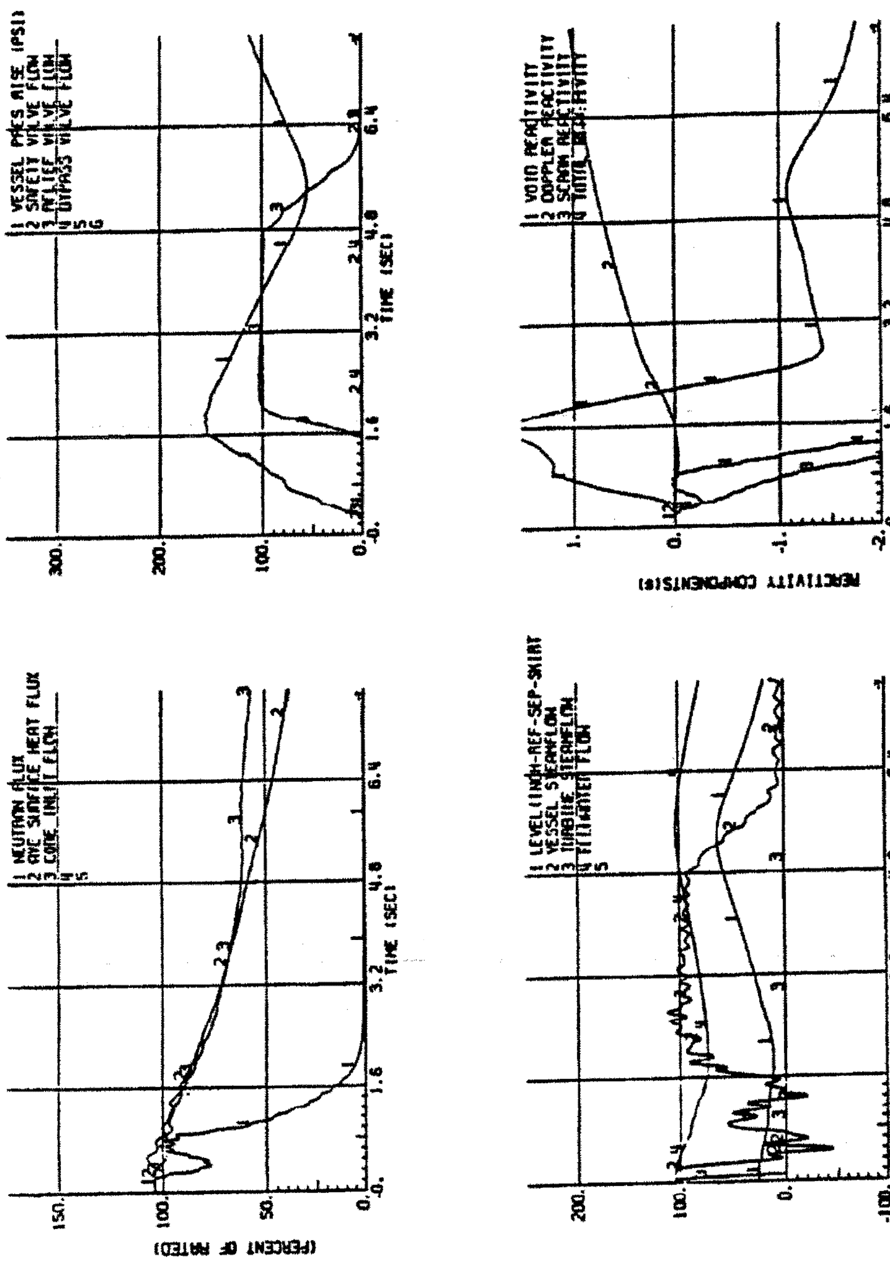
UPDATED FINAL SAFETY ANALYSIS REPORT

TURBINE TRIP, TRIP SCRAM, BYPASS AND RPT - ON
(INITIAL CYCLE)
FIGURE 15.2-4

Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW, but remains the current analysis of record for this event.



GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

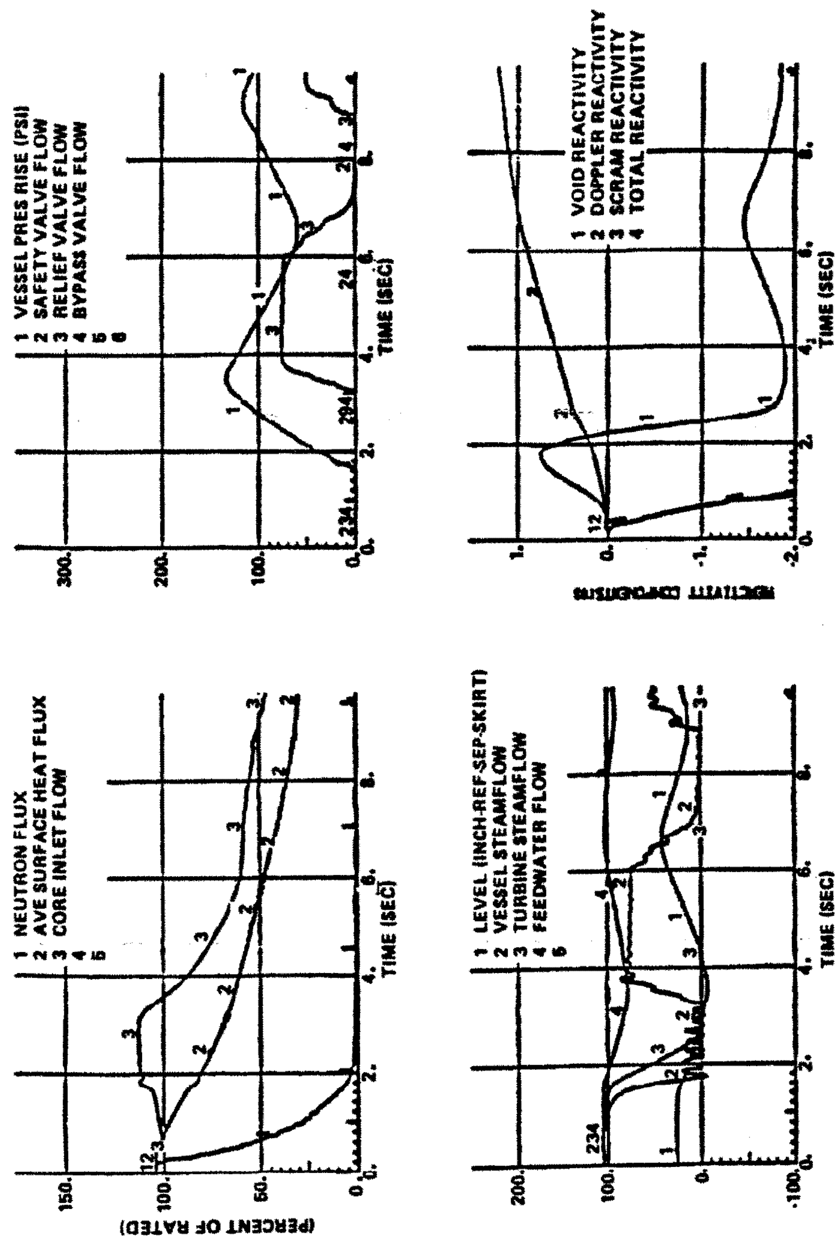
TURBINE TRIP, TRIP SCRAM,
BYPASS - OFF, RPT - ON
(INITIAL CYCLE) [HISTORICAL INFORMATION]
FIGURE 15.2-5

Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.

Figure 15.2-5A Deleted

Figures 15.2-5B through 15.2-5E

Deleted



GRAND GULF NUCLEAR STATION
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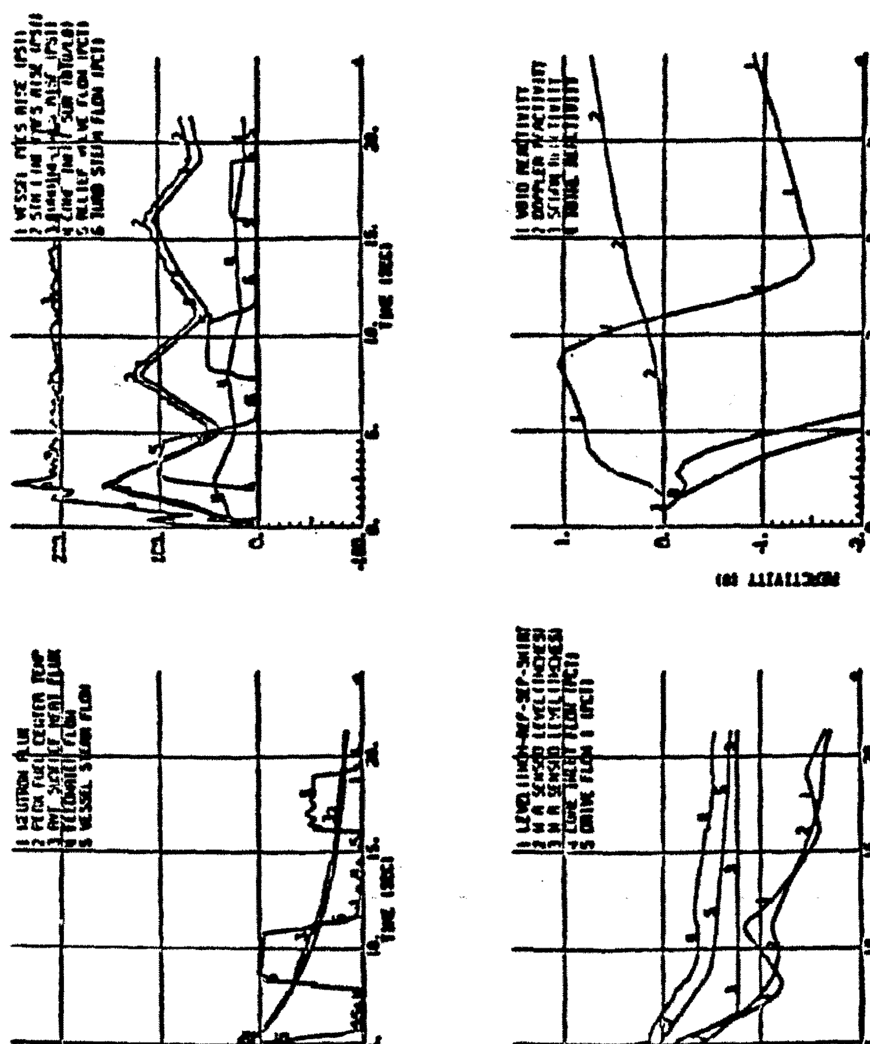
THREE-SECOND CLOSURE OF ALL MAIN STEAM LINE ISOLATION VALVES WITH POSITION SWITCH SCRAM TRIP
(INITIAL CYCLE)
FIGURE 15.2-6

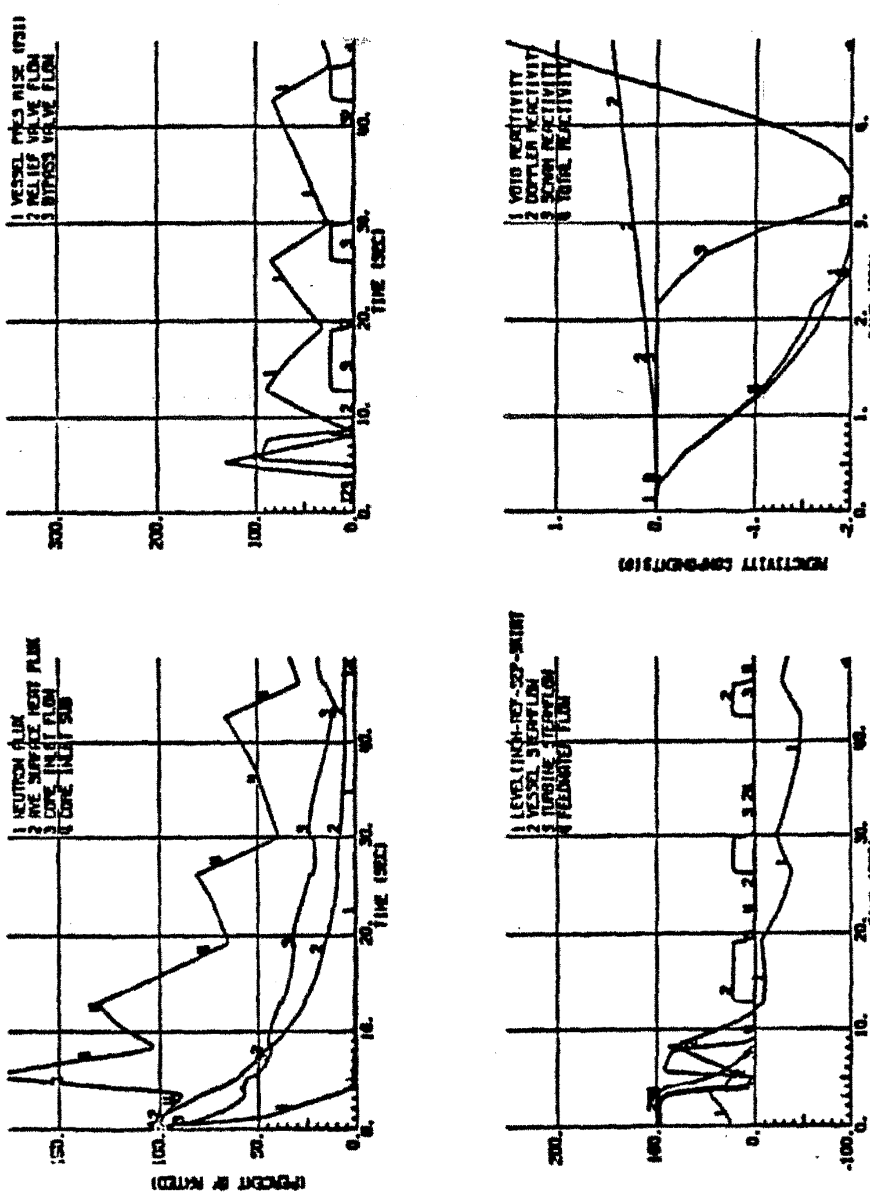
Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW, but remains the current analysis of record for this event.



GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

LOSS OF CONDENSER VACUUM
AT 10 INCHES PER SECOND
(INITIAL CYCLE)
FIGURE 15.2-7

Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW,
but remains the current analysis of record for this event.



GRAND GULF NUCLEAR STATION
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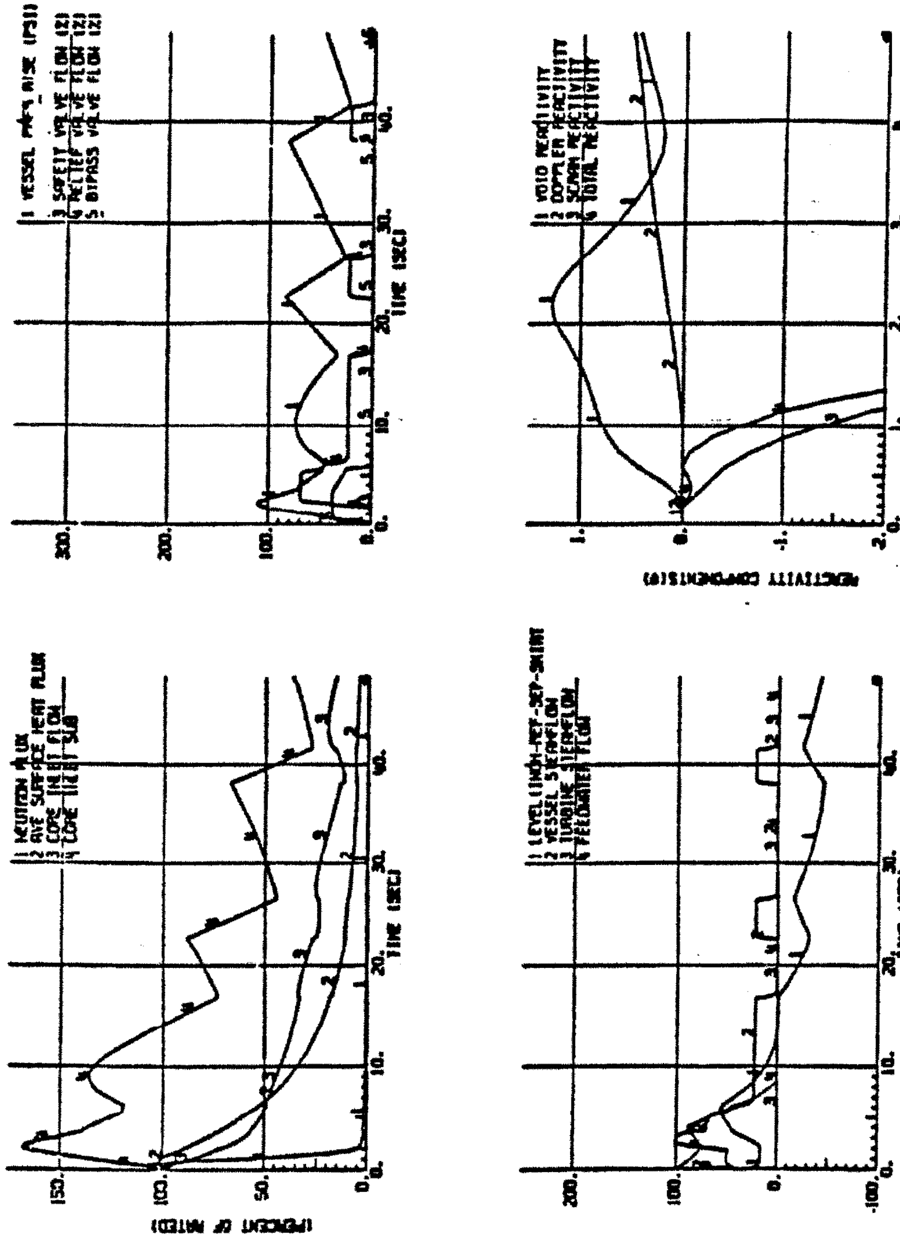
UPDATED FINAL SAFETY ANALYSIS REPORT

LOSS OF SERVICE TRANSFORMER
(INITIAL CYCLE)
FIGURE 15.2-8

Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW, but remains the current analysis of record for this event.



GRAND GULF NUCLEAR STATION
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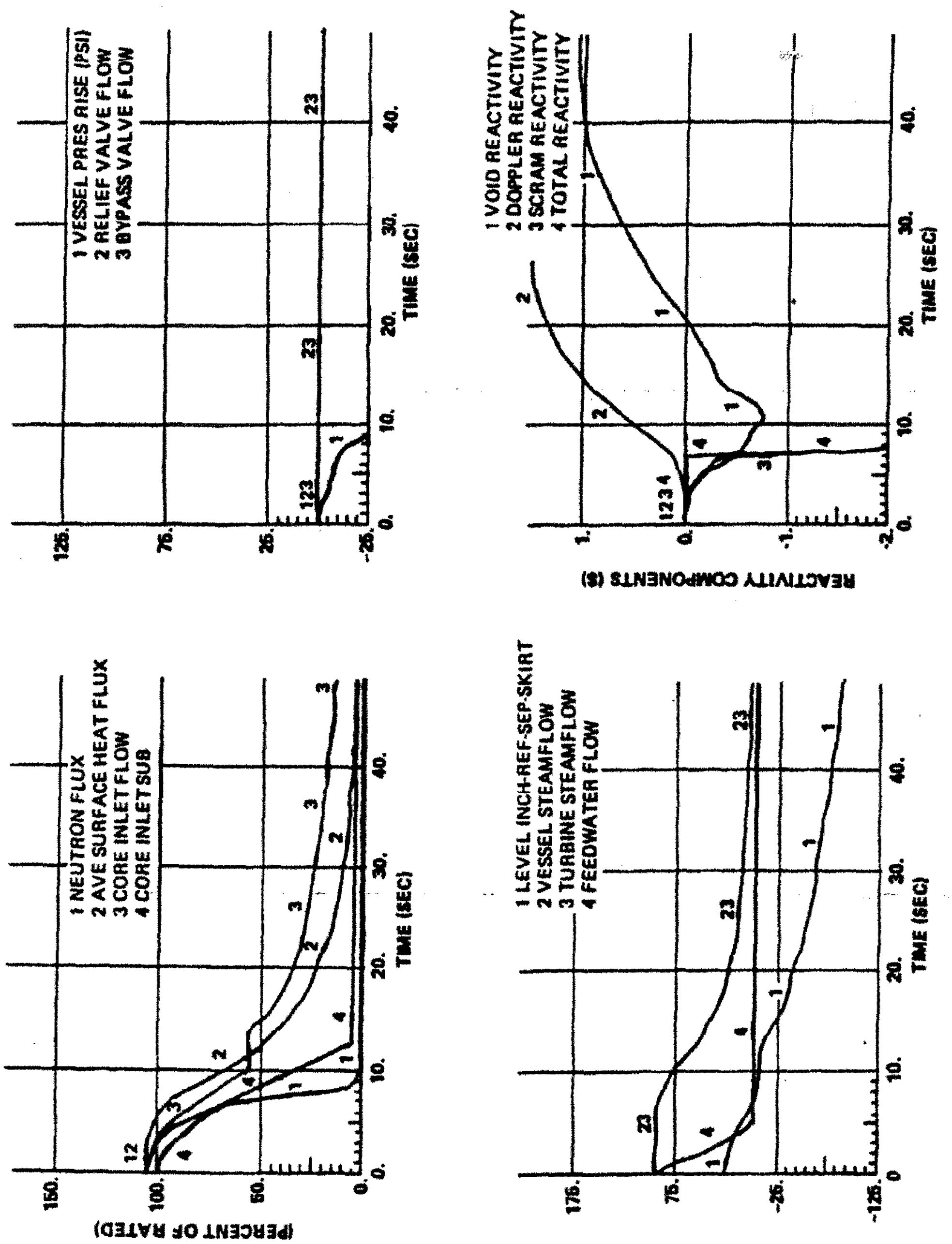
LOSS OF ALL GRID CONNECTIONS
(INITIAL CYCLE)
FIGURE 15.2-9

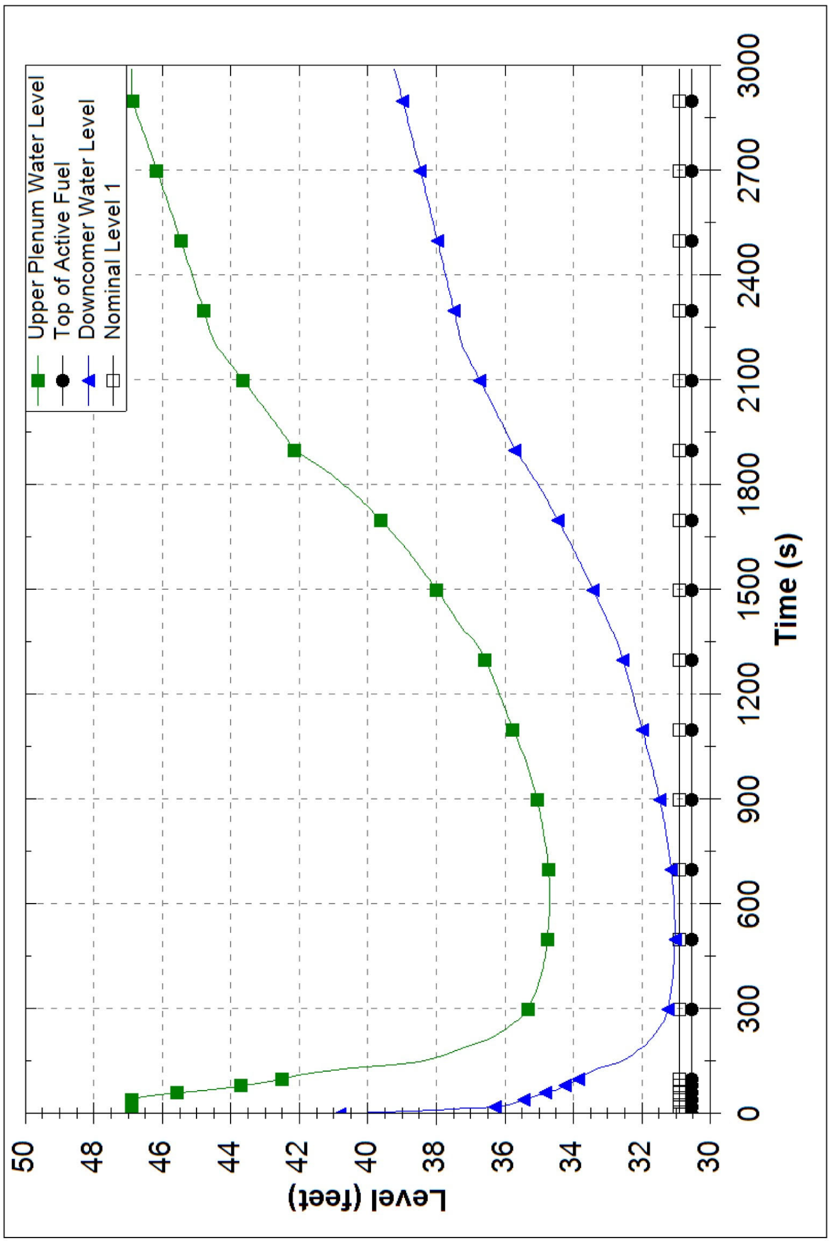
Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW, but remains the current analysis of record for this event.



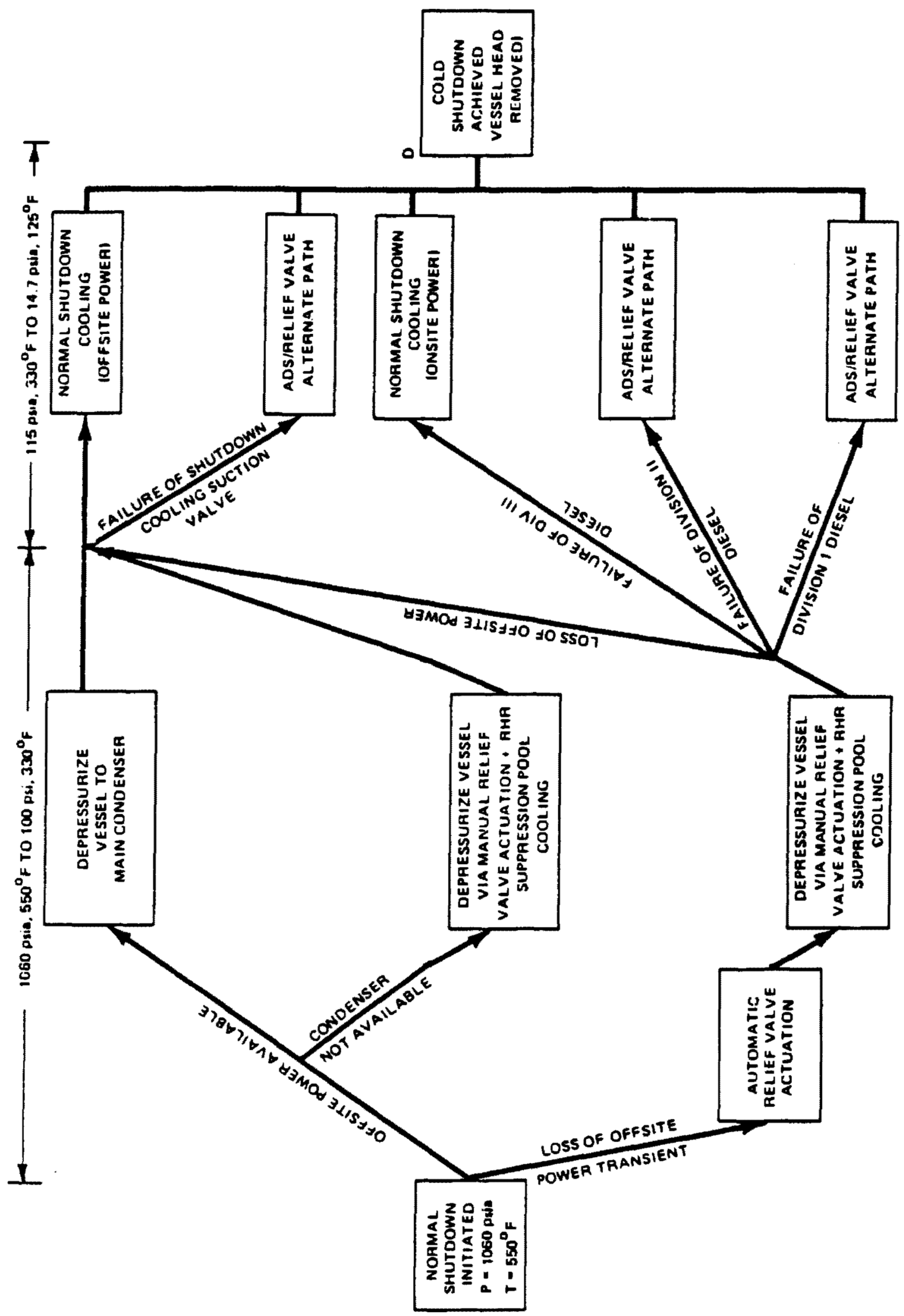
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	LOSS OF ALL FEEDWATER FLOW (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15.2-10

Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW, but remains the current analysis of record for this event.

FIGURE 15.2-10a: Loss of Feedwater Flow



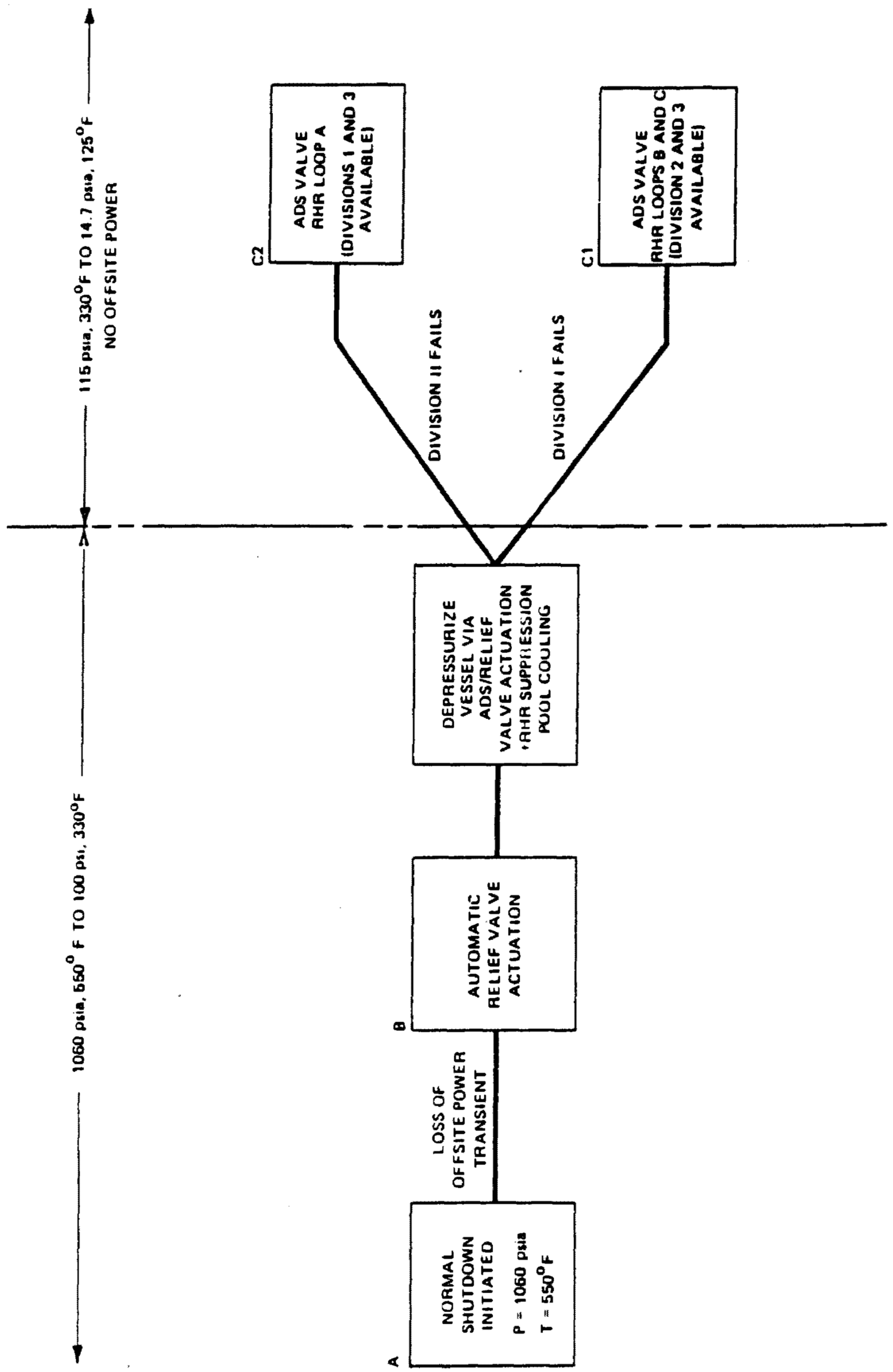
Note: The level is in reference to vessel bottom.



GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

SUMMARY OF PATHS AVAILABLE TO ACHIEVE COLD SHUTDOWN

FIGURE 15.2-11



GRAND GULF NUCLEAR STATION
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UPDATED FINAL SAFETY ANALYSIS REPORT

ADS/RHR COOLING LOOPS

FIGURE 15.2-12 (Sheet 1 of 3)

NOTES FOR FIGURE 15.2-12

ACTIVITY A

Initial pressure = 1060 psia
Initial temperature = 550°F

For purposes of this analysis, the following worst-case conditions are assumed to exist:

1. The reactor is assumed to be operating at 105% of rated flow at licensed conditions (i.e. initially licensed NB rated flow for Activities A, B, C1 and C2(b) and EPU rated flow for ASDC Activity C2(a));

2. a loss of power transient occurs (see subsection 15.2.6); and

3. a simultaneous loss of onsite power (Division 1 or Division 2), which eventually results in the operator not being able to open one of the RHR shutdown cooling line suction valves.

ACTIVITY B

Initial system pressure = 1060 psia
Initial system temperature = 550°F

Operator Actions

During approximately the first 42 minutes, reactor decay heat is passed to the suppression pool by the automatic operation of the reactor relief valves. Reactor water level will be returned to normal by the HPCS and RCIC system automatic operation.

After approximately 10 minutes, it is assumed one RHR heat exchanger will be placed in the suppression pool cooling mode to remove decay heat. At this time, the suppression pool will be 108 degrees. At approximately 40 minutes into the transient, the operator is required to initiate depressurization of the reactor vessel due to the combination of vessel pressure and suppression pool temperature. Controlled depressurization procedures consist of controlling vessel pressure and water level by using the ADS, RCIC and HPCS systems.

NOTES FOR FIGURE 15.2-12 (Cont'd)

When the reactor pressure approaches 100 psig, the operator would normally prepare for operation of the RHR system in the shutdown cooling mode. At this time (68 min), the suppression pool temperature will be 160°F.

ACTIVITY C1 (Division 1 fails, Division 2 available)

System pressure = 100 psi
System temperature = 330°F

Operator Actions

The operator establishes a closed cooling path as follows:

1. Either of the following cooling paths are established:

   (a) Utilizing RHR loop B, water from the suppression pool is pumped through the RHR heat exchanger (where a portion of the decay heat is removed) into the reactor vessel. The cooled suppression pool water flows through the vessel (picking up a portion of the decay heat) out the safety relief valves and back to the suppression pool. This alternate cooling path is shown in Figure 15.2-15.

(b) Utilizing RHR loops B and C together, water is taken from the suppression pool and pump directly into the reactor vessel. The water passes through the vessel (picking up decay heat) and out the safety relief valves returning to the suppression pool as shown in Figure 15.2-16. Suppression pool water is then cooled by operation of RHR loop B in the cooling mode (see Figure 15.2-17). In this alternate cooling path RHR loop C is used for injection and RHR loop B for cooling. Cold shutdown is achieved approximately 10.7 hours after transient occurred.

NOTES FOR FIGURE 15.2-12 (Cont'd)

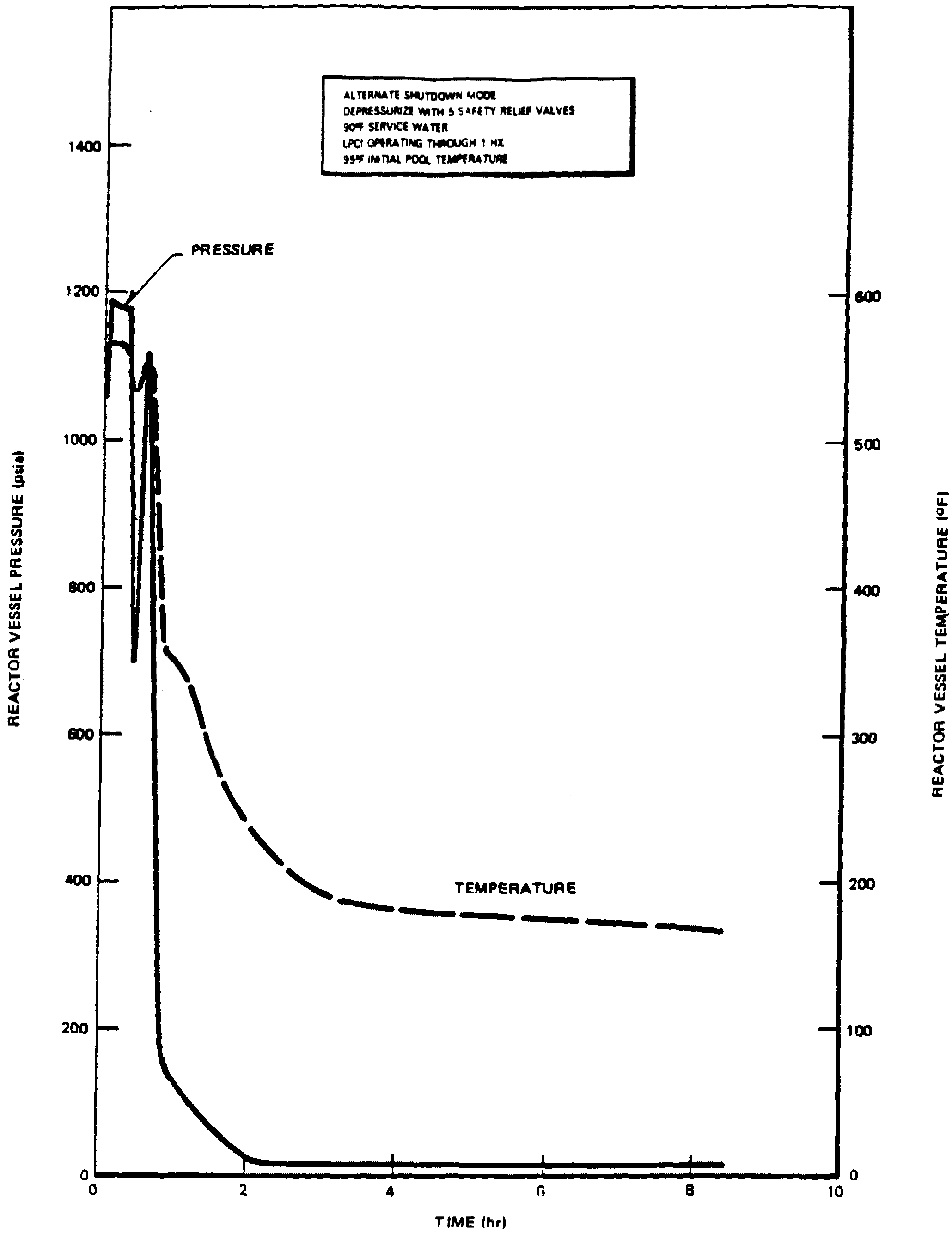
ACTIVITY C2 (Division 2 fails, Division 1 available)

System pressure = 100 psi
System temperature = 330°F

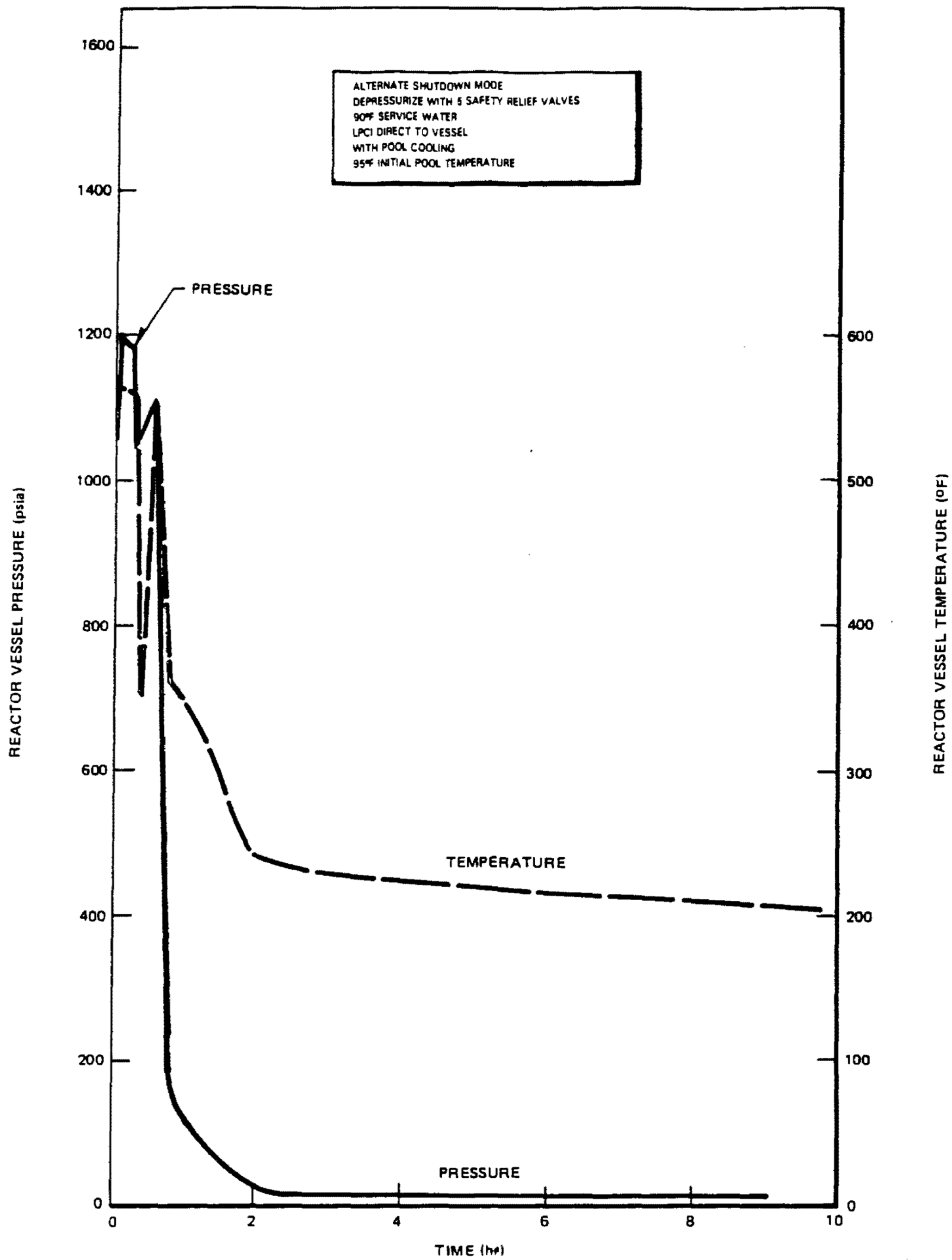
Operator Actions

Activity C2(a)- Utilizing RHR loop A instead of loop B, an alternate cooling path is established as in Activity C1 item 1(a) above. Again, cold shutdown is reached in approximately 27 hours.

Activity C2(b) — Utilizing RHR Loop A and LPCS, water is taken from the suppression pool and pump directly into the reactor vessel. The water passes through the vessel (picking up decay heat) and out the safety relief valves returning to the suppression pool as shown in Figure 15.2-18. Suppression pool water is then cooled by RHR loop A in the cooling mode (See Figure 15.2-18). In this alternate cooling path RHR loop A is used for cooling and LPCS for injection.

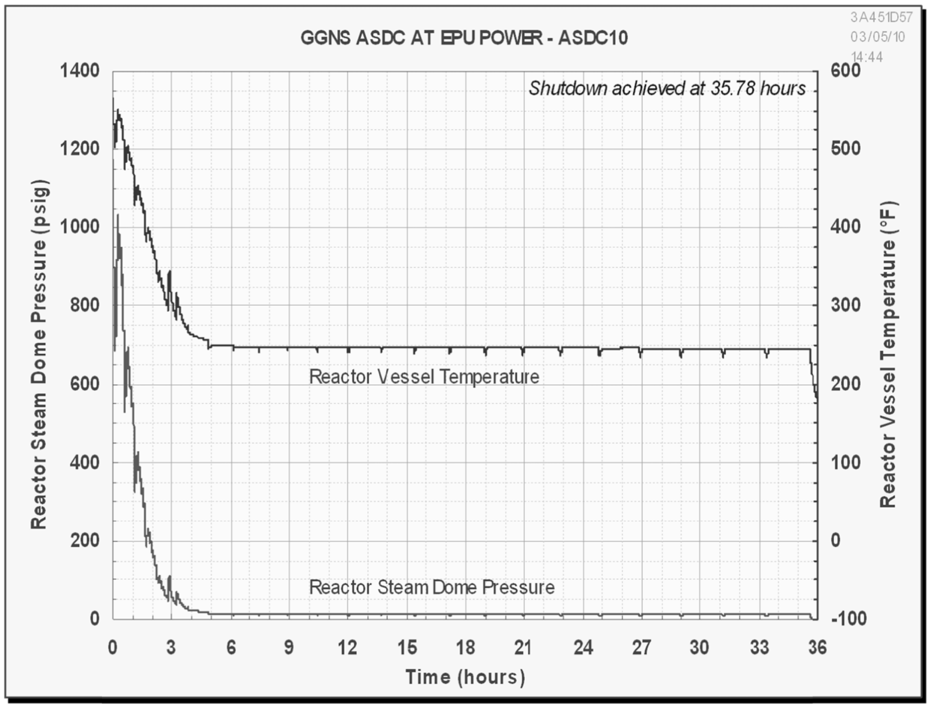


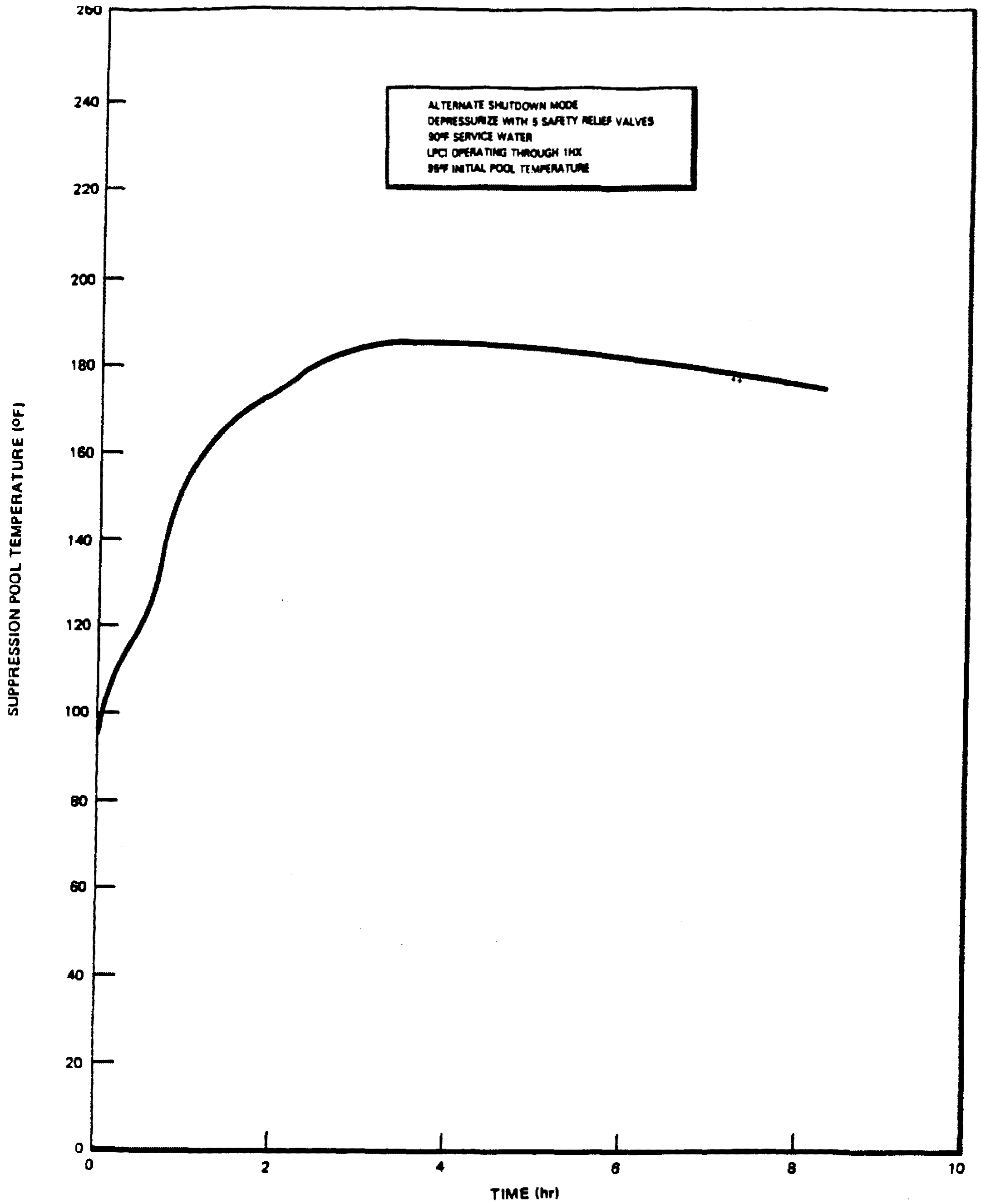
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	VESSEL TEMPERATURE AND PRESSURE VERSUS TIME (ACTIVITY C1(a)) FIGURE 15.2-13a



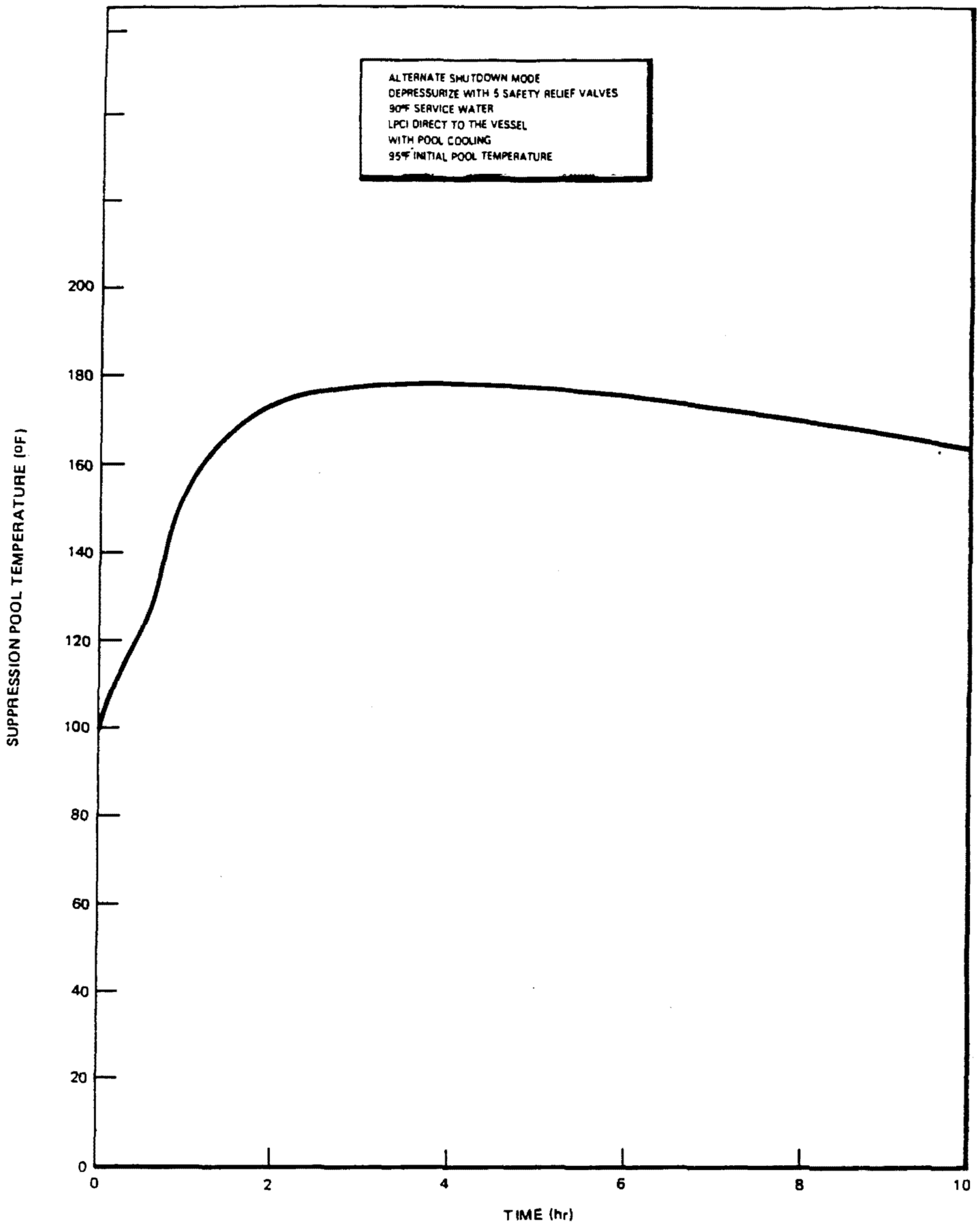
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	VESSEL TEMPERATURE AND PRESSURE VERSUS TIME (ACTIVITY C1(b)) FIGURE 15.2-13b

FIGURE 15.2-13c: Long-Term ASDC RPV Pressure and Temperature Response (Activity C2(a))



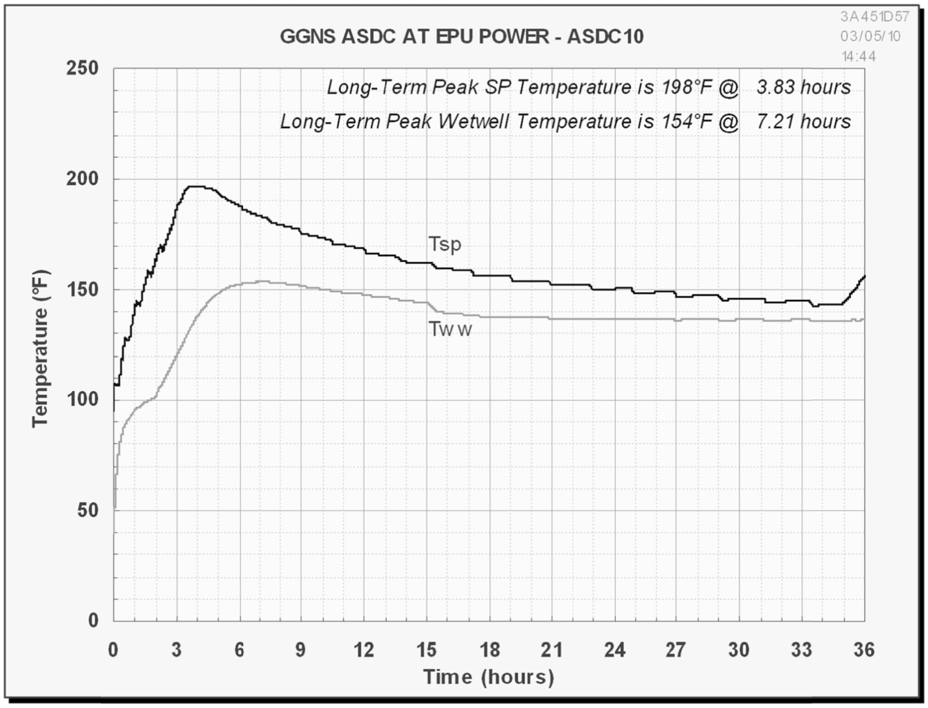


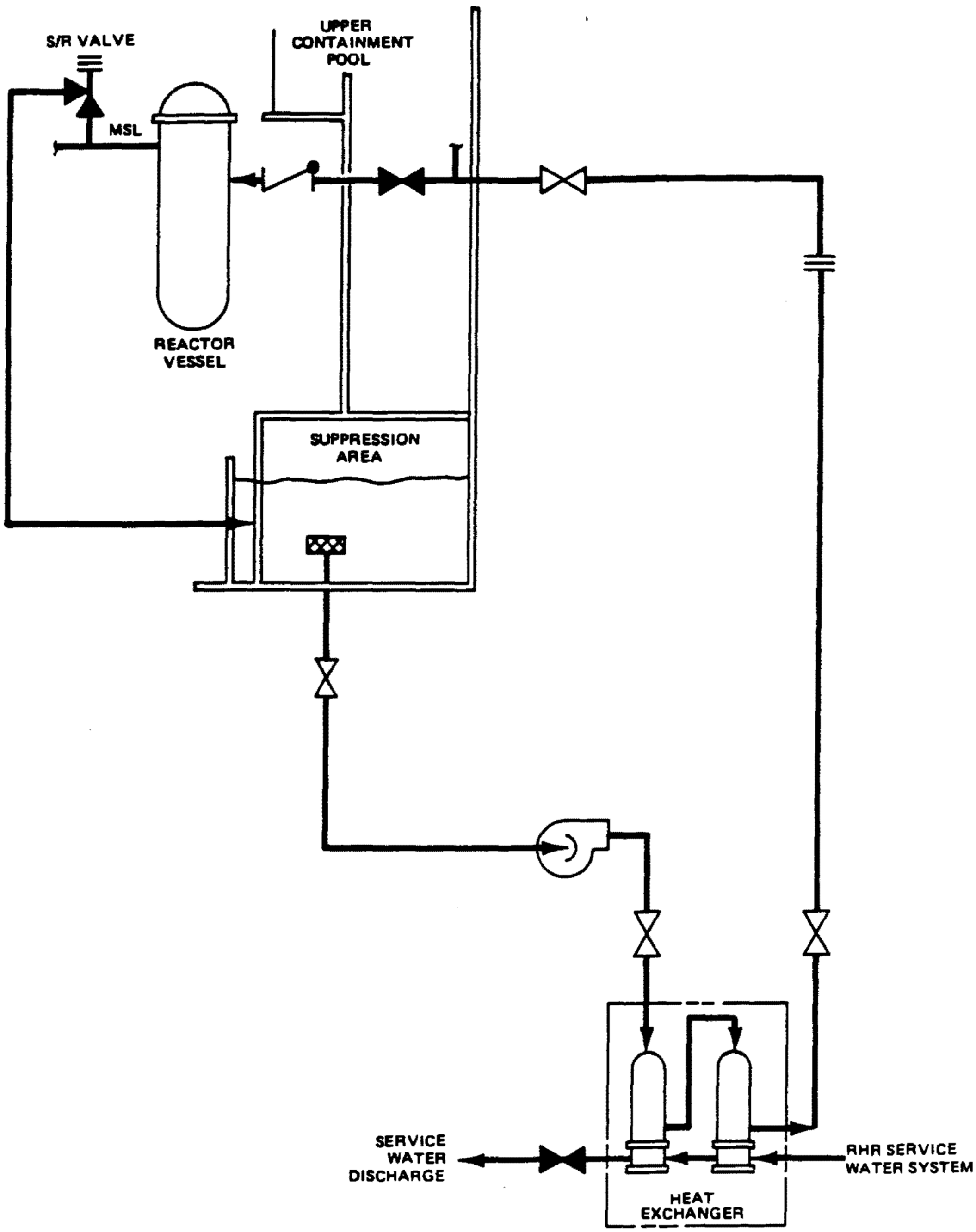
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	SUPPRESSION POOL TEMPERATURE VERSUS TIME (WITH 90° F SERVICE WATER TEMPERATURE) (ACTIVITY C1(a)) FIGURE 15.2-14a



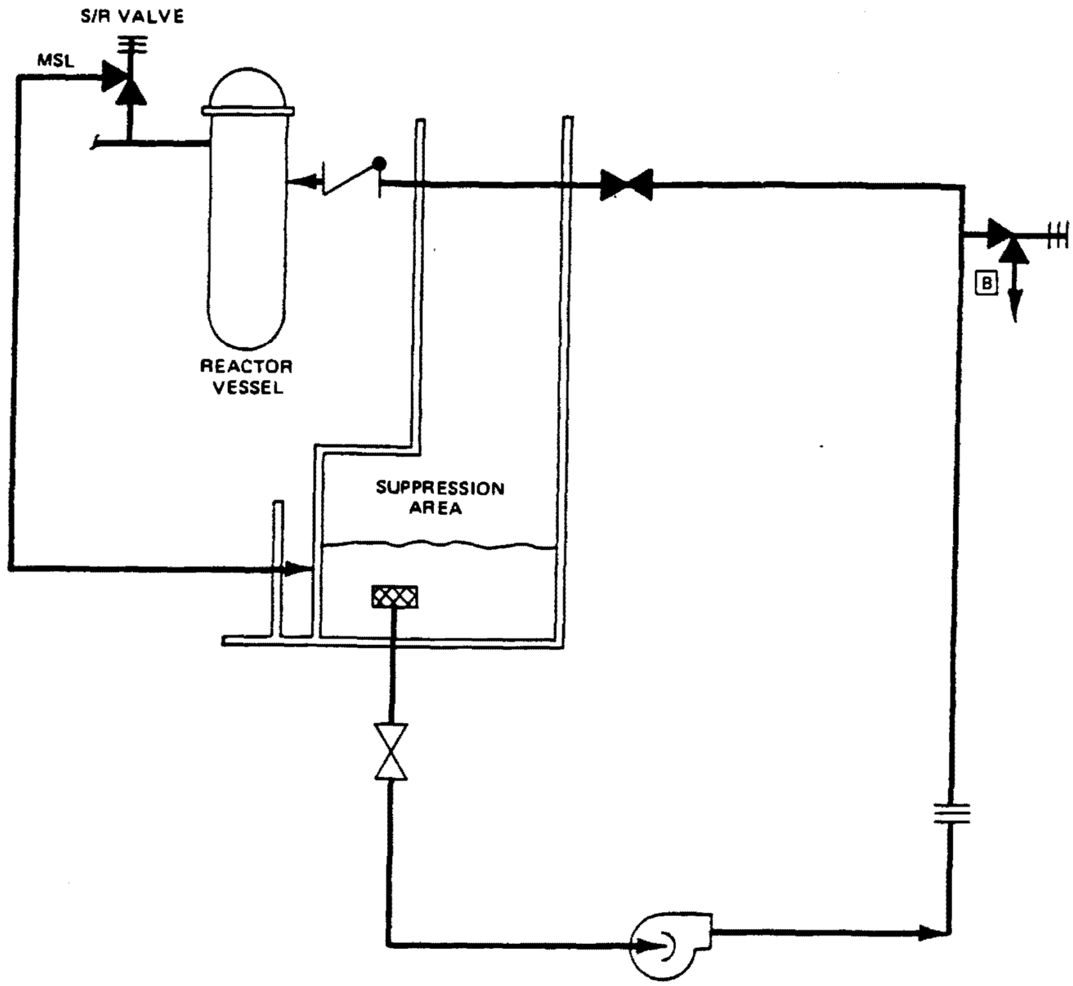
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	SUPPRESSION POOL TEMPERATURE VERSUS TIME (WITH 90° F SERVICE WATER TEMPERATURE) ACTIVITY C1(b) FIGURE 15.2-14b

FIGURE 15.2-14c: Long-Term ASDC SP and WW Temperature Response (Activity C2(a))

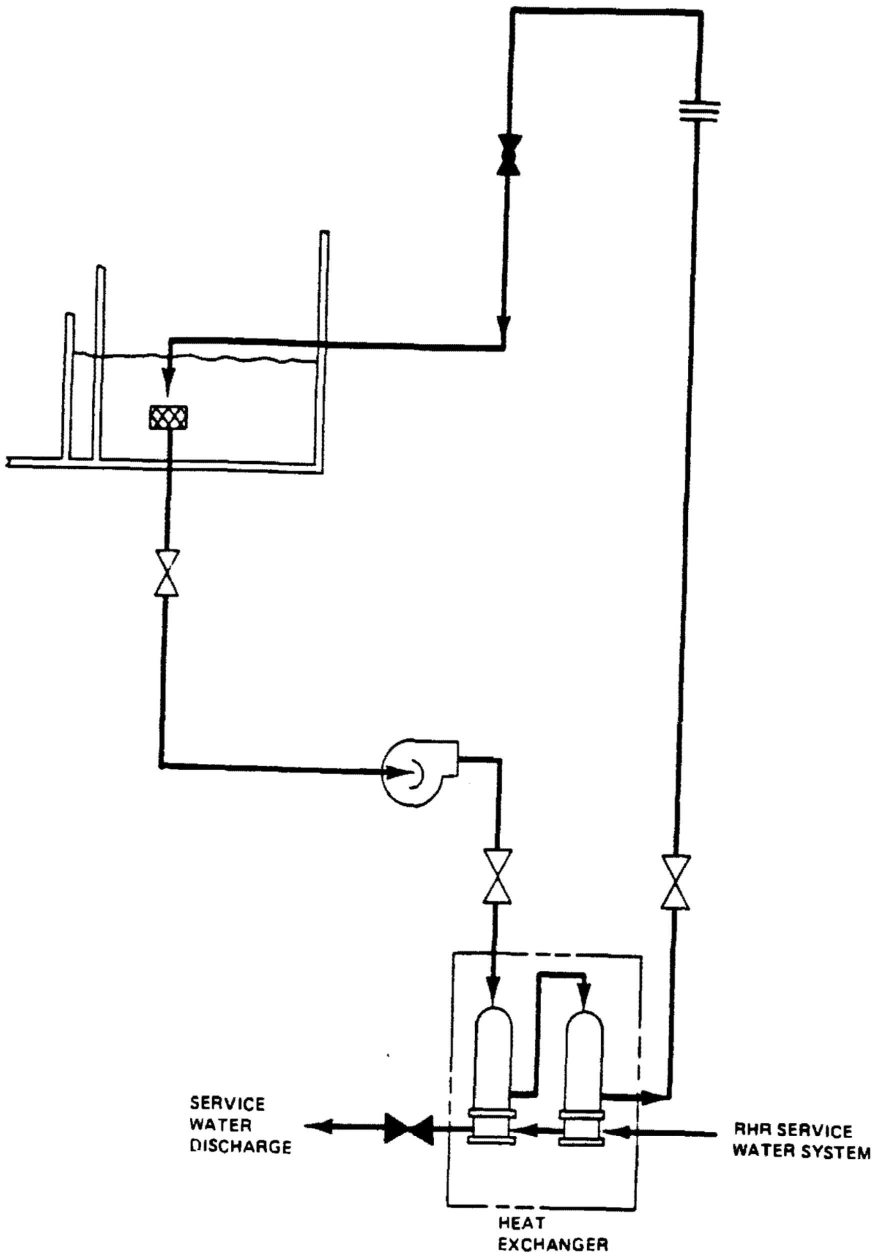




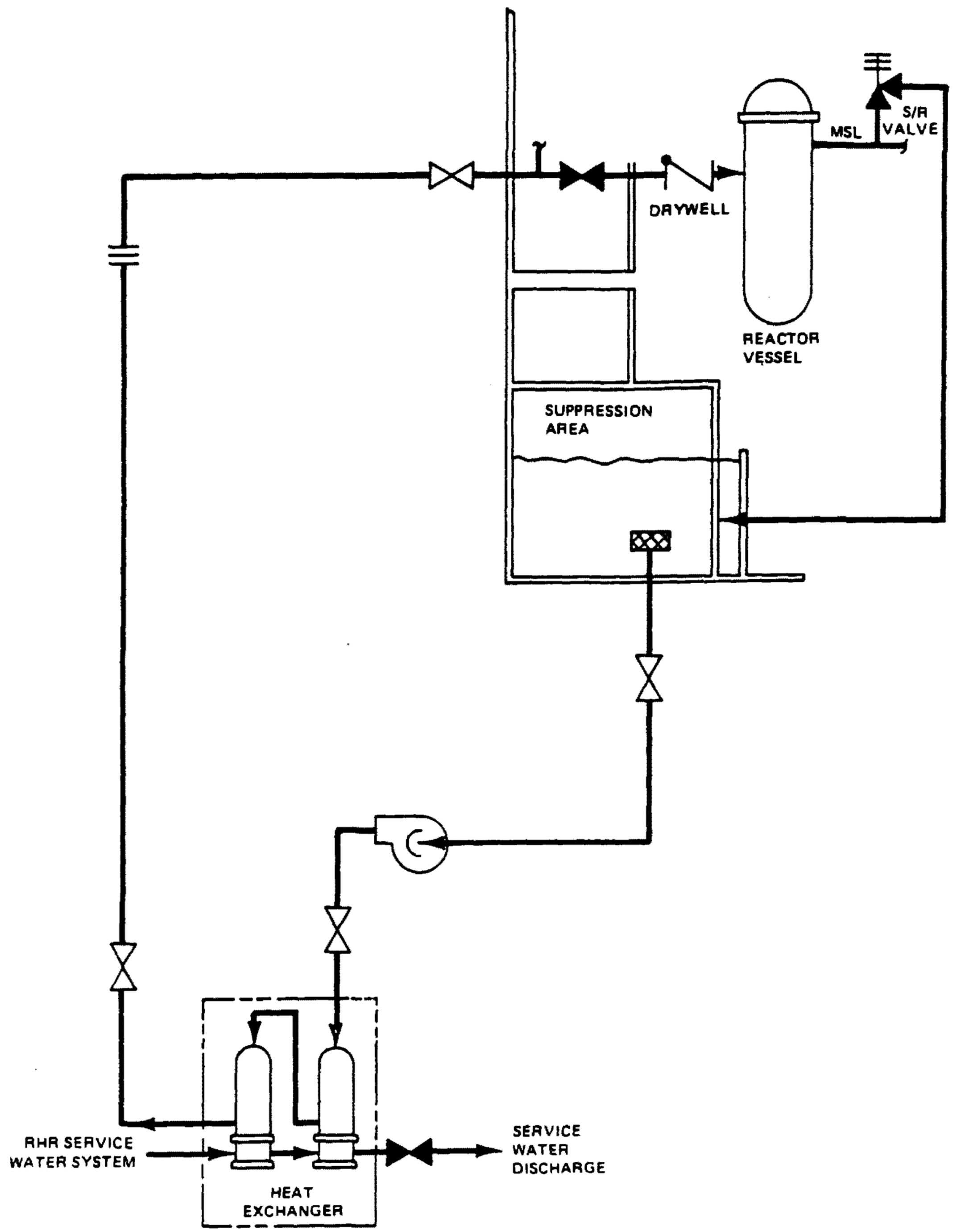
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	ACTIVITY C1 ALTERNATE SHUTDOWN COOLING PATH UTILIZING RHR LOOP B FIGURE 15.2-15



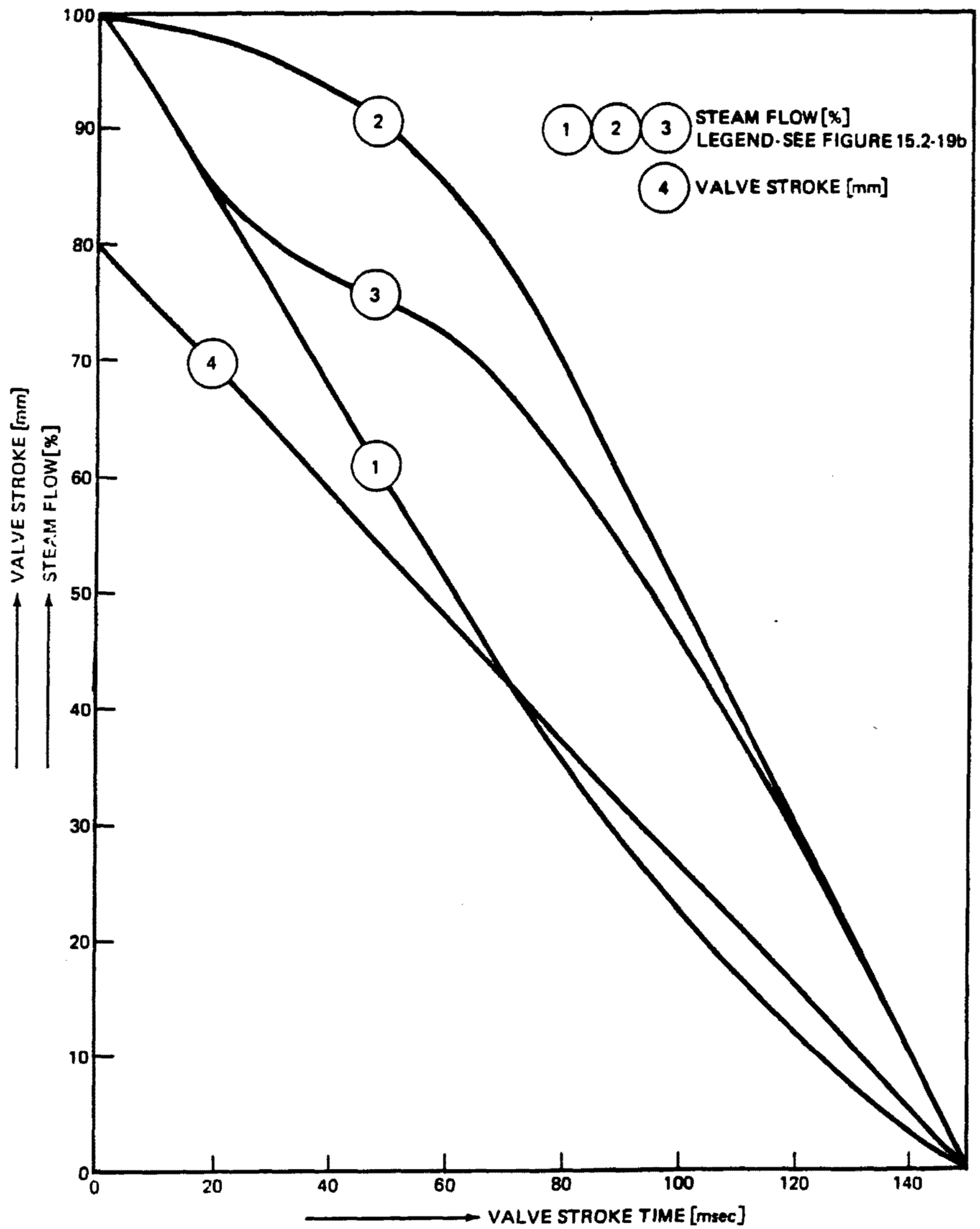
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	RHR LOOP C FIGURE 15.2-16



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	RHR LOOP B (SUPPRESSION POOL COOLING MODE) FIGURE 15.2-17



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	ACTIVITY C2 ALTERNATE SHUTDOWN COOLING PATH UTILIZING RHR LOOP A FIGURE 15.2-18



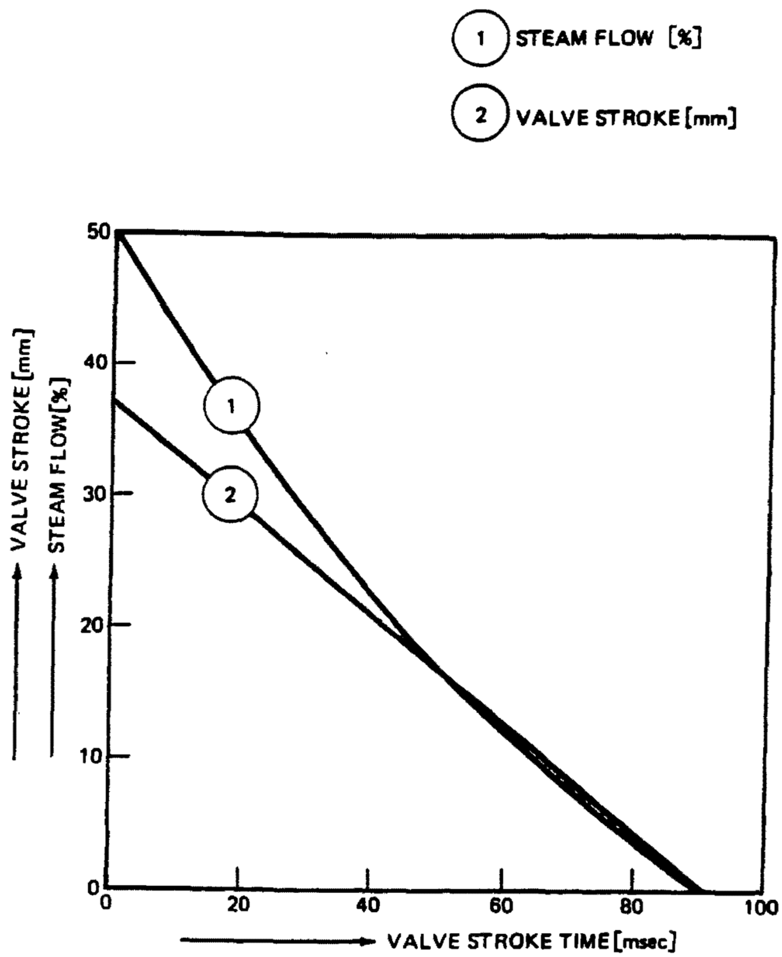
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	STEAM FLOW AND VALVE STROKE VERSUS VALVE STROKE TIME FOLLOWING A LOAD REJECTION FROM 100 PERCENT FIGURE 15.2-19a

CURVE 1 LIMIT-CURVE UNDER THE CONDITION THAT STEAM PRESSURE BEHIND THE CONTROL VALVES REMAINS ON ITS INITIAL VALUE DURING THE STROKE TIME

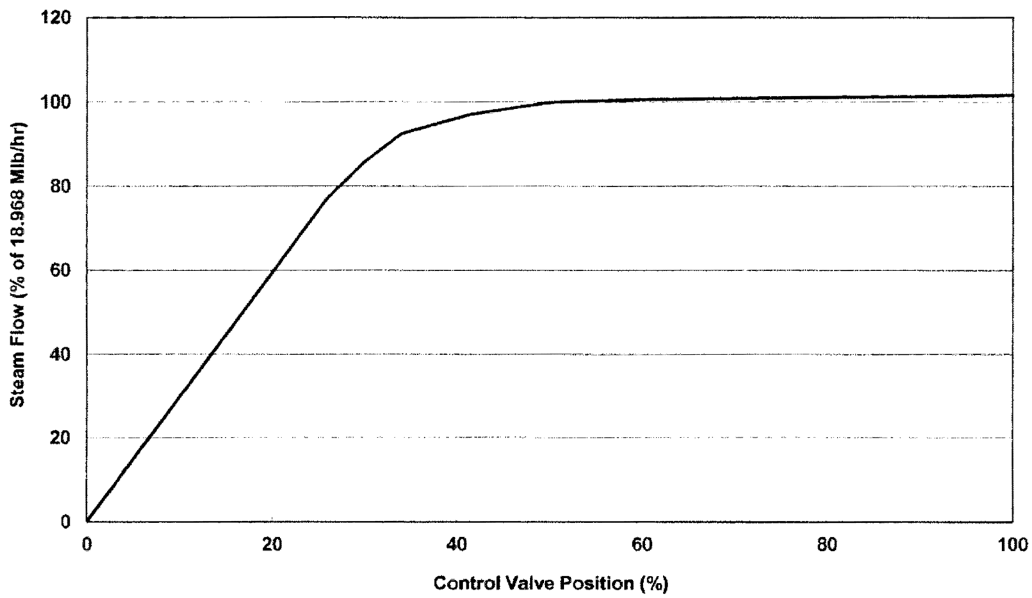
CURVE 2 LIMIT-CURVE UNDER THE CONDITION THAT STEAM PRESSURE BEHIND THE CONTROL VALVES CHANGES ITS VALUE ACCORDING TO THE STEADY-STATE RELATION BETWEEN STEAM FLOW AND FIRST STAGE PRESSURE

CURVE 3 ESTIMATED CURVE REFERRING TO THE MEASUREMENTS DURING LOAD REJECTION TESTS

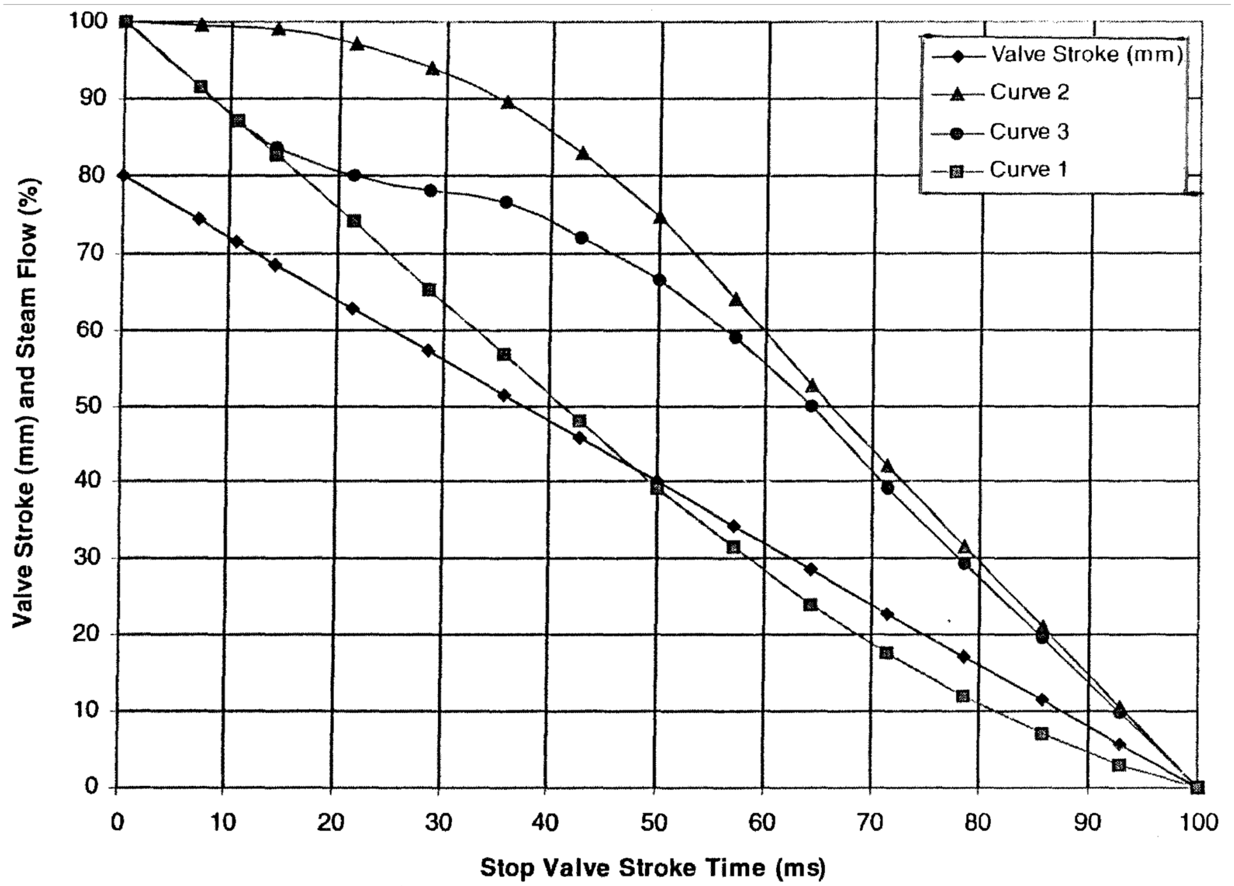
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	LEGEND FOR STEAM FLOW CURVES FIGURE 15.2-19b



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	STEAM FLOW AND VALVE STROKE VERSUS VALVE STROKE TIME FOLLOWING A LOAD REJECTION FROM 50 PERCENT FIGURE 15.2-19c



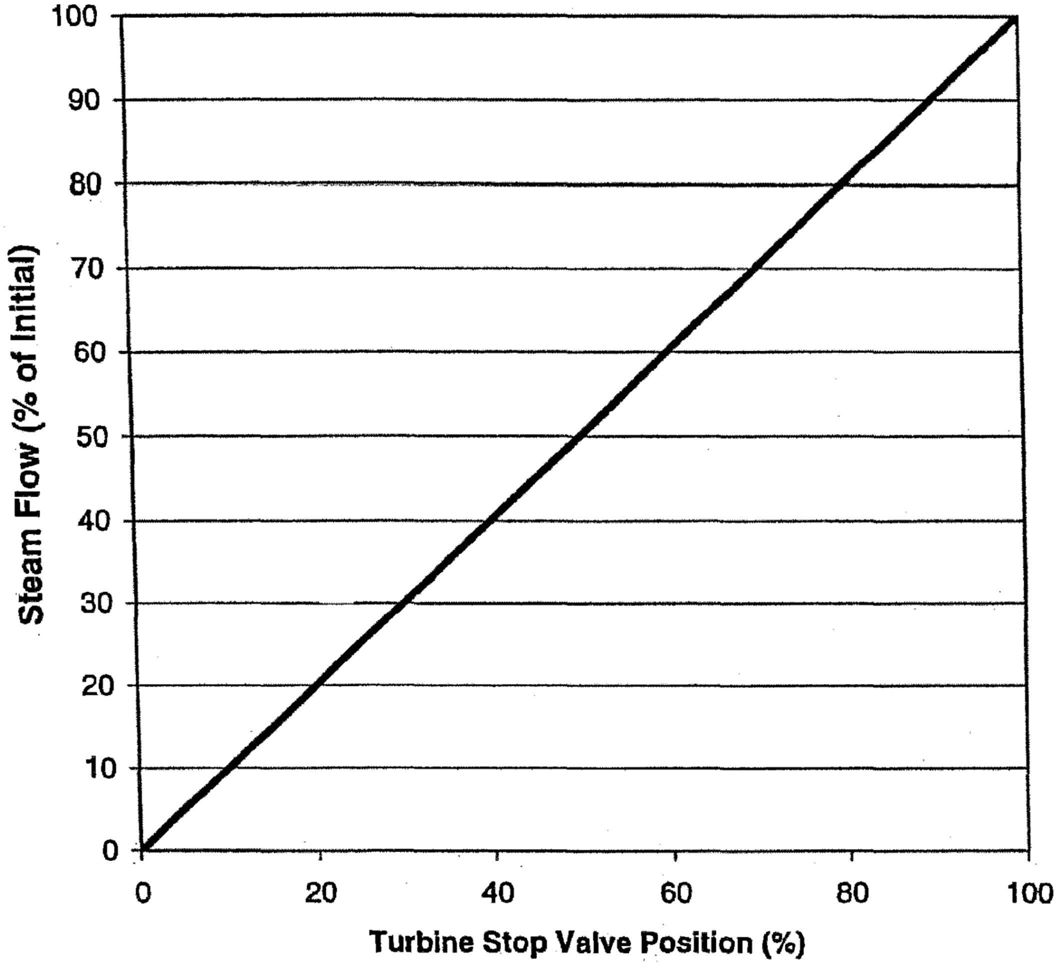
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT
STEAM FLOW VERSUS TURBINE CONTROL VALVE POSITION (CURRENT CYCLE) FIGURE 15.2-19d



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	STEAM FLOW AND MAIN STOP VALVE STROKE VERSUS STOP VALVE STROKE TIME FOLLOWING A LOAD REJECTION FROM 100 PERCENT FIGURE 15.2-20a

CURVE 1	LIMIT CURVE UNDER THE CONDITION THAT STEAM PRESSURE BEHIND THE STOP VALVES REMAINS ON ITS INITIAL VALUE DURING THE STROKE TIME.
CURVE 2	LIMIT CURVE UNDER THE CONDITION THAT STEAM PRESSURE BEHIND THE STOP VALVES CHANGES ITS VALUE ACCORDING TO THE STEADY-STATE RELATION BETWEEN STEAM FLOW AND FIRST STAGE PRESSURE.
CURVE 3	ESTIMATED CURVE BASED ON MEASUREMENTS DURING LOAD REJECTION TESTS.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	LEGEND FOR STEAM FLOW CURVES FIGURE 15.2-20b



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	STEAM FLOW VERSUS TURBINE STOP VALVE POSITION (CURRENT CYCLE) FIGURE 15.2-20c

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

15.3.1 Recirculation Pump Trip

There are two scenarios evaluated for the recirculation pump trip event; the first is a trip of one pump, and the other a trip of two pumps. The reload fuel vendor has determined that these recirculation pump trip scenarios are not limiting events for the current reload cycle. Therefore, this subsection describes the original analysis performed by the NSSS vendor for the initial cycle which remains the current licensing basis for GGNS. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.3.1.1 Identification of Causes and Frequency Classification

15.3.1.1.1 Identification of Causes

Recirculation pump motor operation can be tripped off or tripped to slow speed from high speed by design for intended reduction of other transient core and RCPB effects as well as randomly by unpredictable operational failures. Intentional tripping will occur in response to the events listed in Table 5.4-2.

Random tripping will occur in response to:

a. Operator error

b. Loss of electrical power source to the pumps

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

15.3.1.1.2 Frequency Classification

15.3.1.1.2.1 Trip of One Recirculation Pump

This transient event is categorized as one of moderate frequency.

15.3.1.1.2.2 Trip of Two Recirculation Pumps

This transient event is categorized as one of moderate frequency.

15.3.1.2 Sequence of Events and Systems Operation

15.3.1.2.1 Sequence of Events

15.3.1.2.1.1 Trip of One Recirculation Pump

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

15.3.1.2.1.2 Trip of Two Recirculation Pumps

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

15.3.1.2.1.3 Deleted

15.3.1.2.2 Systems Operation

15.3.1.2.2.1 Trip of One Recirculation Pump

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

15.3.1.2.2.2 Trip of Two Recirculation Pumps

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

Specifically this transient takes credit for vessel level (L9) instrumentation to trip the turbine. Reactor shutdown relies on scram trips from the turbine stop valves. High system pressure is limited by the pressure relief valve system operation.

15.3.1.2.3 The Effect of Single Failures and Operator Errors

15.3.1.2.3.1 Trip of One Recirculation Pump

Since no corrective action is required per subsection 15.3.1.2.2.1, no additional effects of single failures need be discussed. If additional SACF or SOE are assumed (for envelope purposes the other pump is assumed tripped) then the following two pump trip analysis is provided. Refer to Appendix 15A for specific details.

15.3.1.2.3.2 Trip of Two Recirculation Pumps

Table 15.3-2 lists the vessel level (L8 and L9) trip events as the first response to initiate corrective action in this transient. The level (L9) is intended to prohibit moisture carryover to the main turbine. Multiple level sensors are used to sense and detect when the water level reaches the L9 set point. At this point, a single failure will neither initiate nor impede a turbine trip signal. Turbine trip signal transmission circuitry, however, is not built to single failure criterion. The result of a failure at this point would have the effect of delaying the pressurization signature. However, high moisture levels entering the turbine can cause vibration and trip the turbine via turbine supervisory instrumentation.

Scram trip signals from the turbine and L8 are designed such that a single failure will neither initiate nor impede a reactor scram trip initiation. See Appendix 15A for specific details.

15.3.1.3 Core and System Performance

15.3.1.3.1 Mathematical Model

The nonlinear, dynamic model described briefly in subsection 15.1.1.6.3.1 is used to simulate this event.

15.3.1.3.2 Input Parameters and Initial Conditions

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

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

In the analysis of one and two recirculation pump trip transients, a minimum recirculation pump inertia characteristic time of 3 seconds was used which results in a decrease of core flow greater than expected, increasing the coastdown effect. However, a maximum inertia characteristic time of 5 seconds was used for the direct RPT transients, such as turbine generator trip events, in which a slower pump coastdown conservatively represents the protective action of the pump trip. This approach gives the most conservative results in the CPR and peak vessel pressure evaluations for all the applicable transients.

15.3.1.3.3 Results

15.3.1.3.3.1 Trip of One Recirculation Pump

Figure 15.3-1 shows the results of losing one recirculation pump. The tripped loop diffuser flow reverses in approximately 4.5 seconds. However, the ratio of diffuser mass flow to pump mass flow in the active jet pumps increases considerably and produces approximately 155 percent of normal diffuser flow and 70 percent of rated core flow. MCPR does not change significantly thus the fuel thermal limits are not violated. During this transient, level swell is not sufficient to cause turbine trip and scram.

15.3.1.3.3.2 Trip of Two Recirculation Pumps

Figure 15.3-2 shows graphically this transient with minimum specified rotating inertia.MCPR remains unchanged. No scram is initiated directly by pump trip. The vessel water level swell due to rapid flow coastdown is expected to reach the high level trip, thereby shutting down the main turbine and feed pump turbines, and indirectly initiating scrams as a result of the main turbine trip. Subsequent events, such as main steam line isolation and initiation of RCIC and HPCS systems occurring late in this event, have no significant effect on the results.

15.3.1.3.4 Consideration of Uncertainties

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

Actual pump and pump-motor drive line rotating inertias are expected to be somewhat greater than the minimum design values assumed in this simulation. Actual plant deviations regarding inertia are expected to lessen the severity as analyzed. Minimum design inertias were used as well as the least negative void coefficient since the primary interest is in the flow reduction.

15.3.1.4 Barrier Performance

15.3.1.4.1 Trip of One Recirculation Pump

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

15.3.1.4.2 Trip of Two Recirculation Pumps

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

15.3.1.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in subsection 15.2.4.5. Therefore, the radiological exposures noted in subsection 15.2.4.5 cover the consequences of this event.

15.3.2 Recirculation Flow Control Failure - Decreasing Flow

There are two scenarios evaluated for the recirculation flow control failure (decreasing flow) event; the first is fast closure of one main recirculation valve, and the other is fast closure of two main recirculation valves. The reload fuel vendor has determined that these recirculation flow control scenarios are not limiting events for the current reload cycle. Therefore, this subsection describes the original analysis performed by the NSSS vendor for the initial cycle which remains the current licensing basis for GGNS. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.3.2.1 Identification of Causes and Frequency Classification

15.3.2.1.1 Identification of Causes

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

Failure within either loop's controller can result in a maximum valve stroking rate as limited by the capacity of the valve hydraulics.

15.3.2.1.2 Frequency Classification

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

15.3.2.2 Sequence of Events and Systems Operation

15.3.2.2.1 Sequence of Events

15.3.2.2.1.1 Fast Closure of One Main Recirculation Valve

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

15.3.2.2.1.2 Fast Closure of Two Main Recirculation Valves

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

15.3.2.2.1.3 Deleted

15.3.2.2.2 Systems Operation

15.3.2.2.2.1 Fast Closure of One Main Recirculation Valve

Normal plant instrumentation and control is assumed to function. No protection system operation is required.

15.3.2.2.2.2 Fast Closure of Two Main Recirculation Valves

Normal plant instrumentation and control is assumed to function. Credit is taken for scram in response to vessel high water level (L8) trip.

15.3.2.2.3 The Effect of Single Failures and Operator Errors

The single failure and operator error considerations for this event are the same as discussed in Trip of Two Recirculation Pumps, subsection 15.3.1.2.3.2. The fast closure of two recirculation valves instead of one would be the envelope case for the additional SACF or SOE. Refer to Appendix 15A for details.

15.3.2.3 Core and System Performance

15.3.2.3.1 Mathematical Model

The nonlinear dynamic model described briefly in subsection 15.1.1.6.3.1 is used to simulate these transient events.

15.3.2.3.2 Input Parameters and Initial Conditions

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

The less negative void coefficient in Table 15.0-2 was used for these analyses.

15.3.2.3.2.1 Fast Closure of One Main Recirculation Valve

The design specification for the recirculation flow control valve is such that any single loop valve controller failure cannot cause a stroking rate >30 percent per second in the opening direction and >60 percent per second in the closing direction.

These restrictions are based on the valve hydraulic characteristics. For a master controller malfunction, the design requirement is such that the maximum valve stroking rate for valve fast opening or closing in both loops is limited by each individual flow limiter by 11 percent per second. The above stroking rates have been confirmed from field observations.

Failure within either loop controller can result in a maximum stroking rate of 60 percent per second as limited by the valve hydraulics.

15.3.2.3.2.2 Fast Closure of Two Main Recirculation Valves

System limits are described in Section 15.3.2.1.1. Recirculation loop flow is allowed to decrease to approximately 25 percent of minimum. This is the flow expected when the flow control valves are maintained at a minimum open position.

15.3.2.3.3 Results

15.3.2.3.3.1 Fast Closure of One Recirculation Valve

Figure 15.3-3 illustrates the maximum valve stroking rate which is limited by hydraulic means. It is similar in most respects to the trip of one recirculation pump transient. Design of the hydraulic limit on maximum valve stroking rate is intended to make this transient event less severe than the one pump trip, and fuel thermal limits are not threatened.

Transients resulting in decreasing flow (e.g. pump trips, valve closures, etc.) consequentially lead to a decrease in power. This power change is seen as a change in surface heat flux on the fuel. The surface heat flux does not change as rapidly as the change in core flow. Voids form due to the reduced flow and because surface heat flux does not drop off as fast as flow. Vessel water level increases because of the higher void content and the mismatch between the feedwater flow and core inlet flow. This mismatch occurs because the single recirculation valve has closed but the feedwater system is maintaining close to the original flow. The feedwater system receives feedback from the steam flow, i.e., reduction in steam flow signals the feedwater system to reduce flow. Determination of the resultant vessel water level (reaching level 8 or not) is dependent upon all the aforementioned variables and no consistent sequence of events can be expected for different product lines. The analysis, as presented, incorporates all the plant uniqueness and accurately depicts the event for the Grand Gulf design.

15.3.2.3.3.2 Fast Closure of Two Recirculation Valves

Figure 15.3-4 illustrates the expected transient which is similar to a two-pump trip. This analysis is very similar to the two-pump trip described in subsection 15.3.1. Design of limiter operation is intended to render this transient to be less severe than the two-pump trip. MCPR remains greater than the safety limit, therefore, no fuel damage occurs.

15.3.2.3.4 Consideration of Uncertainties

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

These analyses unlike the pump trip series will be unaffected by deviations in pump/pump motor and driveline inertias since it is the main valve that causes rapid recirculation decreases.

15.3.2.4 Barrier Performance

15.3.2.4.1 Fast Closure of One Recirculation Valve

Figure 15.3-3 indicates a reduction in system pressure and no increases are expected.

15.3.2.4.2 Fast Closure of Two Recirculation Valves

The narrow-range level rises to the high level trip set points (L8 and L9) causing scram and trip of the feedwater pumps and main turbine. Safety/relief valves open in the pressure relief mode and briefly discharge steam to the suppression pool. Pressure in the vessel bottom is limited to 1170 psig, well below the ASME code limit. At approximately 30 seconds, the wide range level falls to the low water level trip set point, causing trip of the recirculation pumps and initiation of HPCS and RCIC system. However, there is a delay of up to 30 seconds before the water supply from HPCS and RCIC system enters the vessel.

15.3.2.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in subsection 15.2.4.5. Therefore, the radiological exposures noted in subsection 15.2.4.5 cover the consequences of this event.

15.3.3 Recirculation Pump Seizure

There are two scenarios evaluated for the recirculation pump seizure event; one for two loop operations, and the other for single loop operations. The pump seizure accident during two loop operation has been shown to result in a relatively mild plant response and is not reanalyzed each fuel cycle. The recirculation pump seizure during single loop operations is a potentially limiting event. This subsection discusses both analyses; 1) the analysis performed by the NSSS vendor for two loop operations for the initial fuel cycle which remains the currents analysis for this event, and 2) the analysis performed by the reload fuel vendor for single loop operations for the current fuel cycle. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload fuel vendor for the current cycle, refer to Section 15.0.

15.3.3.1 Identification of Causes and Frequency Classification

The seizure of a recirculation pump is considered in philosophical, probability and functional senses as a design basis accident event. It has been evaluated as being a very mild accident in relation to other design basis accidents such as the LOCA. The analysis has been conducted with consideration to a single or two loop operation.

The recirculation pump is designed to very rigid standards and codes. It is very well instrumented, monitored, and controlled to assure safe and orderly operation. It is designed to meet strict seismic and environmental conditions. It is protected from external disturbance which could negate its inherent capabilities to preclude a self-destruction (seizure or shaft impairment). Refer to Section 5.1 for specific mechanical considerations and Chapter 7 for electrical aspects.

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

[HISTORICAL INFORMATION] [Recirculation pump seizure is analyzed as an infrequent incident for which credit is appropriately taken for use of specifically accident qualified equipment to help terminate the event. The results of infrequent events are compared to more stringent requirements (10 percent of 10 CFR 50.67) than the less frequent category of accidents (10 CFR 50.67), for which only safety-grade equipment credit is given. (See revised Figure 15A.2-3.) If analyzed on the 10 CFR 50.67 basis of credit for only safety systems, this would become a mild, nonlimiting, insignificant accident. However, the anticipated event frequency indicates that the more restrictive infrequent event category as presented in the FSAR is reasonable with allowance given for nonaccident environment qualified equipment/ systems, since the conditions of the event are not related in any way to the hostile and coincident events of accident scenarios. This approach is consistent with all previous review and approval by NRC on previous BWRs.]

15.3.3.1.1 Identification of Causes

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

15.3.3.1.2 Frequency Classification

This event is considered to be a limiting fault but results in effects which can easily satisfy more numerous event limits (i.e., infrequent incident classification).

15.3.3.2 Sequence of Events and Systems Operations

15.3.3.2.1 Sequence of Events

Table 15.3-5 lists the sequence of events for Figure 15.3-5 (Seizure of Recirculation Pump During Two Loop Operation).

Table 15.3-5a lists the sequence of events for Figure 15.3-5A (Seizure of Recirculation Pump During Single Loop Operation).

15.3.3.2.1.1 Deleted

15.3.3.2.2 Systems Operation

The pump seizure is a postulated accident where the operating recirculation pump suddenly stops rotating. This causes a rapid decrease in core flow, a decrease in the rate at which heat can be transferred from the fuel rods and a decrease in the critical power ratio.

In order to properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection, and reactor protection systems for the pump seizure event during Two Loop Operation.

The pump seizure accident analysis during single loop operation, evaluated by the reload fuel vendor, assumes that turbine bypass is unavailable.

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

The event severity of a coincident loss of offsite power with the postulated recirculation pump seizure accident is bounded by the analysis of “Loss of ac Power” as shown in subsection 15.2.6. The only difference in these two events is the core flow coastdown rate.The flow coastdown rate during the pump seizure event coincident with a loss of offsite power is faster than that during the loss of ac power transient. Coincident loss of ac power causes this accident to be a pressurization event. The faster flow coastdown for a pressurization event will result in a less severe thermal power transient due to a negative void reactivity coefficient.

[HISTORICAL INFORMATION] [The “recirculation pump seizure” accident was addressed by the Licensing Review Group, Issue RSB-21. A summary of the resolution of that issue is discussed below.

The recirculation pump seizure event is considered to be an extremely unlikely event and as such falls into the category generally classified as an accident. The event is evaluated as a limiting fault. The potential effects of the hypothetical pump seizure “accident” are very conservatively bounded by the effects of the DBA-LOCA.

This is easily verified by comparison of the two events. In both accidents, the recirculation driving-loop flow decreases extremely rapidly. In the case of seizure, stoppage of the active pump(s) occurs; for the DBA-LOCA, the severance of the line has a similar, but more rapid and severe, influence. Following a pump seizure event, water level is maintained, the core remains submerged, and this provides a continuous core cooling mechanism. However, for the DBA-LOCA, complete flow stoppage occurs and water level decreases due to loss of coolant, thus resulting in uncovery of the reactor core and subsequent overheating of the fuel rod cladding. Also, complete depressurization occurs with the DBA-LOCA, while reactor pressure does not significantly decrease for the pump seizure event. Clearly, the increased temperature of the fuel cladding and the reduced reactor pressure for the DBA-LOCA both combine to yield a much more severe stress and potential for cladding perforation for the DBA-LOCA than for the pump seizure. Therefore, it is concluded that the potential effects of the hypothetical pump seizure accident are very conservatively bounded by the effects of the DBA-LOCA. The following is provided to show the impact of not taking credit for non-safety grade equipment to terminate this event.

a. Level 9 Turbine Trip

The FSAR analysis of the pump seizure event during Two Loop Operation assumes that the vessel water level swell due to pump seizure will cause high water level (Level 9) trips of the main turbine and the feedwater pumps. The safety grade Level 8 trip initiates a reactor scram directly. In the case of the pump seizure without an L9 trip during Two Loop Operation, the event is less severe than the analysis in the FSAR with the L9 trip for the following reason: a pump seizure, should it occur, would result in core flow reduction which reduces the core power and surface heat flux due to the effect of the negative void reactivity coefficient. A turbine trip would cause isolation, which in turn would cause void collapse and slightly increase power. Therefore, a loss of Level 9 trip would result in a less severe event consequence to the fuel than that depicted in subsection 15.3.1.2.

b. Main Turbine Bypass System

As a result of the NRC's concern respecting reactivity effects of pressure transients, GE and the NRC met on November 20 and 21, 1978 for a comprehensive review of turbine trip and load reject transients without bypass.

The principal conclusion of that meeting was that the most limiting BWR transient event which takes credit for non-safety grade equipment is the feedwater controller failure. Analysis indicates that a CPR increase of approximately 0.06 applies to this transient without a functioning main turbine bypass system.

For recirculation pump seizure with a failure of turbine bypass system, the increase of CPR would be less than that for the feedwater controller failure for the following reason. As this event occurs, the reactor power drops significantly within the first 2 seconds due to decreased core flow. Therefore, by the time of turbine trip the reactor power is at a low level. The core power is the main parameter which relates to the fuel thermal limit. The effect of failure of the main turbine bypass system to stop the steam flow retains pressure on the core but contributes only a small positive reactivity feedback. This is a secondary effect of much less significance than the reactivity decrease due to fluid flow decreasing through the core. This increase of core power is more severe for feedwater controller failure (increasing) event than for a recirculation pump failure because it occurs at a higher power level.

c. Relief Function of Safety/Relief Valves

The contribution of MCPR from taking credit for the relief function rather than the safety function of safety/relief valves is not significant because the MCPR always reaches its lowest value before opening of the relief valves.

Analyses of recirculation pump seizure by the NSSS vendor for Two Loop Operation where coolant flow rate drops rapidly have shown that MCPR does not decrease significantly before fuel surface heat flux begins dropping enough to restore greater thermal margins as a plant intrinsically responds to the reduced flow rate. The effect of not taking credit for non-safety grade equipment is a CPR increase of 0.06. Therefore, the MCPR for pump seizure event is still well above the safety limit of 1.06.]

15.3.3.2.3 The Effect of Single Failures and Operator Errors

Single failures in the scram logic originating via the high vessel level (L8) trip are similar to the considerations in subsection 15.3.1.2.3.2, Trip of Two Recirculation Pumps.

Refer to Appendix 15A for further details.

15.3.3.3 Core and System Performance

15.3.3.3.1 Mathematical Model

The nonlinear dynamic models described briefly in Section 15.0 are used to simulate this event.

REDY was used by the NSSS vendor for the pump seizure accident during two loop operation. For the current cycle, ODYN, ISCOR, and TASC are used to calculate the MCPR for the fuel during a pump seizure during single loop operation.

15.3.3.3.2 Input Parameters and Initial Conditions

The analysis for Two Loop Operation has been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

The void coefficient is adjusted to the most conservative value, that is, the least negative value for the Two Loop Operation analysis.

The analysis for Single Loop Operation was performed consistent with plant conditions in Table 15.0-4 for single loop operating conditions.

For the purpose of evaluating consequences to the fuel thermal limits, this transient event is assumed to occur as a consequence of an unspecified, instantaneous stoppage of one recirculation pump shaft while the reactor is operating at 105 percent of the initially licensed NB rated power for Two Loop Operation and 72.2% core power for Single Loop Operations. Also, the reactor is assumed to be operating at thermally limited conditions.

15.3.3.3.3 Results

Figure 15.3-5 presents the results of the accident for the Two Loop Operation analysis. Core coolant flow drops rapidly, reaching its minimum value in approximately 15 seconds. The level swell produces a trip of the main and feedwater turbines and scram since heat flux decreases much more rapidly than the rate at which heat is removed by the coolant. The scram conditions impose no threat to thermal limits. Additionally, the momentary opening of the bypass valves and some of the safety/relief valves limit the pressure well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

Figure 15.3-5A present the results for the pump seizure event during Single Loop Operation. The core coolant flow drops from 54.1% of the rated value to 30% in less than 1.0 second. Thermal hydraulic analysis using the reload fuel vendor's methodology has shown that the two loop MCPRP limit provides the required protection below 70% of rated core power so that the MCPR remains greater than the reference SLO Safety Limit MCPR.

15.3.3.3.3.1 Considerations of Uncertainties

Considerations of uncertainties are included in the GETAB analysis for the NSSS vendor's analysis for Two Loop Operation.

Uncertainties used in the analysis by the reload fuel vendor are addressed in the cycle-specific reload documentation referenced in Section 15.0.

15.3.3.4 Barrier Performance

For the NSSS vendor's analysis of pump seizure during Two Loop Operation, the bypass valves and momentary opening of some of the safety/ relief valves limit the pressure well within the range allowed by the ASME vessel code. No valve openings occur for pump seizure during Single Loop Operation. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

15.3.3.5 Radiological Consequences

While the consequences of a pump seizure during two loop operation do not result in any fuel failure, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in subsection 15.2.4.5. Therefore, the radiological exposures noted in subsection 15.2.4.5 cover the consequences of this event.

The reload fuel vendor's cycle independent analysis for pump seizure during Single Loop Operation shows that the CPR associated with this event does not fall below the safety limit MCPR. Therefore, no fuel failures are postulated (see References 1 and 4) and the radiological exposures noted in subsection 15.2.4.5 cover the consequences of this event.

15.3.4 Recirculation Pump Shaft Break

The reload fuel vendor has determined that the recirculation pump shaft break event is not limiting event for the current reload cycle. Therefore, this subsection describes the original analysis performed by the NSSS vendor for the initial cycle which remains the current licensing basis for GGNS. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.3.4.1 Identification of Causes and Frequency Classification

The breaking of the shaft of a recirculation pump is considered in philosophical, probability, and functional senses as a design basis accident event. It has been evaluated as a very mild accident in relation to other design basis accidents such as the LOCA. The analysis has been conducted with consideration to a single or two loop operation.

The recirculation pump is designed to very rigid standards and codes. It is very well instrumented, monitored, and controlled to assure safe and orderly operation. It is designed to meet strict seismic and environmental conditions. It is protected from external disturbance which could negate its inherent capabilities to preclude self-destruction (shaft breakage). Refer to Chapter 5 for specific mechanical considerations and Chapter 7 for electrical aspects.

The shaft shearing event postulated certainly would not be the mode failure of such a device. Safe shutdown components (e.g., electrical breakers protective circuits) would preclude an instantaneous seizure event.

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

15.3.4.1.1 Identification of Causes

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

15.3.4.1.2 Frequency Classification

This event is considered to be a limiting fault but results in effects which can easily satisfy a more numerous event limit (i.e., infrequent incident classification).

15.3.4.2 Sequence of Events and Systems Operations

15.3.4.2.1 Sequence of Events

A postulated instantaneous break of the pump motor shaft of one recirculation pump as discussed in subsection 15.3.4.1.1 will cause the core flow to decrease rapidly, resulting in water level swell in the reactor vessel. When the vessel water level reaches the high water level setpoint (Level 9), main turbine trip and feedwater pump trip will be initiated. Subsequently, reactor scram and the remaining recirculation pump trip will be initiated due to the turbine trip. Eventually, the vessel water level will be controlled by HPCS and RCIC flow.

15.3.4.2.1.1 Deleted

15.3.4.2.2 Systems Operation

Normal operation of plant instrumentation and control is assumed. This event takes credit for vessel water level (Levels 8 and 9) instrumentation to scram the reactor and trip the main turbine and feedwater pumps. High system pressure is limited by the pressure relief system operation.

Operation of HPCS and RCIC features is expected in order to maintain adequate water level control.

15.3.4.2.3 The Effect of Single Failures and Operator Errors

Effects of single failures in high vessel level (L8 and L9) trip are similar to the considerations in subsection 15.3.1.2.3.2, Trip of Two Recirculation Pumps.

Assumption of SACF or SOE in other equipment has been examined and this has led to the conclusion that no other credible failure exists for this event. Therefore the bounding case has been considered.

Refer to Appendix 15A for more details.

15.3.4.3 Core and System Performance

Since this event is less limiting than the event in subsection 15.3.3, only qualitative evaluation is provided. Therefore no discussion of mathematical mode, input parameters, and consideration on uncertainties, etc., is necessary.

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

The severity of this pump shaft break event is bounded by the pump seizure event (see subsection 15.3.3). This can be demonstrated easily by consideration of these two events. In either of these two events, the recirculation drive flow of the affected loop decreases rapidly. In the case of the pump seizure event, the loop flow decreases faster than the normal flow coastdown as a result of the large hydraulic resistance introduced by the stopped rotor. For the pump shaft break event, the hydraulic resistance caused by the broken pump shaft is less than that of the stopped rotor for the pump seizure event. Therefore, the core flow decrease following a pump shaft break effect is slower than the pump seizure event. Thus, it can be concluded that the potential effects of the hypothetical pump shaft break accident are bounded by the effects of the pump seizure event.

15.3.4.4 Barrier Performance

The bypass valves and momentary opening of some of the safety/relief valves limit the pressure well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

15.3.4.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in subsection 15.2.4.5. Therefore, the radiological exposures noted in subsection 15.2.4.5 cover the consequences of this event.

15.3.5 References

1. ECH-NE-10-00021, Rev. 5, “GNF2 Fuel Design Cycle - Independent Analyses for Grand Gulf Nuclear Station”, Mar. 2020.

2. Deleted

3. Deleted

4. ECH-NE-20-00006, Rev. 0, “GGNS Fuel Design Cycle-Independent Analyses for Grand Gulf Nuclear Station”, Mar. 2020.

TABLE 15.3-1: SEQUENCE OF EVENTS FOR RECIRCULATION PUMP TRIP, ONE PUMP (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT) (FIGURE 15.3-1)

Time-sec	Event
0	Trip of one recirculation pump initiated.
4.5	Jet pump diffuser flow reverses in the tripped loop.
20	Core flow and power level stabilize at new equilibrium conditions.

TABLE 15.3-2: SEQUENCE OF EVENTS FOR RECIRCULATION PUMP TRIP, BOTH PUMPS (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT)
(FIGURE 15.3-2)

Time-sec	Event
0	Trip of both recirculation pumps initiated.
3.0	Vessel water level (L8) trip initiates scram, turbine trip and feedwater pump trip.
3.1	Turbine trip initiates bypass operation.
4.6	Safety/relief valves open due to high pressure.
11.4	Safety/relief valves close.
24.7	Vessel water level (L2) set point reached.
54.7(est)	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.3-3: SEQUENCE OF EVENTS FOR FAST CLOSURE OF RECIRCULATION VALVES, (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT)
ONE VALVE (FIGURE 15.3-3)

Time-sec	Event
0	Initiate fast closure of one main recirculation valve.
1.6	Jet pump diffuser flow reverses in the affected loop.
40	Core flow and power approach new equilibrium conditions.

TABLE 15.3-4: SEQUENCE OF EVENTS FOR FAST CLOSURE OF RECIRCULATION VALVES, (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT) BOTH VALVES (FIGURE 15.3-4)

Time-sec	Event
0	Initiate fast closure of both main recirculation valves.
7.62	Vessel level (L8) trip initiates scram and turbine trip.
7.62	Feedwater pumps tripped off.
7.72	Turbine trip initiates bypass operation.
7.76	Recirculation pumps trip due to turbine trip.
10.69	Safety/relief valves actuate due to high pressure.
16.6	Safety/relief valves close.
31.8	Vessel water level reaches Level 2 set point.
61.8(est)	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.3-5: SEQUENCE OF EVENTS FOR SEIZURE OF RECIRCULATION PUMP DURING TWO LOOP OPERATION (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT) (FIGURE 15.3-5)

Time-sec	Event
0	Single pump seizure was initiated.
0.6	Jet pump diffuser flow reverses in seized loop.
1.9	Vessel level (L8) trip initiates scram.
1.9	Vessel level (L8) trip initiates turbine trip.
1.9	Feedwater pumps are tripped off.
2.0	Turbine trip initiates bypass operation.
2.0	Turbine trip initiates recirculation pumps trip.
3.6	Safety/relief valves open due to high pressure.
11.0	Safety/relief valves close.
24.6	Main bypass valves close to regain pressure regulator control.
27.6	Vessel water level reaches Level 2 (L2) set point.
57.6(est)	HPCS/RCIC flow enters the vessel (not simulated).

TABLE 15.3-5A: SEQUENCE OF EVENTS FOR SEIZURE OF RECIRCULATION PUMP DURING SINGLE LOOP OPERATION (FIGURE 15.3-5A)

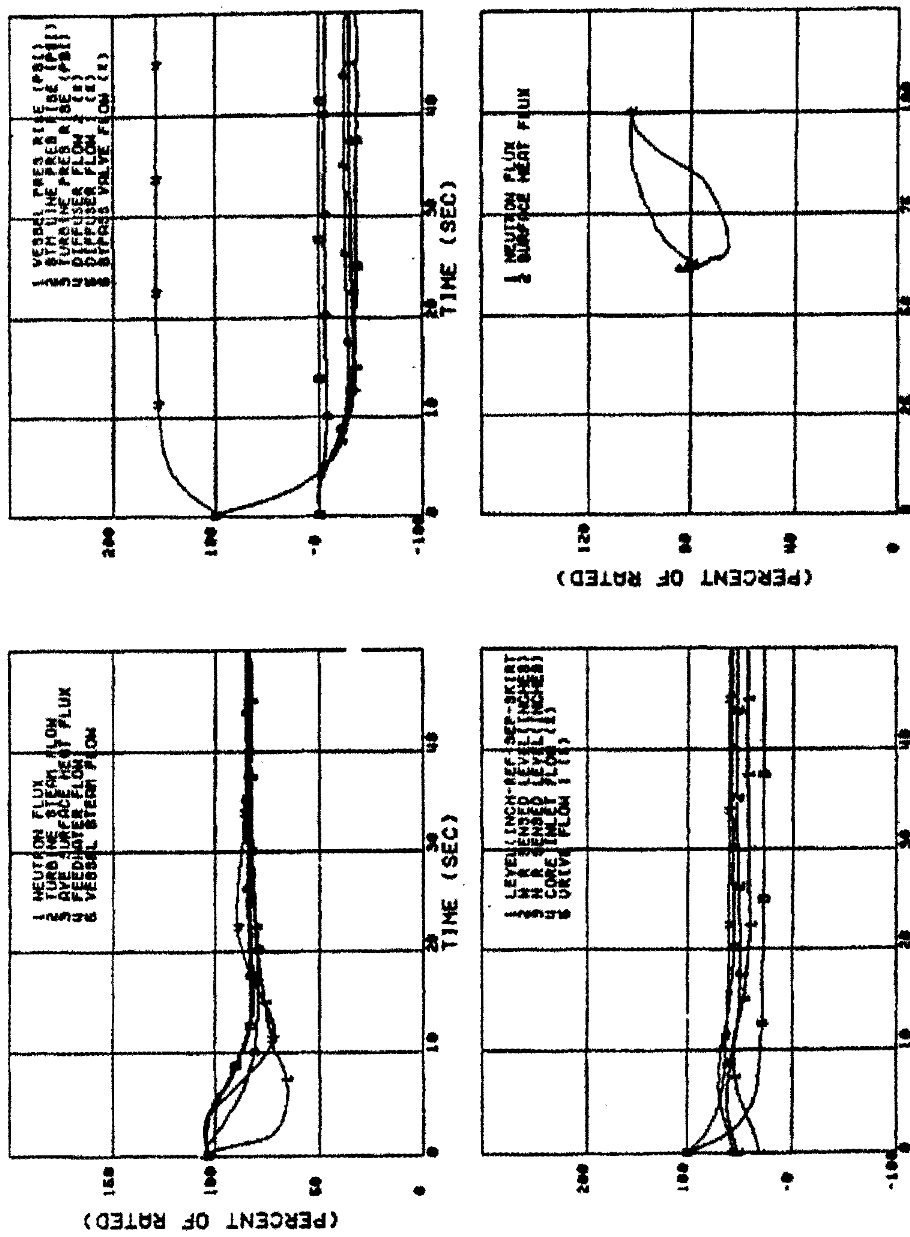
Time-sec	Event
0.0	Single pump seizure initiated
1.35	Minimum core flow
1.63	Minimum power
2.14	Maximum ΔCPR
6.0	Final power/flow 44.2%/32%[4]
NA[1]	Recirculating pump trip
NA[2]	Open first S/RV set
NA[3]	Open bypass valves

[1] Active loop recirculation pump seizes for this event.

[2] S/RV's did not open for this event.

[3] Bypass valves are not credited for this event.

[4] Although this event may place the Plant in an instability region requiring a scram, this is not included in the analysis since the minimum CPR occurs early in the event.



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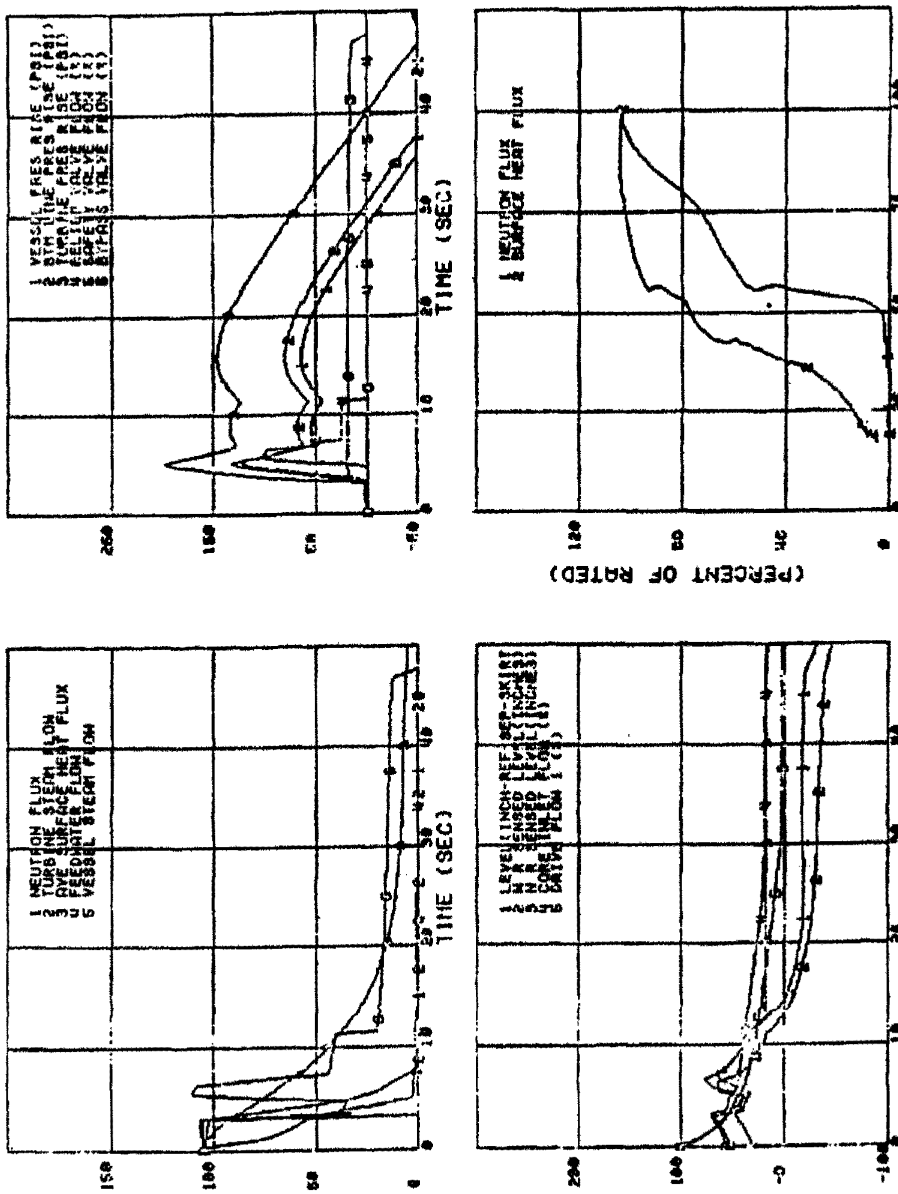
TRIP OF ONE RECIRCULATION PUMP MOTOR
(INITIAL CYCLE)

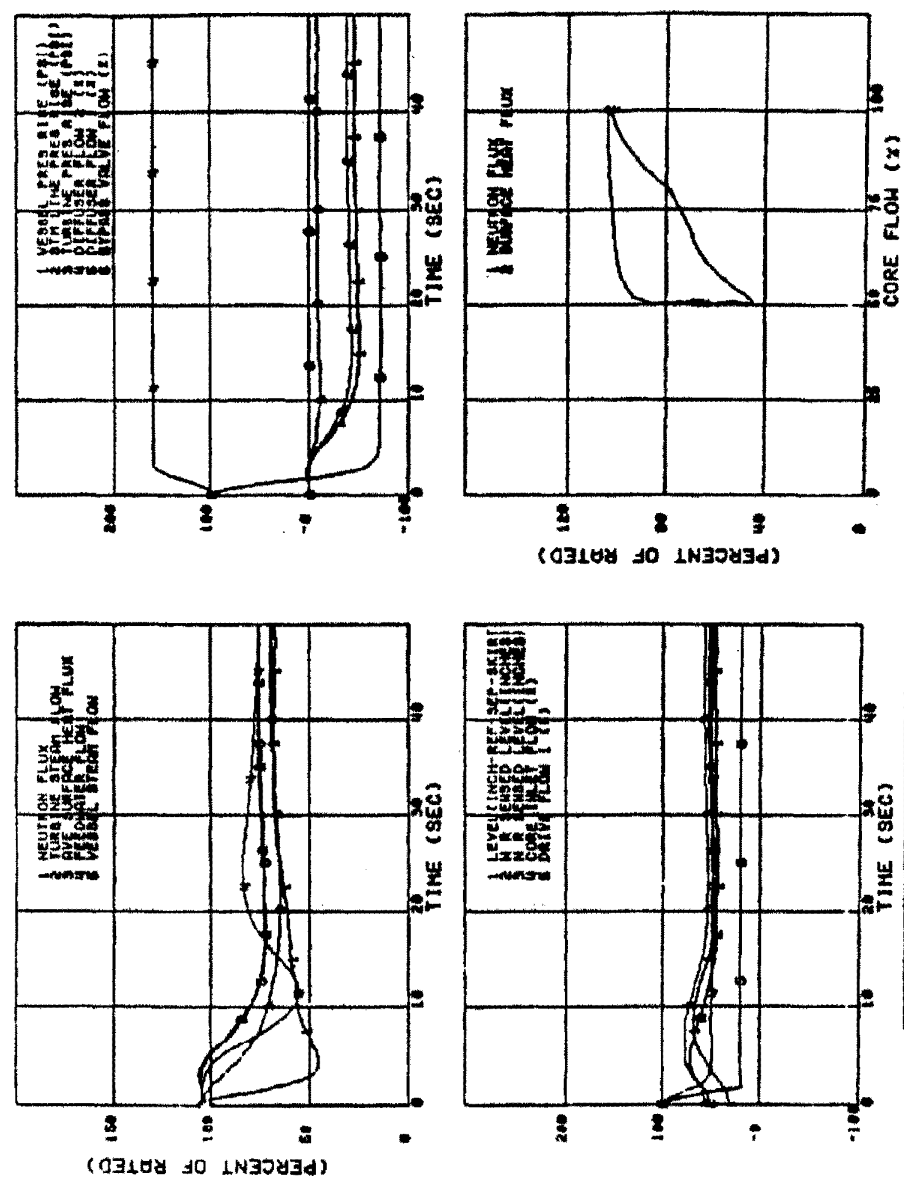
FIGURE 15.3-1

Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW, but remains the current analysis of record for this event.



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	TRIP OF BOTH RECIRCULATION PUMP MOTORS (INITIAL CYCLE) FIGURE 15.3-2

Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW, but remains the current analysis of record for this event.



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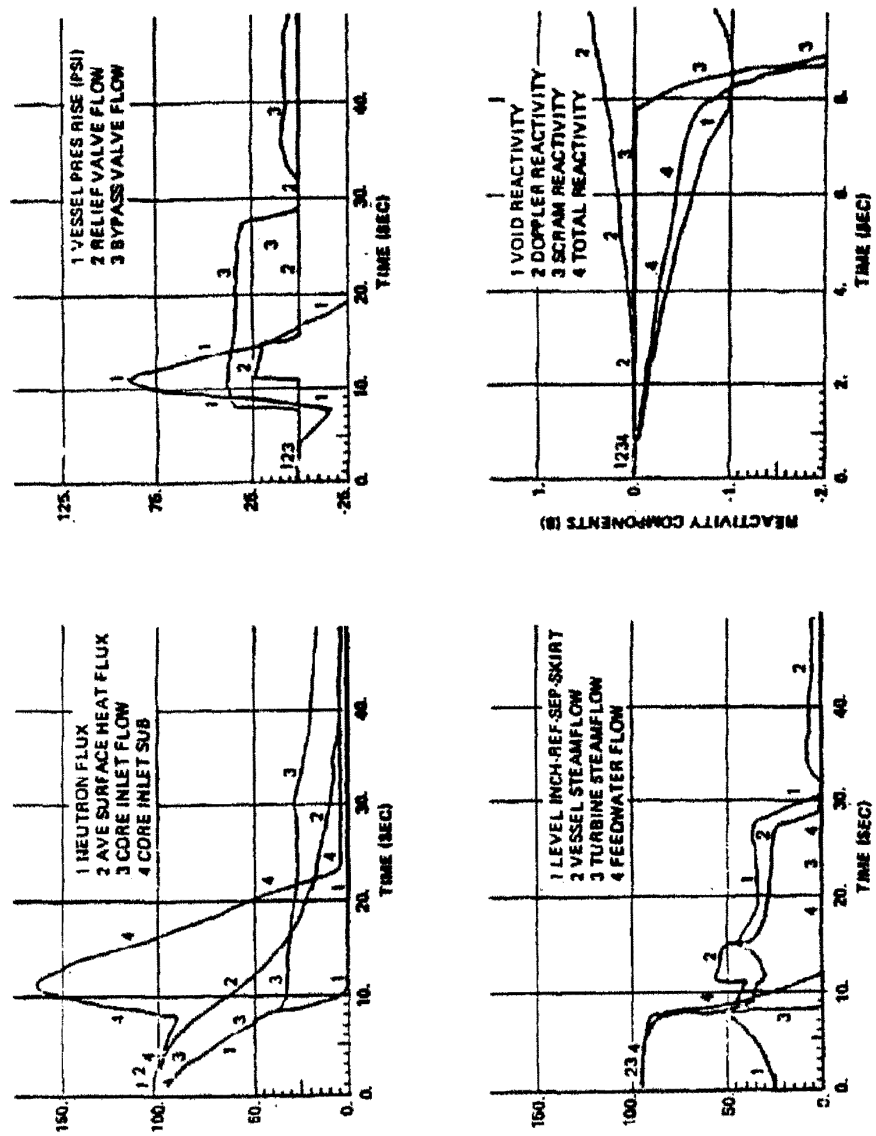
FAST CLOSURE OF ONE MAIN RECIRCULATION VALVE AT 60% PER SECOND
(INITIAL CYCLE)

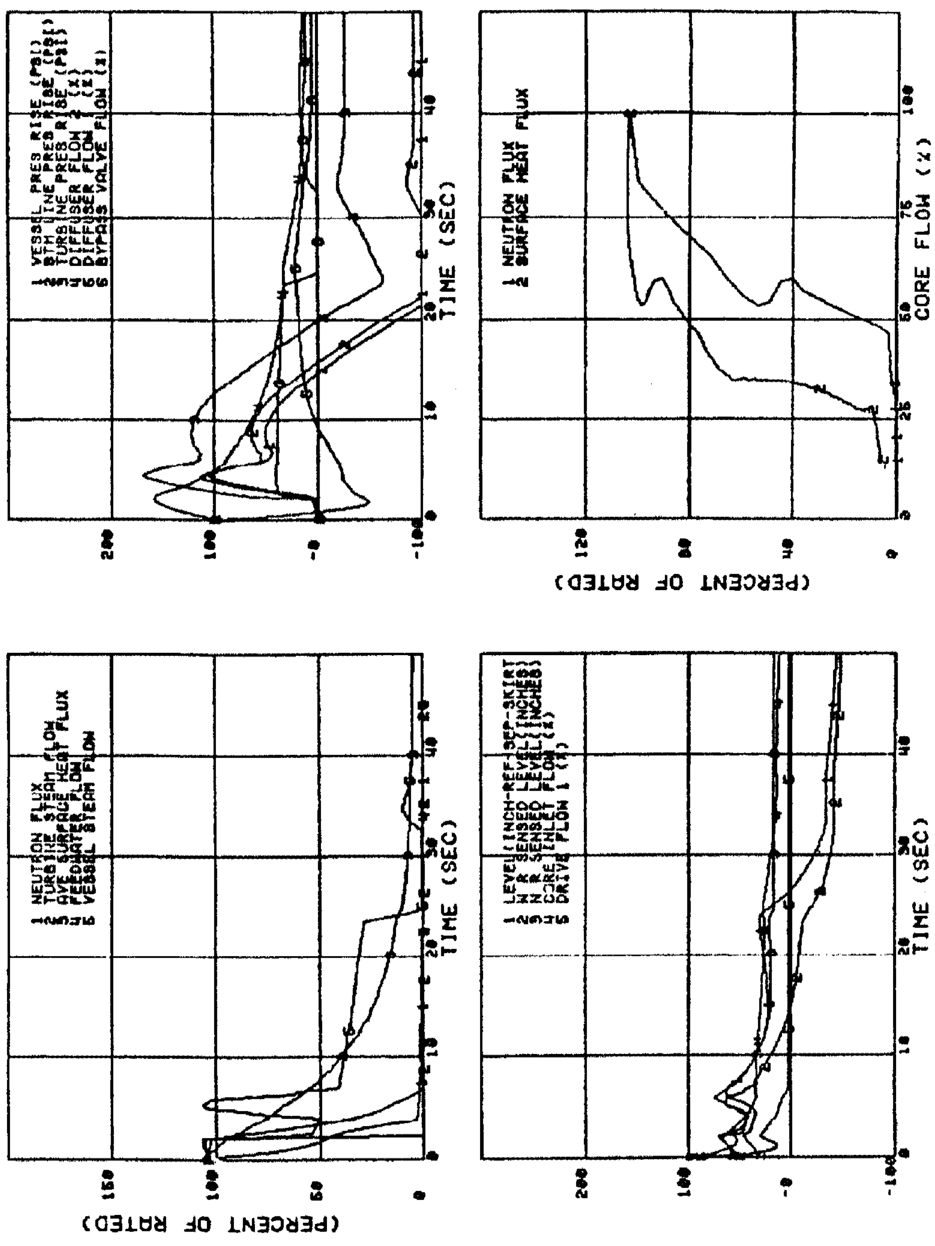
FIGURE 15.3-3

Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW, but remains the current analysis of record for this event.



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FAST CLOSURE OF BOTH MAIN RECIRCULATION VALVES AT 11% PER SECOND (INITIAL CYCLE) FIGURE 15.3-4

Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW, but remains the current analysis of record for this event.



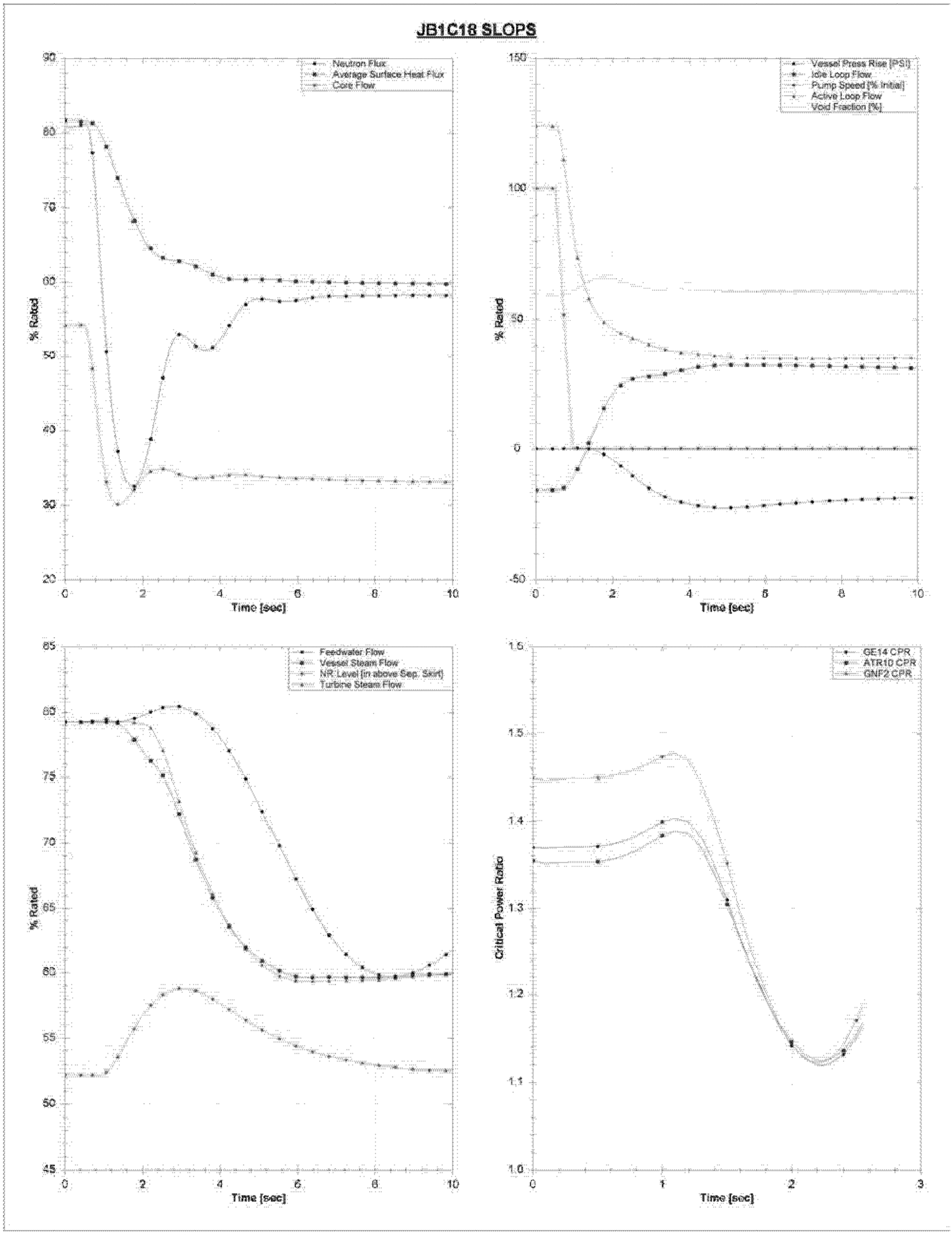
GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

SIEZURE OF ONE RECIRCULATION PUMP
DURING TWO-LOOP OPERATION

(INITIAL CYCLE)

FIGURE 15.3-5

Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW, but remains the current analysis of record for this event.



Seizure of a Recirculation Pump During Single Loop Operation Key Parameters
Figure 15.3-5A

Figure 15.3-5B through 15.3-5D

Deleted

Figure 15.3-5E through 15.3-5K

Deleted

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15.4.1 Rod Withdrawal Error - Low Power

The reload fuel vendor has determined the rod withdrawal error - low power event is not a limiting event for the current reload cycle. Therefore, this subsection describes the original analysis performed by the NSSS vendor for the initial cycle which remains the current licensing basis for GGNS. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.4.1.1 Control Rod Removal Error During Refueling

15.4.1.1.1 Identification of Causes and Frequency Classification

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

15.4.1.1.2 Sequence of Events and Systems Operation

15.4.1.1.2.1 Initial Control Rod Removal

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

15.4.1.1.2.2 Fuel Insertion With Control Rod Removed

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

15.4.1.1.2.3 Second Control Removal

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

15.4.1.1.2.4 Control Rod Removal Without Fuel Removal

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

15.4.1.1.2.5 Deleted

15.4.1.1.2.6 Effect of Single Failure and Operator Errors

If any one of the operations involved in initial failure or error is followed by any other SACF or SOE, the necessary safety actions are taken (e.g., rod block or scram) automatically prior to limit violation. Refer to Appendix 15A for details.

15.4.1.1.3 Core and System Performances

Since the probability of inadvertent criticality during refueling is precluded, the core and system performances were not analyzed. However, it is well known that withdrawal of the highest worth control rod during refueling results in a positive reactivity insertion but not enough to cause criticality. (See subsection 4.3.2 for a description of the methods and results of the shutdown margin analysis.) Additional reactivity insertion is precluded by interlocks. (See subsection 7.6.1.1.) As a result, no radioactive material is ever released from the fuel making it unnecessary to assess any radiological consequences.

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

15.4.1.1.4 Barrier Performance

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

15.4.1.1.5 Radiological Consequences

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

15.4.1.2 Continuous Rod Withdrawal During Reactor Startup

15.4.1.2.1 Identification of Causes and Frequency Classification

The probability of the initial causes or error of this event alone is considered low enough to warrant its being categorized as an infrequent incident. The probability of further single failures postulated for this event is even considerably lower because it is contingent upon the simultaneous failure of two redundant inputs to the rod control and information system (RCIS), concurrent with a high worth rod, out-of-sequence rod selection, plus operator ignorance of continuous alarm annunciations prior to safety system actuations.

15.4.1.2.2 Sequence of Events and Systems Operation

15.4.1.2.2.1 Sequence of Events

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

Continuous control rod withdrawal errors during reactor startup are precluded by the RCIS. The RCIS prevents the withdrawal of an out-of-sequence control rod in the 100 percent to 75 percent control rod density range and limits rod movement to the banked position mode of rod withdrawal from the 75 percent rod density to the preset power level. Since only in-sequence control rods can be withdrawn in the 100 percent to 75 percent control rod density and control rods are withdrawn in the banked position mode from the 75 percent control rod density point to the preset power level, there is no basis for the continuous control rod withdrawal error in the startup and low power range. See subsection 15.4.2 for description of continuous control rod withdrawal above the preset power level. The bank position mode of RCIS is described in GEH Topical Report NEDO-21231.

15.4.1.2.2.2 Deleted

15.4.1.2.2.3 Effects of Single Failure and Operator Errors

If any one of the operations involved the initial failure or error and is followed by another SACF or SOE, the necessary safety actions are taken (e.g., rod blocks) prior to any limit violation. Refer to Appendix 15A for details.

15.4.1.2.3 Core and System Performance

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

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

15.4.1.2.4 Barrier Performance

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

15.4.1.2.5 Radiological Consequences

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

15.4.2 Rod Withdrawal Error at Power

15.4.2.1 Identification of Causes and Frequency Classification

15.4.2.1.1 Identification of Causes

The RWE transient results from a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously until the rod withdrawal limiter (RWL) function of the rod control and information system (RCIS) blocks further withdrawal.

15.4.2.1.2 Frequency Classification

The frequency of occurrence for the RWE is assumed to be moderate, since definite data do not exist. The frequency of occurrence diminishes as the reactor approaches full power by virtue of the reduced number of control rod movements. A statistical approach using appropriate conservative acceptance criteria, shows that consequences of the majority of RWEs would be very mild and hardly noticeable.

15.4.2.2 Sequence of Events and Systems Operation

15.4.2.2.1 Sequence of Events

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

15.4.2.2.2 System Operations

While operating in the power range in a normal mode of operation, the reactor operator makes a procedural error and withdraws the maximum worth control rod or gang of control rods continuously until the rod withdrawal limiter inhibits further withdrawal. The rod withdrawal limiter utilizes rod position indications of the selected rod as input. The basis for the precalculated rod block is to ensure that safety limits will not be exceeded before the rod withdrawal is blocked assuming that the fuel adjacent to the selected rod is operating at the highest power consistent with fuel operating limits.

The rod withdrawal limiter device protects against the single operator error. Protection against multiple errors withdrawing the same rod is procedural. The System Operating Instruction for RCIS requires the operator to determine the cause of a rod block before further rod withdrawal can take place.

There are restrictions on rod motion over the whole operating range. At power less than the low power set point, the rod pattern control system function of the rod control and information system (RCIS) mitigates the consequences of a rod withdrawal error.

The rod pattern control system performs redundant functions which provide protection in the event of a rod drop accident or a rod withdrawal error. At power greater than the low power set point, the rod withdrawal limiter function of the RCIS mitigates the consequences of the rod withdrawal error.

The 100 percent rod withdrawal error analyses report represents the most limiting conditions. The analysis at 70 percent power assumes the same limiting condition, but the power and flow are reduced along the rated load line.

The Technical Specifications will specify the rod withdrawal limiter allowable increments as a function of reactor power over the whole operating domain (greater than the low power set point).

Any rod withdrawal when no rod movement was demanded will trigger an audible rod drift alarm. Rod drift is a rare, abnormal event as determined from experience with operating BWRs. Depending on the severity of the malfunction which caused the rod to drift, transition boiling could be reached for worst case rod drift events; this does not necessarily indicate that there will actually be any fuel damage. Upon hearing the rod drift alarm, the operator will take action to mitigate most rod drift events.

During the course of this event, normal operation of plant instrumentation and controls is assumed, although no credit is taken for this except as described above. No operation of any engineered safety feature is required during this event.

15.4.2.2.3 Single Failure or Single Operator Error

The effect of operator errors has been discussed above. It was shown that operator errors (which initiated this transient) cannot impact the consequences of this event due to the RCIS system. The RCIS system is designed to be single-failure proof; therefore, termination of this transient is assured. See Appendix 15A for details.

15.4.2.3 Core and System Performance

15.4.2.3.1 Mathematical Model

The consequences of a RWE are calculated utilizing the three-dimensional coupled nuclear-thermal-hydraulics computer program (Reference 23). This model calculates the changes in power level, power distribution, core flow, and critical power ratio under steady-state conditions, as a function of control blade position. For this transient, the rate of power increase is slow compared to the fuel thermal time constant and core hydraulic transport times, so that the steady-state assumption is adequate.

Reference 8 documents GE's generic RWE analysis applicable to Grand Gulf Unit 1 operation within the maximum extended operating domain (MEOD). As indicated in subsection 15.4.2.3.4, the possible number of combinations of rod control patterns and reactor states is very large making it impractical to identify a single worst case set of initial conditions for a RWE transient. The approach taken by the reload vendor is the same as for the initial cycle: a large number of transient simulations were performed for a wide range of operating state points and rod patterns for BWR/6 reactors; and a statistical analysis was performed on the simulations results. The principal results of interest for each simulation are initial MCPR and ΔCPR. The objective of the RWE generic transient analysis was to determine statistically bounding (95% probability/95% confidence) values for changes in the core limiting MCPR from minimum CPR calculated before and after a hypothesized RWE transient event.

The generic analysis for the basic BWR/6 power and flow operating domain is described in Reference 8. Additional simulations were performed for operating statepoints in the extended regions of the power and flow operating domain. The additional data from these analyses were combined with the results from Reference 16 to expand the data base. The statistical analysis of the expanded database was used to determine operating limits for the MEOD power and flow map. The MEOD analysis is described in Reference 16.

For the current cycle, the reload vendor performed a cycle specific RWE analysis. The limiting result between the generic analysis and cycle specific analysis is reported in the current cycle Supplemental Reload Licensing Report (Ref. 28).

15.4.2.3.2 Input Parameters and Initial Conditions

The reactor core is assumed to be operating at full or part power conditions prior to RWE initiation. A statistical analysis of the rod withdrawal error results (Refs. 8, 16, 18, and 23) initiated from a wide range of operating conditions (exposure, power, flow, rod patterns, etc.) has been performed, confirming allowable rod withdrawal increments applicable to all BWR/6 plants using GE 8x8, GE11, GE14, GNF2, and GNF3, or SPC 8x8-2, 9x9-5, and Atrium-10 fuel. These rod withdrawal increments were determined such that the design basis ΔMCPR (minimum critical power ratio) for rod withdrawal errors initiated from various operating conditions and mitigated by the rod withdrawal limiter system withdrawal restrictions, provides at least a 95 percent probability at the 95 percent confidence level that any randomly occurring RWE will not result in a larger ΔMCPR. MCPR was verified to be the limiting thermal performance parameter. The 1 percent strain limit on the clad was always a less limiting parameter.

15.4.2.3.3 Results

The calculated results of the generic BWR/6 analyses demonstrate that, should a rod or gang be withdrawn a distance equal to the allowable rod withdrawal increment, there exists at least a 95 percent probability at the 95 percent confidence level that the resultant ΔMCPR will not be greater than the design basis ΔMCPR. Furthermore, the peak LHGR will be substantially less than that calculated to yield 1 percent strain in the fuel clad.

Table 15.4-17 shows the results of the generic RWE analysis for a typical cycle. These results are the ΔMCPRs 95/95 taken from Reference 13. The current cycle RWE MCPR results are based on the reload vendor's generic analysis (Reference 23) and the cycle specific safety limit MCPR.

These results of the generic analyses in References 8 and 16 show that a control rod or gang can be withdrawn in increments of 12 inches at power levels ranging from 70 to 100 percent of rated, and 24 inches at power levels ranging from 20 to 70 percent (Table 15.4-2). See subsection 15.4.1.2 for RWEs below 20 percent reactor power. The 20 percent and 70 percent reactor core power levels correspond to the low power set point and high power set point of the rod withdrawal limiter.

For each current cycle, the reload fuel vendor performs a cycle specific analysis. If the cycle specific results are not bounded by the generic BWR/6 analysis for the one-foot withdrawal increment, additional analyses are performed to confirm that the generic results for the 70% power two-foot withdrawal increment are bounding. The RWE OLMCPR result provided in the SRLR (Ref. 28) is the more limiting of the cycle specific and the generic OLMCPRs.

15.4.2.3.4 Consideration of Uncertainties

The most significant uncertainty for this transient is the initial control rod pattern and the location of the rods or gang improperly selected and withdrawn. Because of the near-infinite combinations of control patterns and reactor states, all possible states cannot be analyzed. However, high worth control rods were included in the statistical analysis, and enough points have been evaluated so as to clearly establish the 95%/95% confidence level. This effectively bounds the results from any actual operator error of this type with the indicated probabilities.

Quasi-steady-state conditions were assumed for thermal-hydraulic conditions. Although the uncertainty introduced by this assumption is not conservative, the magnitude of the effects neglected is insignificant relative to the result of the transient.

15.4.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since this is a localized event with very little change in the gross core characteristics. Typically, an increase in total core power for RWEs initiated from rated conditions is less than 4 percent and the changes in pressure are negligible.

15.4.2.5 Radiological Consequences

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

15.4.2.6 Initial Cycle

[HISTORICAL INFROMATION] [The approach taken in the initial cycle is the same as for the current cycle, i.e., a generic study using statistical analysis of a large number of individual RWE events. The consequences of a RWE for the initial cycle are calculated using the computer program described in Reference 1. Inputs and initial conditions are similar to the generic study. The reactor was assumed to be on the existing MCPR and MLHGR limits prior to the RWE. The RWEs were initiated at a wide range of operating conditions (exposure, power, flow, rod patterns, xenon conditions, etc.). The generic study for the initial cycle (Reference 8) was used to establish allowable rod withdrawal increments. These increments were determined such that the design basis ΔMCPR for rod withdrawal errors initiated from the operating limit and mitigated by the rod withdrawal limiter system withdrawal restrictions, provided a 95% probability at the 95% confidence level that any randomly occurring RWE would not result in a larger MCPR. These analyses were used to establish the allowable withdrawal increments.]

15.4.3 Control Rod Maloperation (System Malfunction or Operator Error)

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

15.4.4 Abnormal Startup of Idle Recirculation Pump

The reload fuel vendor has determined that the abnormal startup of idle recirculation pump event is not limiting event for the current reload cycle. However, this event was re-analyzed due to a plant modification to change the recirculation flow control valve minimum position. Therefore, this event description was updated from the initial cycle analysis to reflect that new analysis which is the current licensing basis for GGNS. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0. This event is confirmed to be bounded by generic off rated limits for introduction of new GNF fuel. Reference 10 documents this evaluation.

15.4.4.1 Identification of Causes and Frequency Classification

15.4.4.1.1 Identification of Causes

This action results directly from the operator's manual action to initiate pump operation. It assumes that the remaining loop is already operating.

15.4.4.1.1.1 Normal Restart of Recirculation Pump at Power

This transient is categorized as an incident of moderate frequency.

15.4.4.1.1.2 Abnormal Startup of Idle Recirculation Pump

This transient is categorized as an incident of moderate frequency.

15.4.4.2 Sequence of Events and Systems Operation

15.4.4.2.1 Sequence of Events

Table 15.4-3 lists the sequence of events for Figure 15.4-3.

15.4.4.2.1.1 Deleted

15.4.4.2.2 Systems Operation

This event assumes and takes credit for normal functioning of plant instrumentation and controls. No protection systems action is anticipated inasmuch as the intent for starting the pump in the first place is to do so without initiating a scram. No engineered safety feature action occurs as a result of the transient.

15.4.4.2.3 The Effect of Single Failures and Operator Errors

Attempts by the operator to start the pump at higher power levels will result in a reactor scram on flux. This situation involves an operator error in that the idle loop is started when the drive flow in the active loop is above 50% of rated drive flow.

This action would violate technical specification instructions. The analysis as performed is the maximum allowable power level assuming the operator has violated the drive flow requirement for second loop startup. See Appendix 15A for details.

[HISTORICAL INFORMATION] [The original GGNS analysis of this event considered an additional operator error associated with the loop differential temperature. As described in GE SIL 517, Supplement 1, the analyses supporting MEOD eliminated this second operator error.]

15.4.4.3 Core and System Performance

15.4.4.3.1 Mathematical Model

The predicted dynamic behavior has been determined using a computer simulated, analytical model. The computer model used in GNF analysis of this event is described in detail in Reference 23.

15.4.4.3.2 Input Parameters and Initial Conditions

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

One recirculation loop is idle and filled with cold water. (Normal procedure when starting an idle loop with one pump already running requires that the indicated idle loop temperature be no more than 50˚F lower than the indicated active loop temperature.)

The active recirculation loop is producing 54.1% of rated core flow, the maximum possible in single loop operation. The core power is 60% of rated power. Higher power levels were found to result in a scram and consequently, a milder transient.

The idle recirculation pump suction and discharge block valves are open and the recirculation flow control valve is closed to its minimum open position. (For single pump on low frequency motor generator set, normal procedure requires leaving the flow control valve in the operating loop in the maximum position to maintain the loop temperature within the required limits for restart.)

15.4.4.3.3 Results

The transient response to the incorrect startup of a cold, idle recirculation loop is shown in Figure 15.4-3. Shortly after the pump begins to move, a surge in flow from the started jet pump diffusers causes the core inlet flow to rise. The motor approaches synchronous speed in approximately 3 seconds because of the assumed minimum pump and motor inertia.

A short-duration neutron flux peak well below the scram setpoint is produced as the colder, increasing core flow reduces the void volume. Surface heat flux follows the slower response of the fuel and peaks at slightly above 80 percent of rated before decreasing after the cold water washes out of the loop at about 25 seconds. No damage occurs to the fuel barrier and MCPR remains above the safety limit as the reactor settles out at its new steady state condition.

For the introduction of new fuel, this event was analyzed at various power and flow points in Reference 10. The results are bounded by the off-rated power dependent limits. No damage occurs to the fuel barrier and MCPR remains above the safety limit.

15.4.4.3.4 Consideration of Uncertainties

This particular transient is analyzed for an initial power level that is much higher than that expected for the actual event. The much slower thermal response of the fuel mitigates the effects of the rather sharp neutron flux spike and even in this high range of power, no threat to thermal limits is possible.

15.4.4.4 Barrier Performance

No evaluation of barrier performance is required for this event since no significant pressure increases are incurred during this transient. See Figure 15.4-3.

15.4.4.5 Radiological Consequences

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

15.4.5 Recirculation Flow Control Failure with Increasing Flow

There are two scenarios evaluated for the recirculation flow control failure with increasing flow event; one with fast opening and one with slow opening of the recirculation flow control valves. The NSSS vendor's analyses showed that the fast opening of one or both recirculation valves is a relatively mild transient. The slow opening of the recirculation valves, however, is analyzed for introduction of new fuel to establish cycle independent flow dependent limits. The analysis is documented in References 11 and 29. This subsection describes the analyses performed by the NSSS vendor for fast opening of the recirculation valves for the initial cycle which remains the current analysis for this event. It also describes the analyses performed by the reload fuel vendor of the slow opening of the recirculation control valve for introduction of new fuel. Additional discussion on the relationship between analyses performed by the NSSS vendor and the reload fuel vendor are provided in Section 15.0.

The results of the slow flow excursion transient analyses are used to determine two flow dependent thermal limits: MCPRf and LHGRFACf. The transient scenario assumes a failure of the recirculation flow control system such that the reactor recirculation flow increases slowly to the physical maximum attainable by the recirculation system. The mode of operation analyzed for the current cycle of Grand Gulf Unit 1 is "loop manual" only. Since only one recirculation valve would open, this mode of operation corresponds to a single recirculation loop flow excursion event.

15.4.5.1 Identification of Causes and Frequency Classification

15.4.5.1.1 Identification of Causes

Failure of the master controller or neutron flux controller can cause an increase in the core coolant flow rate. Failure within a loop's flow controller can also cause an increase in core coolant flow rate.

The slow opening of one recirculation valve is credible during the loop manual operating mode when the recirculation valves are under independent control. The core flow increase is assumed to be slow enough such that the event can be analyzed using steady state analysis methods.

15.4.5.1.2 Frequency Classification

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

15.4.5.2 Sequence of Events and Systems Operation

15.4.5.2.1 Sequence of Events

15.4.5.2.1.1 Fast Opening of One Recirculation Valve

Table 15.4-4 lists the sequence of events for Figure 15.4-4.

15.4.5.2.1.2 Fast Opening of Two Recirculation Valves

Table 15.4-5 lists the sequence of events for Figure 15.4-5.

15.4.5.2.1.3 Slow Opening of One Recirculation Valve.

No credit is taken for the reactor protection system trip setpoints during this event. Therefore, a sequence of events table is not meaningful.

15.4.5.2.1.4 Slow Opening of Two Recirculation Flow Control Valves

Since operation in modes other than loop manual is not allowed for GGNS, slow opening of two recirculation valves is not credible.

15.4.5.2.2 Systems Operation

The analysis for fast opening of the recirculation flow control valve assumes and takes credit for normal functioning of plant instrumentation and controls, and the reactor protection system. Operation of engineered safeguards is not expected.

The slow opening of the recirculation valve results in the plant reaching a new steady state at higher values of core flow and power except in cases where pressurization occurs due to assuming bypass valves are inoperable.

15.4.5.2.3 The Effect of Single Failures and Operator Errors

The fast opening of one or two recirculation valves leads to a quick rise in reactor power level. Corrective action first occurs in the high flux trip which, being part of the reactor protection system, is designed to single failure criteria. (See Appendix 15A for details.) Therefore, shutdown is assured. Operator errors are not of concern here in view of the fact that automatic shutdown events follow so quickly after the postulated failure.

Single failures and single operator errors associated with corrective action are not of concern for the slow opening of a recirculation flow control valve event because the valve is assumed to have failed open to its maximum position, and no credit is taken in the analysis for reactor protection system operation.

15.4.5.3 Core and System Performance

15.4.5.3.1 Mathematical Model

The nonlinear NSSS vendor's dynamic model described briefly in Section 15.0 is used to simulate the fast opening of the recirculation valves events.

The results of the slow flow excursion transient analyses by the reload vendor were used to establish new flow dependent thermal limits of MCPRf. For current analysis (Reference 11), the change in critical power along the flow ascension path is calculated with ISCOR (Reference 23).

The LHGRFACf analysis by the reload vendor is performed with the PANACEA (Ref. 25) neutronic code assuming a single pump runup slow flow excursion.

15.4.5.3.2 Input Parameters and Initial Conditions

15.4.5.3.2.1 Fast Opening of Recirculation Flow Control Valves

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

In each of these transient events the most severe transient results when initial conditions are established for operation at the low end of the rated flow control rod line. Specifically, this is 54 percent of the initially licensed NB rated power and 34 percent core flow. The maximum stroking rate of the recirculation loop valves for a master controller failure driving two loops is limited by individual loop controls to 11 percent per second.

Maximum stroking rate of a single recirculation loop valve for a loop controller failure is limited by hydraulics to 30 percent per second.

The initial operating MCPR for the transient described above is 1.545.

These analyses assume no maximum flow control set point. Instead, it is assumed that the valve or valves are opened at the maximum stroking rate to the fully open position.

The overall method used to calculate CPR is described in Chapter 6 of Reference 9.

15.4.5.3.2.2 Slow Opening of a Recirculation Flow Control Valve

For the $MCPR_p$ analysis, peaking factors were selected such that the bundle with the least margin would reach the safety limit MCPR at the maximum achievable flow. The $MCPR_f$ limit for maximum achievable core flow is based on the assumption that the recirculation system equipment is capable of 105% of rated flow on the limiting rod line. The initial conditions for the slow recirculation flow increase event were established for a set of core flows along the limiting rod line, starting from a minimum of 20% of rated flow. An initial flow mismatch between the two recirculation loops is assumed consistent with the technical specifications limits. The loop with the lower initial flow is assumed to run up with the recirculation flow control valve opening to its full open position. The position of the other recirculation flow control valve is assumed to be unchanged. For flow rates less than 30% rated flow, the recirculation system operates at low speed which restricts the maximum flow possible.

The $LHGRFAC_f$ analysis was performed in a manner similar to the $MCPR_f$ analysis. A series of flow excursion analyses were performed starting from different initial power/flow conditions. Variations in the cycle exposure and control rod patterns were also considered. The final conditions were determined based on the maximum attainable core flow rate. Xenon was conservatively assumed to remain constant during the event. The $LHGRFAC_f$ operating limits were established to bound the limiting results. The $LHGRFAC_f$ multipliers were established to ensure that the LHGR during the flow run-up does not violate the LHGR overpower limit for the current GNF2 fuel (Ref. 28). Because of restrictions in flow rates attainable for operation with core flows less than 30% of rated, the $LHGRFAC_f$ conservatively remains constant for core flow rates between 20% and 30%.

15.4.5.3.3 Results

15.4.5.3.3.1 Fast Opening of One Recirculation Valve

Figure 15.4-4 shows the analysis of a failure where one recirculation loop main valve is opened at its maximum stroking rate of 30 percent per second.

The rapid increase in core flow causes a sharp rise in neutron flux initiating a reactor scram at approximately 1.1 seconds. The peak neutron flux reached was 316 percent of NB rated value, while the accompanying average fuel surface heat flux reaches 73 percent of NB rated at approximately 2 seconds. MCPR remains above safety limit. Reactor pressure is discussed in subsection 15.4.5.4.

15.4.5.3.3.2 Fast Opening of Two Recirculation Valves

Figure 15.4-5 illustrates the failure where both recirculation loop main valves are opened at a maximum stroking rate of 11 percent per second. It is very similar to the above transient. Flux scram occurs at approximately 1.3 seconds, peaking at 256 percent of NB rated while the average surface heat flux reaches 72 percent of NB rated at approximately 2.0 seconds. MCPR remains above the safety limit of 1.06 and fuel center temperature increases 145˚F.

As indicated above, this is the most severe set of conditions under which this transient may occur. The results expected from an actual occurrence of this transient will be less severe than those calculated.

15.4.5.3.3.3 Slow Opening of a Recirculation Flow Control Valve

The change in CPR during the slow opening of one recirculation flow control valve from different initial core flows is used in determining the flow-dependent MCPR limit ($MCPR_f$). The $MCPR_f$ limit ensures that the acceptance criterion of maintaining the MCPR above the safety limit during the flow increase is satisfied for all core final flows.

The $LHGRFAC_f$ limits ensure that the design LHGR for each fuel type is not exceeded during the flow increase for all initial flows.

15.4.5.3.4 Considerations of Uncertainties

Some uncertainties in void reactivity characteristics, scram time and worth are expected to be more optimistic and will therefore lead to reducing the actual severity over that which is simulated herein.

15.4.5.4 Barrier Performance

15.4.5.4.1 Fast Opening of One Recirculation Valve

This transient results in a very slight increase in reactor vessel pressure as shown in Figure 15.4-4 and therefore represents no threat to the RCPB.

15.4.5.4.2 Fast Opening of Two Recirculation Valves

This transient results in a very slight increase in reactor vessel pressure as shown in Figure 15.4-5 and therefore represents no threat to the RCPB.

15.4.5.4.3 Slow Opening of a Recirculation Flow Control Valve

This event results in a final pressure corresponding to the final steady state power. Because of the quasi steady state nature of the event, there is no threat to the reactor coolant pressure boundary.

15.4.5.5 Radiological Consequences

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

15.4.6 Chemical and Volume Control System Malfunctions

Not applicable to BWRs. This is a PWR event.

15.4.7 Misplaced Bundle Accident

The reload fuel vendor has determined that the misplaced bundle event is a limiting event which requires analysis for the current reload cycle. However, the misplaced bundle accident is analyzed generically in GESTAR11 (Reference 23). The applicability of the generic analysis is confirmed for each fuel cycle. For a description of the initial fuel cycle analysis of this event by the NSSS vendor, refer to subsection 15.4.7.6. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.4.7.1 Identification of Causes and Frequency Classification

15.4.7.1.1 Identification of Causes

The event discussed in this section is the improper loading of a fuel bundle and subsequent operation of the core. Two types of loading errors are possible, the mislocation of an assembly and the misorientation of the assembly. Three errors must occur for the mislocation event to take place. First, a bundle must be misloaded into a wrong position in the core. Second, the bundle which was supposed to be loaded where the mislocation occurred would have to be overlooked and also put in an incorrect location. Third, the misplaced bundles would have to be overlooked during the core verification performed following core loading. For the misorientation event, two things must take place. First, the assembly must be rotated while being lowered into position. Second, the misoriented bundle would have to be overlooked during the core verification performed following the core loading.

15.4.7.1.2 Frequency of Occurrence

This event occurs either when a fuel bundle is loaded into the wrong location in the core or when the orientation with respect to the control blade corner is misaligned while being loaded. It is assumed the bundle is misplaced to the worst possible location, and the plant is operated with the misplaced bundle. This event is categorized as an infrequent incident based on the data described in GESTAR II (Ref. 23). The fuel loading error rate experience documented in GESTAR II shows ≈0.17 errors per Plant per Lifetime compared to the RG 1.70 infrequent incident threshold of 1 error per Plant per Lifetime.

15.4.7.2 Sequence of Events and Systems Operation

The postulated sequence of events for the misplaced bundle and misoriented bundle fuel loading errors are presented in Table 15.4-6.

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

15.4.7.2.1 Effects of Single Failure and Operator Errors

This analysis already represents the worst case (i.e., operation of a misplaced bundle with three SACF or SOE) and there are no further operator errors which can make the event results any worse. It is felt that this section is not applicable to this event. Refer to Appendix 15A for further details.

15.4.7.3 Core and System Performance

15.4.7.3.1 Mathematical Model

As referenced in GESTAR II (Ref. 23), industry standard or approved models are used to calculate the offsite and control room dose consequences of the assumed failure of fuel rods from this event.

15.4.7.3.2 Input Parameters and Initial Conditions

The typical Grand Gulf core configuration and additional bundle and core design details are provided in Section 4.3. The Grand Gulf core design is a conventional scatter load with the lowest reactivity bundles placed in the peripheral region of the core. The loading pattern is designed to maximize cycle energy and minimize power peaking factors. This type of design shows a strong relationship between local reactivity and MCPR.

The adverse consequences from an incident of a fuel loading error (either a mislocated fuel bundle or a misoriented fuel bundle) could be the failure of one or more fuel rods in a single fuel bundle that is operating in a higher-than-normal power range. The results of such an incident would be similar to a fuel bundle operating with one or more leaking fuel rods. However, the radiological consequences, even though minor, would be difficult to assess for each fuel bundle in the core for each operating cycle. The GESTAR II (Ref. 23) basis provides a clearly bounding generic analysis for this event, in that it is assumed that all of the fuel rods in five fuel bundles experience instantaneous failure during normal operation. Grand Gulf confirmed that the core verification and radiological parameters are within the limits of the generic analysis as described in GESTAR II such that this generic Fuel Loading Error analysis is applicable.

The generic analysis considers a fuel loading error residing in a cell. Instead of one or two rods failing, it is assumed that all the fuel rods in a mislocated fuel assembly or a misoriented fuel assembly and all the rods in the four adjacent fuel assemblies experience instantaneous failure during normal operation. To further assure that the fuel bundles containing the maximum fission products for release are included, all five bundles (array independent) are multiplied by a factor to account for variations in fission product inventory over the operational cycle and variations in cycle-dependent bundle power as a ratio to the end of cycle average bundle power.

15.4.7.3.3 Results

The GESTAR II generic analysis applicable to the current cycle conservatively determined that worst case scenarios of the misoriented and mislocated bundle events would not result in exceeding a small fraction (< 10%) of the 10CFR50.67 limit or the General Design Criteria 19 control room dose limits.

15.4.7.3.4 Considerations of Uncertainties

The consideration of uncertainties is not relevant to the conservative fuel failure assumptions considered in the generic GESTAR II analysis (Ref. 23), which is applicable to the current cycle.

15.4.7.4 Barrier Performance

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

15.4.7.5 Radiological Consequences

The relevant radiological criteria for the misplaced fuel bundle event is that the dose consequences do not exceed a small fraction (less than 10%) of 10CFR50.67. For the current cycle, the generic GESTAR II analysis (Ref. 23) was shown to be applicable. The resulting offsite doses are within this criterion.

15.4.7.6 Initial Cycle

The reload fuel vendor has determined that the misplaced bundle accident is a limiting event which requires analysis for the current fuel cycle. This subsection describes the analysis performed by the NSSS vendor for the initial fuel cycle. For a description of the current fuel cycle analysis of this event, refer to subsection 15.4.7. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.4.7.6.1 Identification of Causes and Frequency Classification (Initial Cycle)

The potential causes of this event are the same as for the current cycle. The probability of occurrence of this event is also the same as that for the current cycle. Refer to subsection 15.4.7.1.

15.4.7.6.2 Sequence of Events and Systems Operation (Initial Cycle)

The sequence of events and systems operation are the same for the initial cycle as those described in subsection 15.4.7.2 for the current cycle.

15.4.7.6.3 Core and System Performance (Initial Cycle)

15.4.7.6.3.1 Mathematical Model

[HISTORICAL INFORMATION] [A three-dimensional BWR simulator model is used to calculate the core performance resulting from this event. This model is described in detail in Reference 1.]

15.4.7.6.3.2 Input Parameters and Initial Conditions

[HISTORICAL INFORMATION] [The initial core consisted of three bundle types with average enrichments that were high, medium, or low, with correspondingly different gadolinia concentrations. The fuel bundle loading error involves interchanging a bundle of one enrichment with another bundle of a different enrichment. The following fuel loading errors can be conceived for an initial core:

1. A high-enriched bundle is misloaded into a low-enriched bundle location.

2. A medium-enriched bundle is misloaded into a low-enriched bundle location.

3. A low-enriched bundle is misloaded into a high-enriched bundle location.

4. A low-enriched bundle is misloaded into a medium-enriched bundle location.

5. A medium-enriched bundle is misloaded into a high-enriched bundle location.

6. A high-enriched bundle is misloaded into a medium-enriched bundle location.

Since all low-enriched bundles are located on the core periphery, the two possible fuel loading errors consisting of the misloading of high or medium-enriched bundles into a low- enriched bundle location, i.e., types 1 and 2, are not significant. In these cases, the higher reactivity bundles are moved to a region of lower importance, resulting in an overall improvement in performance.

The third type of fuel loading error, as identified above, results in the largest enrichment mismatch. However, it does not result in an unacceptable operating consequence. Consider a fuel bundle loading error at the beginning-of-cycle (BOC) with the low-enriched bundle (which should be loaded at the periphery) interchanged with a high-enriched bundle located adjacent to a Local Power Range Monitor (LPRM) and predicted to have the highest LHGR and/or lowest CPR in the core. After the loading error has occurred and has gone undetected, it is assumed, for purposes of conservatism, that the operator uses a control pattern that places the limiting bundle in the four bundle array containing the misplaced bundle, on thermal limits as recorded by the LPRM. As a result of loading the low-enriched bundle in an improper location, the average power in the four bundles decreases. Normally, the reading of the LPRM will show a decrease in thermal flux due to the decreased power. However, in this case an increase in the thermal flux occurs due to decreased neutron absorption in the low-enriched bundle. The effects of the softer neutron spectrum due to the decreased thermal absorption are larger than the power depression effect of the lower fission rate resulting in a net increase in instrument reading. Thus, a fuel loading error of this kind does not result in undetected reductions in thermal margins during power operations.

The fourth and fifth type of fuel loading errors are of the same kind (lower enrichment into higher enrichment) as the third type, and also do not result in a nonconservative operating error.

The fuel bundle loading error with greatest impact on thermal margin is of the sixth type, which occurs when a high-enriched bundle is interchanged with a medium-enriched bundle located away from an LPRM. Since the medium and high enrichment bundles have a corresponding medium and high gadolinia content, the maximum reactivity difference occurs at end of cycle (EOC), where the gadolinia is burned out. After the loading errors are made and have gone undetected, the operator assumes that the mislocated bundle is operating at the same power as the instrumented bundle in the mirror image location and operates the plant until EOC. For the purpose of conservatism, it is assumed that the mirror image bundle is on thermal limits, as recorded by the LPRM. As a result of placing the instrumented bundle on limits, the mislocated bundle violates the initial cycle operating MCPR limit.

A summary of input parameters for this analysis is given in Table 15.4.7a.]

15.4.7.6.3.3 Results

[HISTORICAL INFORMATION] [A bounding analysis was performed to quantify the worst fuel bundle loading error for initial core. A summary of the results of that analysis is presented in Table 15.4-8a. As can be seen, MCPR remains well above the MCPR safety limit, and MLHGR does not exceed the 1 percent plastic strain limit for the clad. Therefore, no violation of fuel limits occurs as a result of this event.]

15.4.7.6.3.4 Considerations of Uncertainties

[HISTORICAL INFORMATION] [In order to assure the conservatism of this analysis, major input parameters are taken as a worst case, i.e., the bundle is placed in location with the highest LHGR and/or the lowest CPR in the core and the bundle is operating on design thermal limits. This assures that the ΔCPR and the ΔLHGR are the upper bounds for the error.]

15.4.7.6.4 Barrier Performance (Initial Cycle)

[HISTORICAL INFORMATION] [An evaluation of the barrier performance was not made for this event since it is a very mild and highly localized event. No perceptible change in the core pressure would be observed.]

15.4.7.6.5 Radiological Consequences (Initial Cycle)

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

15.4.8 Deleted

15.4.9 Control Rod Drop Accident (CRDA)

The reload fuel vendor has determined that the control rod drop accident is a limiting event which requires analysis for the current fuel cycle. The NRC approved control rod drop accident analysis for banked position withdrawal sequence plants (such as GGNS) described in Reference 23 can be applied to any fuel cycle. Results of the control rod drop accident analysis for the current fuel cycle are presented in the supplemental reload licensing report. For a description of the initial fuel cycle analysis of this event by the NSSS vendor, refer to subsection 15.4.9.6. For additional information on the relationship between analysis performed by the NSSS vendor for initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.4.9.1 Identification of Causes and Frequency Classification

15.4.9.1.1 Identification of Causes

The control rod drop accident is the result of a postulated event in which a high worth control rod that is inserted into the core becomes decoupled from its drive mechanism. The mechanism is then withdrawn, but the decoupled control rod is assumed to be stuck in place. At a later optimum moment, the control rod suddenly falls free and drops out of core or to the drive position. This results in the removal of large negative reactivity from the core and results in a localized power excursion.

A more detailed discussion is given in Reference 2.

15.4.9.1.2 Frequency of Classification

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

15.4.9.2 Sequence of Events and System Operation

15.4.9.2.1 Sequence of Events

Before the control rod drop accident (CRDA) is possible, the sequence of events presented in Table 15.4-9 must occur. No operator actions are required to terminate this transient.

15.4.9.2.2 Systems Operation

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

The rod pattern control system (RPCS) of the rod control and information system (RCIS) limits the worth of any control rod which could be dropped by regulating the withdrawal sequence. This system prevents the movement of an out-of-sequence rod in the 100 to 75 percent rod density range, and from the 75 percent rod density point to the preset power level the RPCS will only allow bank position mode rod withdrawals or insertions. The banked position mode of this system is described in Reference 3.

The RPCS uses redundant input to provide absolute assurance of control rod drive position. If either of the diverse input were to fail the other would provide the necessary information.

The termination of this excursion is accomplished by automatic safety features of inherent shutdown mechanisms. Therefore, no operator action during the excursion is required. Other normal plant instrumentation and controls are assumed to function. These functions include an automatic isolation of the mechanical vacuum pumps. Even though the mechanical vacuum pumps are non-safety related components, credit for pump trip and isolation is implicit in the assumptions in Appendix C to Reg. Guide 1.183 (Ref. 7) and has been specifically reviewed and approved by the NRC in the Safety Evaluation Report for Reference 22.

15.4.9.2.3 Effect of Single Failures and Operator Errors

The consequences of this event are mitigated by the RPCS and APRM scram. The RPCS is designed as a redundant system and therefore provides single failure protection. The APRM scram system is designed to a single failure criteria. Therefore, termination of this transient within the limiting results discussed below is assured.

No operator error (in addition to the one that initiates this event) can result in a more limiting case since the reactor protection system will automatically terminate the transient.

Appendix 15A provides a detailed discussion on this subject.

15.4.9.3 Core and System Performance

15.4.9.3.1 Mathematical Model

The reload fuel vendor's analytical methods, assumptions, and conditions for evaluating the excursion aspects of the control rod drop accident are described in detail in Reference 23. This is considered to provide a realistic yet conservative assessment of the associated consequences. The data presented in Reference 3 shows that the RPCS banked position mode reduces the control rod worths to the degree that the detailed analyses presented in References 2, 4, and 5 are not necessary. References 2, 3, 4, and 5 provide sensitivity studies which demonstrate large margin to the allowable peak fuel enthalpy for rod worths below 1% Δk.

The reload fuel vendors' methodology relies on the baseline analyses which shows there is no possible rod worth which (if dropped at the design rate of the velocity limiter) could result in a peak enthalpy of 280 cal/g at reactor powers greater than 10% rated. Furthermore, the baseline analyses show that the most limiting condition to experience a CRDA occurs in the hot standby state.

The reload fuel vendors' CRDA methodology (Reference 23) conservatively assumes an adiabatic boundary condition at the pellet-gap interface and no direct moderator heating. This prevents heat transfer from the fuel rod to the coolant, thus the deposited enthalpy is equivalent to the energy produced in the fuel. The peak power is converted to a maximum deposited enthalpy using a ratio of the local powers of the bundles surrounding the dropped control rod.

Control rod drop accident (CRDA) results from banked position withdrawal sequencing (BPWS) plants have been statistically analyzed and documented in a generic analysis documented in Reference 23. The results show that, in all cases, the peak fuel enthalpy in a CRDA would be less than the 280 cal/gm design limit even with a maximum incremental rod worth corresponding to 95% probability at the 95% confidence level.

15.4.9.3.2 Input Parameters and Initial Conditions

The core at the time of rod drop accident is assumed to contain no xenon, to be in a hot-startup condition, and to have the control rods in a sequence consistent with the RPCS. For conservatism, eight rods are assumed to be inoperable and remain in the fully inserted position. The location of the inoperable rods are chosen to maximize the worth of the dropped rod. For the current cycle, the licensing configuration with inoperable rods as well as the nominal configuration without inoperable rods are both analyzed. Removing xenon, which competes well for neutron absorptions, increases the fractional absorptions, or worth, of the control rods.

Since the maximum incremental rod worth is maintained at very low values (by the RPCS), the postulated CRDA cannot result in peak enthalpies in excess of 280 calories per gram for any plant condition. The data presented in subsection 15.4.9.3.3 show the maximum control rod worth. Other input parameters and initial conditions are shown in Table 15.4-10.

15.4.9.3.3 Results

The radiological evaluations are based on the assumed failure of 16 fuel bundles. The number of rods which exceed the damage threshold is less than this assumed damage for all plant operating conditions or core exposures provided the peak enthalpy is less than the 280 cal/gm design limit.

The results of reload fuel vendor studies indicate that the maximum incremental rod worth is well below the worth required to cause a CRDA which would result in 280 cal/gm peak fuel enthalpy (Reference 23). The conclusion is that the 280 cal/gm design limit is not exceeded and the assumed failure of 16 fuel bundles for the radiological evaluation is conservative. For the current cycle, there is ample margin to the peak fuel enthalpy demonstrated by the compliance checks. Similarly, for GNF2 and GNF3 fuel the conclusion that the 280 cal/gm design limit is not exceeded has been confirmed and the assumed failure of 1200 fuel rods for GNF2 and GNF3 fuel is bounding.

15.4.9.4 Barrier Performance

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

15.4.9.5 Radiological Consequences

The design basis analysis is based on the alternative source term in Regulatory Guide 1.183 (Reference 7). The dose calculation methodology used is the same as that used for the design basis LOCA analysis of Section 15.6.5. Specific parametric values used in the CRDA evaluation are presented in Table 15.4-12.

15.4.9.5.1 Fission Product Release from Fuel

The failure of 16 fuel bundles is used for this analysis. The mass fraction of the fuel in the damaged rods, which reaches or exceeds the initiation temperature of fuel melting (taken as 2804°C) is estimated to be 0.0077.

Fuel reaching melt conditions is assumed to release 100 percent of the noble gas inventory, 30 percent of the iodine inventory, and 25 percent of the alkali metal inventory. The remaining fuel in the damaged rods is assumed to release 10 percent of both the noble gas and iodine inventories, and 12 percent of the alkali metal inventory.

A maximum equilibrium inventory of fission products in the damaged bundles is based on continuous operation at a bounding bundle power level. No delay time is considered between departure from the above power condition and the initiation of the accident.

15.4.9.5.2 Fission Product Transport to the Environment

The transport pathway is shown in Figure 15.4-7 and consists of carryover with steam to the turbine condenser and leakage from the condenser to the environment. No credit is taken for MSIV closure or for the turbine building.

Of the activity released from the fuel, 100 percent of the noble gases, 10 percent of the iodines, and 1 percent of the alkali metals are assumed to be carried to the condenser. The 10 percent iodine fraction is considered to be conservative relative to the maximum possible amount of iodine which could realistically be transported to the condenser following a CRDA event (Reference 22).

Of the activity reaching the condenser, 100 percent of the noble gases, 10 percent of the iodines (due to partitioning and plateout), and 1 percent of the alkali metals remain airborne. The activity airborne in the condenser is assumed to leak directly to the environment at a rate of 1.0 percent per day for 24 hours, at which time the leakage is assumed to terminate. Radioactive decay is accounted for during residence in the condenser, however it is neglected after release to the environment.

The initial activity airborne in the condenser is presented in Table 15.4-13.

15.4.9.5.3 Results

The calculated exposures from the design basis analysis are presented in Table 15.4-15 and are well within the guidelines of Reg. Guide 1.183 and 10 CFR 50.67.

15.4.9.6 Initial Cycle

The reload fuel vendor has determined that the control rod drop accident is a limiting event which requires analysis for the current fuel cycle. This subsection describes the analysis performed by the NSSS vendor for the initial fuel cycle. For a description of the current fuel cycle analysis of this event, refer to subsection 15.4.9. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.4.9.6.1 Identification of Causes and Frequency Classification (Initial Cycle)

The potential causes of this event are the same as for the current cycle. The probability of occurrence of this event is also the same as that for the current cycle. Refer to subsection 15.4.9.1.

15.4.9.6.2 Sequence of Events and Systems Operation (Initial Cycle)

The sequence of events and systems operation are the same for the initial cycle as those described in subsection 15.4.9.2 for the current cycle.

15.4.9.6.3 Core and System Performance (Initial Cycle)

15.4.9.6.3.1 Mathematical Model

[HISTORICAL INFORMATION] [For the initial cycle, the analytical methods, assumptions, and conditions for evaluating the excursion aspects of the control rod drop accident are described in References 2, 4 and 5. The data presented in Reference 3 show that the RPCS banked position mode reduces the control rod worths to the degree that the detailed analyses presented in References 2, 4 and 5 or the bounding analyses presented in Reference 6 are not necessary. Instead, compliance checks were made to verify that the maximum rod worth did not exceed 1 percent Δk.

If this criterion were not met, then the bounding analyses were performed. The rod worths were determined using the BWR simulator model described in Reference 1. Detailed evaluations, when necessary, were made using the methods described in References 2, 4 and 5.]

15.4.9.6.3.2 Input Parameters and Initial Conditions

[HISTORICAL INFORMATION] [For the initial cycle analysis, the core at the time of rod drop is assumed to be at the point in the cycle which results in the highest incremental rod worth, to contain no xenon, to be in a hot-startup condition, and to have the control rods in Sequence A at 50 percent rod density (groups 1-4 withdrawn). The 50% control rod density (black and white rod pattern), which nominally occurs at the hot-startup condition, ensures that withdrawal of a rod results in the maximum increment of reactivity.

Because the maximum incremental rod worth was maintained very low, the postulated CRDA could not result in peak enthalpies greater than 280 calories per gram as in the current cycle. Other input parameters and initial conditions are shown in Table 15.4-10.]

15.4.9.6.3.3 Results

[HISTORICAL INFORMATION] [Radiological evaluations were based on the assumed failure of 770 fuel rods for the initial cycle.

The results of the compliance check calculation, as shown in Table 15.4-11 indicated that the maximum incremental rod worth was well below the worth required to cause a CRDA which would result in 280 calories per gram peak fuel enthalpy. Assuming 770 fuel pins fail for the radiological evaluation was conservative.]

15.4.9.6.4 Barrier Performance (Initial Cycle)

[HISTORICAL INFORMATION] [As in the current cycle, an evaluation of the barrier performance was not made since this is a highly localized event with no significant change in the gross core temperature or pressure.]

15.4.9.6.5 Radiological Consequences (Initial Cycle)

[HISTORICAL INFORMATION] [The radiological consequences of this event for the initial cycle are lower than for the current cycle because the number of fuel pins assumed to fail is higher in the current cycle and these rods are assumed to operate at a greater power peaking than in the initial cycle analysis. Also, MSIV closure is not assumed for the current cycle.]

15.4.10 References

1. Woolley, J. A., "Three Dimensional Boiling Water Reactor Core Simulator," NEDO-20953, May 1976.

2. Stirn, R. C. et al., "Rod Drop Accident Analysis for Large BWRs," March 1972 (NEDO-10527).

3. Paone, C. J., "Bank Position Withdrawal Sequence," January 1977 (NEDO-21231).

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

5. Stirn, R. C., et al., "Rod Drop Accident Analysis for Large BWRs," January 1973, Supplement 2 (NEDO-10527).

6. "GE BWR Generic Reload Application for 8x8 Fuel," (NEDO-20360).

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12. Deleted

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14. Deleted

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17. Deleted

18. XN-NF-825(P) (A), "BWR/6 Generic Rod Withdrawal Error Analysis, MCPRρ," Exxon Nuclear Company, Richland, WA, April 1985.

19. Deleted

20. Deleted

21. Deleted

22. Licensing Topical Report NEDO-31400A, Rev. 0, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor," BWR Owner's Group, October, 1992.

23. NEDE-24011-P-A (Including US Supplement) "General Electric Standard Application for Reactor Fuel (GESTAR II)," Latest Approved Version.

24. Deleted

25. NEDE-30130-P-A, "Steady-State Nuclear Methods," GE Nuclear Energy, April 1985.

26. Deleted

27. Deleted

28. ECH-NE-20-00009, Rev. 0, “Supplemental Reload Licensing Report for Grand Gulf-Reload 22 Cycle 23,” February 2020.

29. ECH-NE-20-00006, Rev. 0, “GNF3 Fuel Design Cycle-Independent Analyses for Grand Gulf Nuclear Station”, Mar. 2020.

**TABLE 15.4-1: SEQUENCE OF EVENTS
RWE IN POWER RANGE**

Elapsed Time (sec)	Event
0	Core is operated in a typical control rod pattern on limits
0	Operator withdraws a single rod or gang of rods continuously
~1	The local power in the vicinity of the withdrawn rod (or gang) increases. Gross core power increases.
~4*	RWL blocks further withdrawal
~25	Core stabilizes at slightly higher core power level.

*For a 1.0-foot RWL incremental withdrawal block. Time would be longer for a larger block, since rods are withdrawn at approximately 3 inches per second.

TABLE 15.4-2: Deleted

TABLE 15.4-3: SEQUENCE OF EVENTS FOR STARTUP OF IDLE RECIRCULATION PUMP (FIGURE 15.4-3)

Time (sec)	Event
0	Start Pump Motor
1.5	Jet pump diffuser flow on idle loop becomes positive.
2.8	Peak neutron flux
3	Pump motor of idle loop at full speed
9.5	Minimum value of core inlet enthalpy
9.5	Peak heat flux
25+	Reactor variables settle into new steady state.

TABLE 15.4-4: SEQUENCE OF EVENTS FOR FAST OPENING OF ONE RECIRCULATION LOOP VALVE (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT) (FIGURE 15.4-4)

Time-sec	Event
0	Simulate failure of single loop control.
1.1	Reactor APRM high flux scram trip initiated.
4.4 (est)	Turbine control valves start to close upon falling turbine pressure.
30.6 (est)	Turbine control valves closed. Turbine pressure below pressure regulator set points.
31.8 (est)	Vessel water level (L8) trip initiates main turbine and feedwater turbine trips.
31.9 (est)	Main turbine stop valves closed. Bypass does not open as turbine inlet pressure remains below pressure regulator set points.
>50.0 (est)	Reactor variables settle into new steady state.

TABLE 15.4-5: SEQUENCE OF EVENTS FOR FAST OPENING OF BOTH RECIRCULATION LOOP VALVES (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT) (FIGURE 15.4-5)

Time-sec	**Event**
0	Initiate failure of master controller.
1.3	Reactor APRM high flux scram trip initiated.
4.3 (est)	Turbine control valves start to close upon falling turbine pressure.
9.0 (est)	Turbine control valves closed. Turbine pressure below pressure regulator setpoints.
10.0	Vessel water level reaches Level 2.
10.2	Recirculation pumps tripped due to Level 2 trip.
27.0	Vessel water level (L8) trip initiates main turbine and feedwater turbine trips.
27.1	Main turbine stop valves closed. Bypass does not open as turbine inlet pressure remains below pressure regulator set points.
27.2 (est)	Turbine bypass valves to regulate pressure.
>50 (est)	Reactor variables settle into new steady state.

TABLE 15.4-6: SEQUENCE OF EVENTS FOR MISLOCATED BUNDLE ACCIDENT

(1) During core loading operation, bundle is placed in the wrong position.

(2) Subsequently, the bundle intended for this position is placed in the position of the previous bundle.

(3) During core verification procedure, error is not observed.

(4) Plant is brought to full power operation without detecting misplaced bundle.

(5) Plant continues to operate throughout the cycle.

SEQUENCE OF EVENTS FOR MISORIENTED BUNDLE ACCIDENT

(1) During core loading operation, bundle is misoriented about its vertical axis 180° from its prescribed orientation relative to the control blade.

(2) During core verification procedure, error is not observed.

(3) Plant is brought to full power operation without detecting misoriented bundle.

(4) Plant continues to operate throughout the cycle.

TABLE 15.4-7: Deleted

TABLE 15.4-7A: INPUT PARAMETERS AND INITIAL CONDITIONS FOR FUEL BUNDLE LOADING ERROR (INITIAL CYCLE) [HISTORICAL INFORMATION]

(1)	Power, % Rated	100
(2)	Flow, % Rated	100
(3)	MCPR Operating Limit	1.18
(4)	MLHGR Operating Limit, kw/ft	13.4
(5)	Average core exposure	End of Cycle

NOTE: Core conditions are assumed to be normal for a hot, operating core at EOC.

TABLE 15.4-8: Deleted

TABLE 15.4-8A: RESULTS OF MISPLACED BUNDLE ANALYSIS (INITIAL CYCLE) [HISTORICAL INFORMATION]

(1)	MCPR limit	1.18
(2)	MCPR with misplaced bundle	1.08
(3)	ΔCPR for event	0.10
(4)	LHGR limit	13.4
(5)	LHGR with misplaced bundle	14.7
(6)	ΔLHGR for event	1.3

TABLE 15.4-9: SEQUENCE OF EVENTS FOR ROD DROP ACCIDENT

Approximate Elapsed Time	Event
	Reactor is executing standard startup procedure withdrawing control rods in accordance with RPCS limitations.
	Maximum worth control rod blade becomes decoupled from the CRD.
	Operator selects and withdraws the control rod drive of the decoupled rod along with the other control rods assigned to the RCIS group.
	Decoupled control rod sticks in the fully inserted or an intermediate bank position.
0	Control rod becomes unstuck and drops to the drive position at the nominal measured velocity plus three standard deviations.
<1 second	Reactor goes on a positive period and initial power increase is terminated by the Doppler coefficient.
<1 second	APRM 120% power signal scrams reactor.
<5 seconds	Scram terminates accident.

TABLE 15.4-10: INPUT PARAMETERS AND INITIAL CONDITIONS FOR ROD WORTH COMPLIANCE CALCULATION

Initial Analysis

1.	Reactor Power, % Rated	0.
2.	Reactor Flow, % Rated	0.0
3.	Core Average Exposure, MWd/t	0.0
4.	Control Rod Fraction	~.50
5.	Average Fuel Temperature, °C	286
6.	Average Moderator Temperature, °C	286
8.	Xenon State	None

Typical Reload Cycle Analysis

1.	Reactor Power, % Rated	0.0
2.	Reactor Flow, % Rated	0.0
3.	Core Average Exposure, GWd/MTU	13.5
4.	Control Rod Fraction	~.90
5.	Average Fuel Temperature, °C	288
6.	Average Moderator Temperature, °C	288
7.	Xenon State	None

TABLE 15.4-11: INCREMENT WORTH OF THE MOST REACTIVE ROD USING BPWS INITIAL CYCLE ANALYSIS

[HISTORICAL INFORMATION]

Core Condition	Control Rod Group*	Banked At Notch	Control Rod (I,J)	Drops From-To	Increase In K_{eff}
BOC-L:	7	4	(28,37)	0 8	0.0023
Sequence A;	7	8	(28,37)	0 12	0.0035
Rod Groups 1-4 Withdrawn;	7	12	(28,37)	0 48	0.0040
Rod Groups 5,6,8,9,10 Fully Inserted	7	48	(28,37)	0 48	0.0029

NOTES:

1. The following assumptions were made to ensure that the rod worths were conservatively high for the BPWS:
   a) BOC
   b) Hot Startup
   c) No Xenon

2. In the generic analysis (NEDO-21231), the most reactive rod in the withdrawal sequence was a group 9 rod withdrawal after the withdrawal of groups 5 and 6 for the equilibrium cycle case (Table 4-3). However, for cycle 1, the generic analyses show that the most reactive rod in the withdrawal sequence is, in fact, a group 7 rod withdrawal. Similarly, the Grand Gulf plant specific calculations show group 7 to have the highest worth in cycle 1. However, the location of the highest incremental worth control rod is variable and dependent on the radial power shape and control rod pattern (refer to last paragraph on page 4-9 of NEDO-21231 for more information). The change in location between cycle 1 and the equilibrium cycle can be attributed to the change in radial power shape.

3. Comparison of the rod worth values for Grand Gulf with those similarly located rods in Table 4-3 of NEDO-21231 shows that the Grand Gulf values are lower than the generic values. This difference is due to the size of the plant. A comparison between different size plants, but of similar core design, beginning of cycle one with rod group 1-4 of sequence at A withdrawn is shown in Table 15.4-16.

*For Definition of Rod Groups, See NEDO-21231

TABLE 15.4-12: CONTROL ROD DROP ACCIDENT EVALUATION PARAMETERS

			Design Basis Assumptions
I.	Data and assumptions used to estimate radioactive source from postulated accidents		
	A.	Bundle Power	10.6 MWt
	B.	Fuel damaged	16 bundles
	C.	Iodine fractions	
		(1) Organic	0.03
		(2) Elemental	0.97
		(3) Particulate	0
II.	Data and assumptions used to estimate activity released.		
	A.	Condenser leak rate (%/day)[1]	1.0
	B.	Isotope Release Fractions	Reg. Guide 1.183 (Ref. 7)
III.	Dispersion Data		
	A.	Exclusion Area χ/Q (696m)	
		0-2 hrs	6.50E -04 s/m^3
	B.	LPZ χ/Q (3218m)	
		0-2 hrs	1.45E - 04
		2 hrs - 8 hrs	7.10E - 05
		8 hrs -24 hrs	5.00E - 05
	C.	Control Room χ/Q	
		0-2 hrs	7.00E - 04
		2-8 hrs	6.00E - 04
		8 hrs - 24 hrs	2.55E - 04
	D.	Control Room Parameters	Table 15.6-13
IV.	Dose Data		
	A.	Initial activity concentrations in condenser.	Table 15.4-13

TABLE 15.4-12: CONTROL ROD DROP ACCIDENT EVALUATION PARAMETERS (Continued)

		Design Basis Assumptions
B.	Dose Conversion Factors	Federal Guidance Reports 11 and 12
C.	Doses	Table 15.4-15

(1) Assumes mechanical vacuum pumps are automatically isolated.

TABLE 15.4-13: CONTROL ROD DROP ACCIDENT INITIAL ACTIVITY AIRBORNE IN CONDENSER

Isotope	Curies	Isotope	Curies
BR-82	5.76E+01	XE-129M	2.20E+01
BR-83	7.85E+02	XE-131M	5.83E+03
BR-84	1.47E+03	XE-133	9.64E+05
KR-83M	8.14E+04	XE-133M	3.00E+04
KR-85	5.61E+03	XE-135	2.91E+05
KR-85M	1.90E+05	XE-135M	2.08E+05
KR-87	3.83E+05	XE-137	9.04E+05
KR-88	5.42E+05	XE-138	9.39E+05
KR-89	6.88E+05	RB-86	2.62E-01
I-128	1.83E+02	RB-88	6.18E+01
I-130	4.28E+02	RB-89	8.07E+01
I-131	4.95E+03	CS-132	2.82E-02
I-132	7.07E+03	CS-134	2.16E+01
I-133	9.89E+03	CS-134M	6.01E+00
I-134	1.11E+04	CS-136	6.01E+00
I-135	9.21E+03	CS-137	7.22E+00
		CS-138	1.15E+02
		CS-135M	6.90E+00

TABLE 15.4-14: Deleted

TABLE 15.4-15: CONTROL ROD DROP ACCIDENT RADIOLOGICAL EFFECTS

	TOTAL EFFECTIVE DOSE EQUIVALENT (REM)
Exclusion Area (2 hrs.)	0.198
Low Population Zone (24 hrs.)	0.092
Control Room (72 hrs.)	<0.29

Table 15.4-16: COMPARISON OF ROD WORTH VALUES
(Initial Cycle)
[HISTORICAL INFORMATION]

REFERENCE	CORE SIZE	CONTROL ROD GROUP	BANKED AT NOTCH	CONTROL ROD (I-J)	DROPS FROM-TO	ΔK INCREASE
Table 4-4 NEDO-21231	368	7	12	18-27	00-48	0.0082
BWR/5	444	7	12	9-9	00-48	0.0074
BWR/5	560	7	12	22-31	00-48	0.0052
BWR/5	764	7	12	26-35	00-48	0.00424
Grand Gulf TABLE 15.4-11	800	7	12	9-9	00-48	0.004

This data clearly shows that the larger plant has the lower rod worths. In this case, loading pattern difference is not a factor because the region-wise K_∞ is essentially uniform

TABLE 15.4-17: RESULTS OF CONTROL ROD WITHDRAWAL ERROR ANALYSIS (GENERIC BWR/6 ANALYSIS FOR A TYPICAL CYCLE)

Initial Power Level	ΔCPR
100%[1]	0.10
70%[1]	0.18
70%[2]	0.34
20%[2]	0.48

[1] One foot ganged rod withdrawal

[2] Two foot ganged rod withdrawal.

TABLE 15.4-18: Deleted

Figure 15.4-1

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Figure 15.4-2

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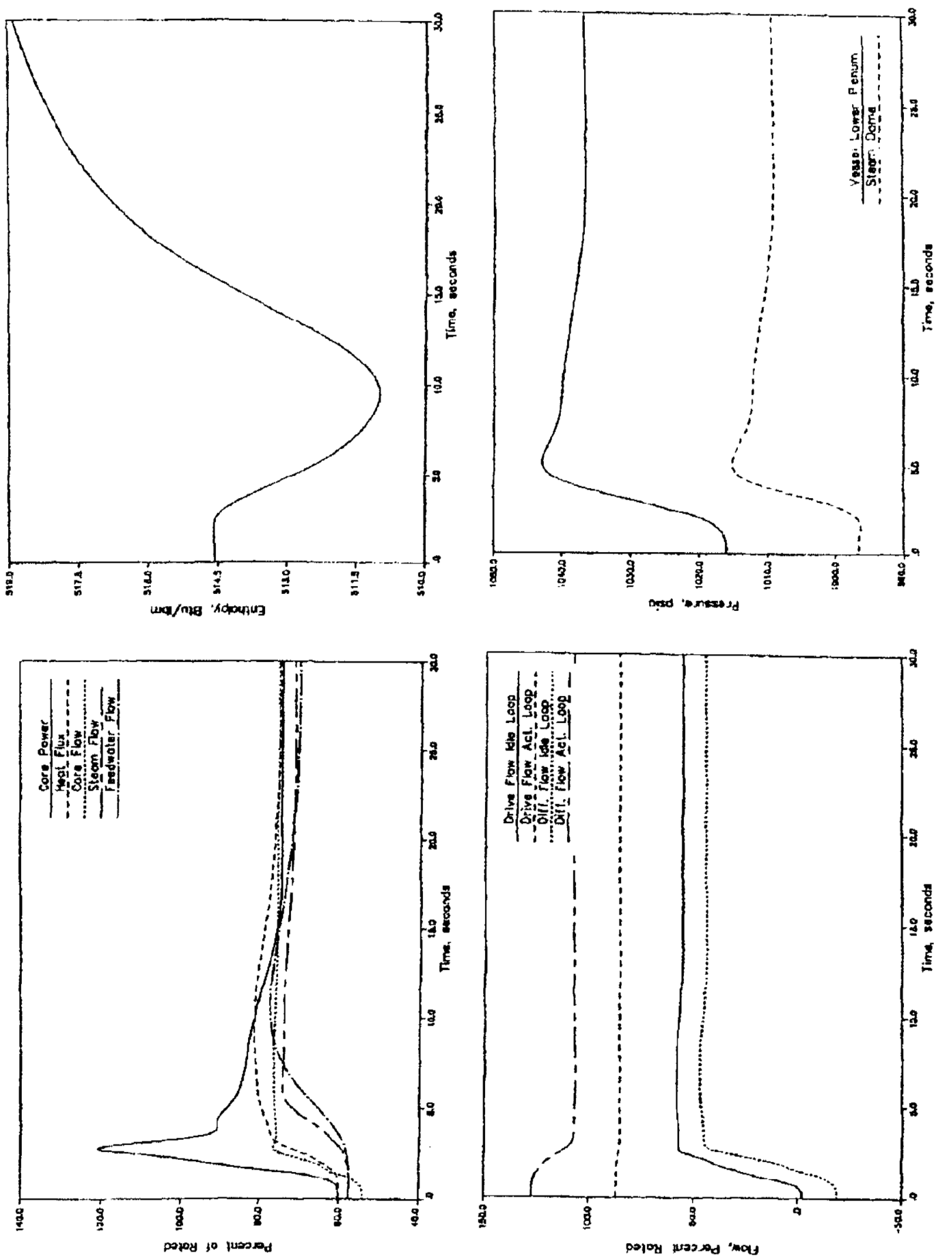

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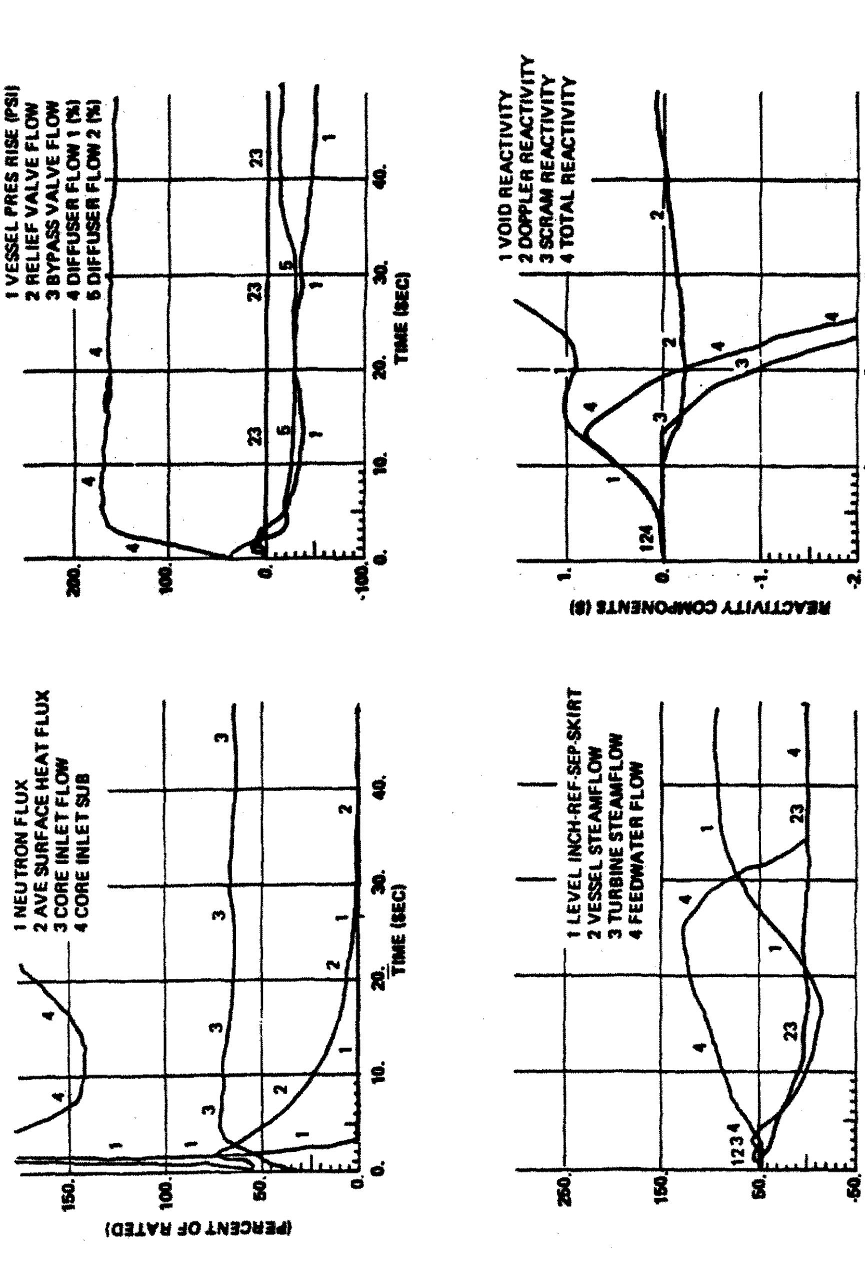
STARTUP OF IDLE RECIRCULATION LOOP PUMP
(ATRIUM-10 Reload Cycle)
FIGURE 15.4-3



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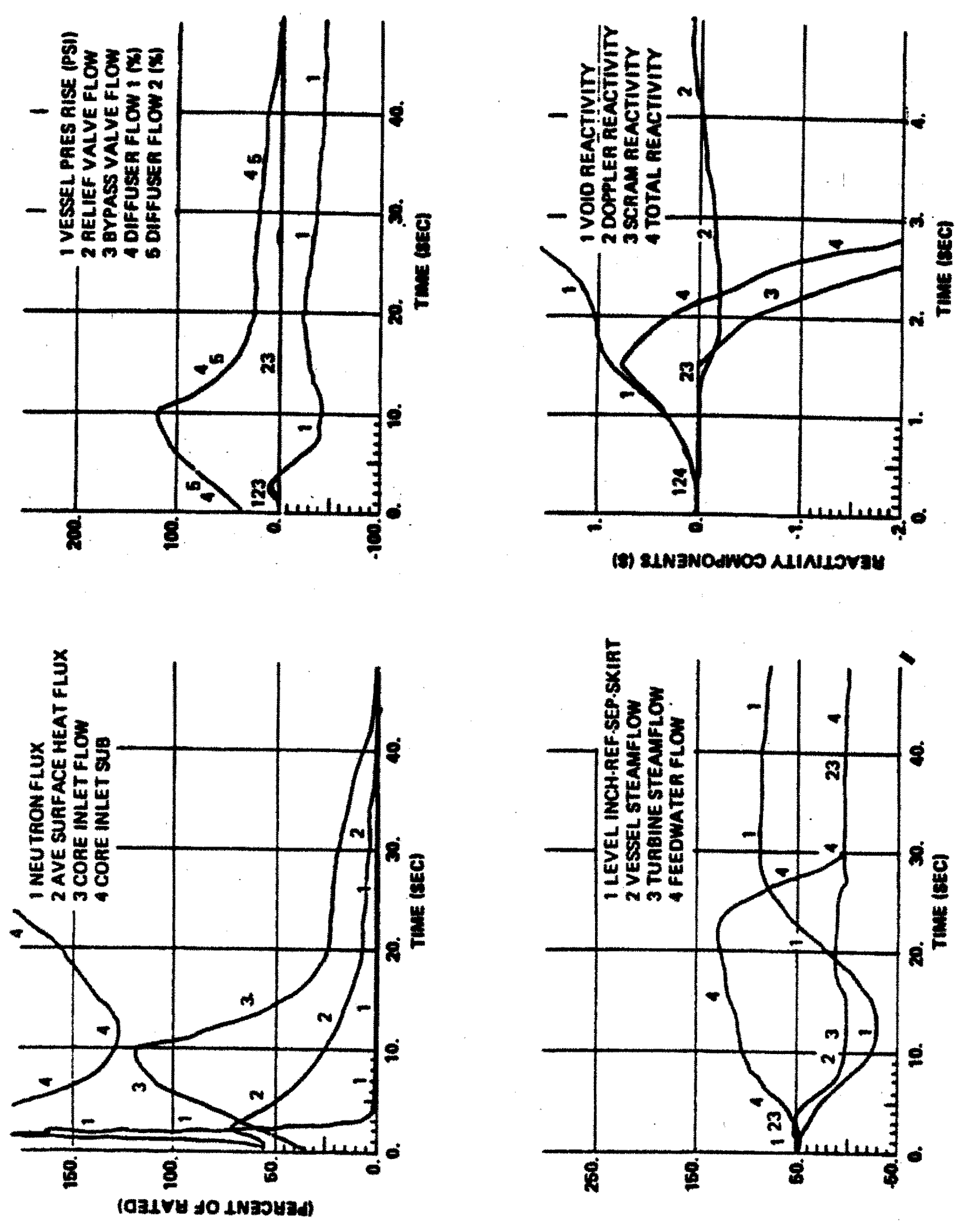
FAST OPENING OF ONE MAIN RECIRCULATION LOOP VALVE AT 30% PER SECOND
(INITIAL CYCLE)
FIGURE 15.4-4

Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW, but remains the current analysis of record for this event.



Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW, but remains the current analysis of record for this event.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FAST OPENING OF BOTH MAIN RECIRCULATION LOOP VALVES AT 11% PER SECOND (INITIAL CYCLE) FIGURE 15.4-5

Figure 15.4-6

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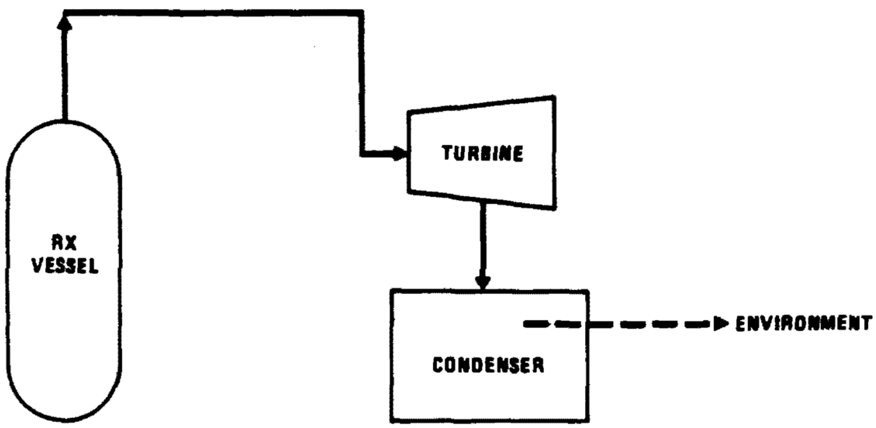


GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	LEAKAGE PATH MODEL FOR CONTROL ROD DROP ACCIDENT FIGURE 15.4-7

15.5 INCREASE IN REACTOR COOLANT INVENTORY

15.5.1 Inadvertent HPCS Startup

The reload fuel vendor has determined that the inadvertent HPCS startup event is not a limiting event for the current reload cycle. Therefore, this subsection describes the original analysis performed by the NSSS vendor for the initial cycle which remains the current licensing basis for GGNS. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

15.5.1.1 Identification of Causes and Frequency Classification

15.5.1.1.1 Identification of Causes

Inadvertent startup of the HPCS system is postulated for this analysis, e.g., operator error.

15.5.1.1.2 Frequency Classification

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

15.5.1.2 Sequence of Events and Systems Operation

15.5.1.2.1 Sequence of Events

Two alternative sequences are credible: In the first sequence, the reactor vessel level control system is unable to compensate for the level increase resulting from HPCS injection. When the level rises to the Level 8 trip setpoint, the HPCS injection path is closed; the main turbine is tripped and a scram is initiated. In the second sequence, the addition of HPCS water does not result in a level increase significant enough to cause a Level 8 trip. A new, stable operating state is established at a slightly different power and vessel pressure and lower steam and feedwater flow rates.

The effects of level increase to the Level 8 and 9 trip setpoints for HPCS injection (i.e., the first sequence) are bounded by similar events that result in increases in reactor vessel inventory (e.g., Feedwater Controller Failure - Maximum Demand, Section 15.1.2). Therefore, the analyses results provided in this section address only the second sequence.

Table 15.5-1 lists the sequence of events for the second sequence. The transient response for this sequence is shown in Figure 15.5-1.

15.5.1.2.1.1 Deleted

15.5.1.2.2 System Operation

In order to simulate properly the expected sequence of events the analysis of this event assumes normal functioning of plant instrumentation and controls. Specifically, the pressure regulator and the vessel level control which respond directly to this event.

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

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

15.5.1.2.3 The Effect of Single Failures and Operator Errors

For conditions when the reactor vessel level control system compensates for the effects of HPCS injection, inadvertent operation of the HPCS results in a mild depressurization. Corrective action by the pressure regulator and/or level control is expected to establish a new stable operating state.

The effect of a single failure in the pressure regulator will aggravate the transient depending upon the nature of the failure. Pressure regulator failures are discussed in subsections 15.1.3 and 15.2.1.

The effect of a single failure in the level control system has rather straightforward consequences including level rise or fall by improper control of the feedwater system. Increasing level will trip the turbine and automatically trip the HPCS system off. This trip signature is already described in the failure of feedwater controller with increasing flow. Decreasing level will automatically initiate scram at the L3 level trip and will have a signature similar to loss of feedwater control-decreasing flow.

The single failures relevant to the “inadvertent HPCS start-up” transient are either the pressure regulator failure or level control failures. Neither failure is expected because both systems are in normal continuous operation at the time of the hypothesized event, and no significant change in their function is demanded by the event. They should simply continue their normal function. Inadvertent start-up of the HPCS results in a mild depressurization. Upon depressurization due to addition of cooler water to the upper plenum, the pressure regulator tends to regulate the vessel pressure by closing the turbine control valves. When an active failure of the regulator system is considered (such that the turbine control valves would be kept wider open), further depressurization would be caused which would lead the event along a path similar to the pressure regulator failure - open transient (subsection 15.1.3). No significant change in thermal margin protection would occur. Since the water level rises when this transient begins, the level control system tends to reduce the feedwater flow and mitigate the level increase. When an active failure of the level control system is considered, the water level continues to increase. This situation is similar to the "feedwater controller failure with increasing flow" transient (subsection 15.1.2) and is bounded by that event because of lower vessel pressure for the level controller failure which would ease the reduction of thermal margin. Therefore, an acceptable consequence is expected.

15.5.1.3 Core and System Performance

15.5.1.3.1 Mathematical Model

The detailed nonlinear dynamic model described briefly in subsection 15.1.1.6.3.1 is used to simulate this transient.

15.5.1.3.2 Input Parameter and Initial Conditions

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

The lowest injection water temperature of the HPCS system was assumed to be 40°F with an enthalpy of 11 Btu/lb. The transient as analyzed is very mild. If water at 32 F was injected, an additional 1 to 2 percent of the core average voids would collapse. The maximum neutron flux would increase to approximately 105 percent of initially licensed NBR. No significant change (<1 percent) in the core average surface heat flux would occur, and CPR would remain unchanged.

Inadvertent start-up of the HPCS system was chosen to be analyzed since it provides the greatest auxiliary source of cold water into the vessel.

15.5.1.3.3 Results

Figure 15.5-1 shows the simulated transient event for the manual flow control mode. It begins with the introduction of cold water into the upper core plenum. Within 3 seconds the full HPCS flow is established at approximately 8.7 percent of the rated feedwater flow rate. This flow is nearly 174 percent the HPCS flow at rated pressure. No delays were considered because they are not relevant to the analysis.

Addition of cooler water to the upper plenum causes a reduction in steam flow which results in some depressurization as the pressure regulator responds to the event. In the automatic flow control mode, following a momentary decrease, neutron power settles out at a level slightly above operating level. In manual mode the flux level settles out slightly below operating level. In either case, pressure and thermal variations are relatively small and no significant consequences are experienced. MCPR is not changed significantly, therefore fuel thermal margins are maintained.

15.5.1.3.3.1 Consideration of Uncertainties

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

15.5.1.4 Barrier Performance

Figure 15.5-1 indicates a slight pressure reduction from initial conditions, therefore, no further evaluation is required as RCPB pressure margins are maintained.

15.5.1.5 Radiological Consequences

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

15.5.2 Chemical Volume Control System Malfunction (or Operator Error)

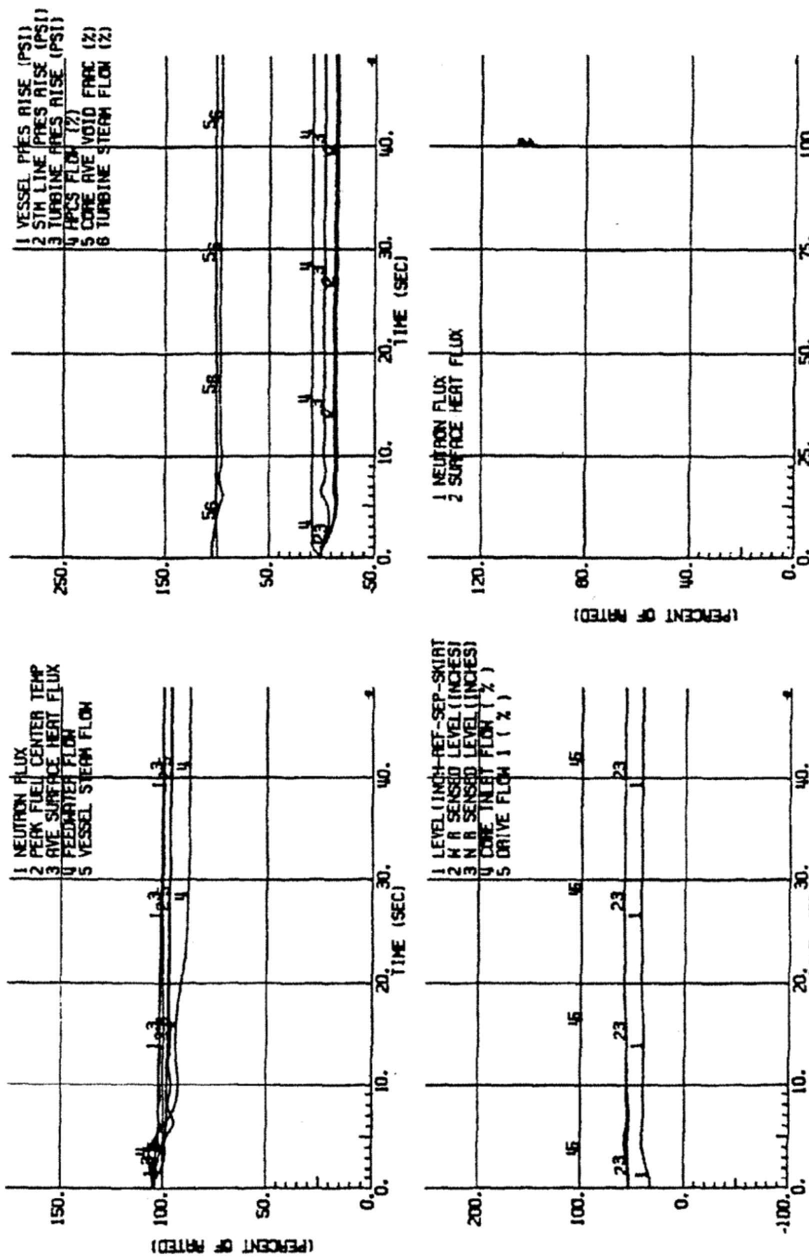
This section is not applicable to BWR. This is of PWR interest.

15.5.3 BWR Transients Which Increase Reactor Coolant Inventory

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

TABLE 15.5-1 SEQUENCE OF EVENTS FOR INADVERTENT STARTUP OF HPCS (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT) (FIGURE 15.5-1)

Time-sec	Event
0	Simulate HPCS cold water injection.
3	Full flow established for HPCS.
5	Depressurization effect stabilized.
20	Reactor variables settle into new steady state.



Note: Initial cycle analysis is based on the originally licensed power level of 3833 MW, but remains the current analysis of record for this event.

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INADVERTENT STARTUP OF HPCS
(INITIAL CYCLE)

FIGURE 15.5-1

15.6 DECREASE IN REACTOR COOLANT INVENTORY

15.6.1 Inadvertent Safety/Relief Valve Opening

This event is discussed and analyzed in subsection 15.1.4.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

Standard Review Plan 15.6.2 covers the radiological consequences of failures outside the containment of small lines connected to the reactor coolant pressure boundary, such as instrument lines and sample lines.

The Grand Gulf design has no instrument or sample lines connected to the reactor coolant pressure boundary which penetrate the primary containment. Therefore, SRP 15.6.2 is not applicable.

15.6.3 Steam Generator Tube Failure

This subsection is not applicable to the direct cycle BWR. This is a PWR related event.

15.6.4 Steam System Piping Break Outside Containment

The reload fuel vendor has determined that the steam system piping break outside containment event is not a limiting event for the current reload cycle. However, the main steam line break event was re-analyzed during the implementation of EPU and, therefore, this subsection describes the current analysis performed by the GEH as part of EPU implementation. The radiological consequences represent the calculation of record following analyses associated with the alternative source term and EPU. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

This event involves the postulation of 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 the broken line, and actuate the necessary protective features. This postulated event represents the envelope evaluation of steam line failures outside containment.

15.6.4.1 Identification of Causes and Frequency Classification

15.6.4.1.1 Identification of Causes

A main steam line break is postulated without the cause being identified. These lines are designed to high quality engineering codes and standards, and to restrictive seismic and environmental requirements. However, for the purpose of evaluating the consequences of a postulated large steam line rupture, the failure of a main steam line is assumed to occur.

15.6.4.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.4.2 Sequence of Events and Systems Operation

15.6.4.2.1 Sequence of Events

Accidents that result in the release of radioactive materials directly outside the containment are the results of postulated breaches in the reactor coolant pressure boundary or the steam power conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the four main steam lines. The sequence of events and approximate time required to reach the event is given in Table 15.6-1.

15.6.4.2.1.1 Deleted

15.6.4.2.2 Systems Operation

A postulated guillotine break of one of the four main steam lines outside the containment results in mass loss from both ends of the break. The flow from the upstream side is initially limited by the flow restrictor upstream of the inboard isolation valve. Flow from the downstream side is initially limited by the total area of the flow restrictors in the three unbroken lines. 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. Refer to Figure 15.6-1.

A discussion of plant and reactor protection system action and ESF action is given in Sections 6.2, 6.3, 7.2, and 7.3.

15.6.4.2.3 The Effect of Single Failures and Operator Errors

The effect of single failures has been considered in analyzing this event. The ECCS aspects are covered in Section 6.3. The break detection and isolation considerations are defined in Sections 7.3 and 7.6. All of the protective sequences for this event are capable of SACF and SOE accommodation and yet completion of the necessary safety action. Refer to Appendix 15A for further details.

15.6.4.3 Core and System Performance

Quantitative results (including math models, input parameters, and consideration of uncertainties) for this event are given in Section 6.2. The temperature and pressure transients resulting as a consequence of this accident are insufficient to cause fuel damage.

15.6.4.3.1 Input Parameters and Initial Conditions

Refer to Section 6.2 for initial conditions.

15.6.4.3.2 Results

There is no fuel damage as a consequence of this accident. Refer to Section 6.2 for ECCS analysis.

15.6.4.3.3 Considerations of Uncertainties

Sections 6.2 and 7.3 contain discussions of the uncertainties associated with the ECCS performance and the containment isolation systems, respectively.

15.6.4.4 Barrier Performance

Since this break occurs outside the containment, barrier performance within the containment envelope is not applicable.

15.6.4.5 Radiological Consequences

The design basis analysis is based on NRC Regulatory Guide 1.183. Specific values of parameters used in the evaluation are presented in Table 15.6-2.

15.6.4.5.1 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 steam lines prior to the break. The level of activity is consistent with an offgas release rate of 100 Ci/sec - MWT after 30 minutes delay (399,000 μCi/sec) for noble gases. Consistent with Regulatory Guide 1.183, two cases are evaluated: (1) an equilibrium iodine case with an iodine concentration in the reactor coolant of 0.2 μCi/gm dose equivalent I-131, and (2) an iodine spiking case with an iodine concentration in the reactor coolant of 4.0 μCi/gm dose equivalent I-131. The iodine concentrations in the reactor coolant are listed below in μCi/gm.

	Equilibrium Iodine	Iodine Spiking
I-131	8.5E-02	1.7E+00
I-132	7.6E-01	1.5E+01
I-133	5.7E-01	1.1E+01
I-134	1.3E+00	2.7E+01
I-135	8.1E-01	1.6E+01

Because of its short half-life, N-16 is not considered in the analysis.

15.6.4.5.2 Fission Product Transport to the Environment

The transport pathway is a direct unfiltered release to the environment. The MSIV detection and closure time of 5.5 sec results in a discharge of 27,750 lb of steam and 112,250 lb of liquid from the break. Assuming all the activity in this discharge becomes airborne, the release of activity to the environment is presented in Table 15.6-3.

15.6.4.5.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.6-4 and are a small fraction of the guidelines of 10 CFR 50.67.

15.6.5 Loss-of-Coolant Accidents (Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary - Inside Containment)

The reload fuel vendor has determined that the loss-of-coolant accidents (LOCAs) event is not a limiting event for the current reload cycle. However, the DBA-LOCA event was re-analyzed during the implementation of EPU and, therefore, this subsection describes the current analysis performed by the GEH as part of EPU implementation. The radiological consequences represent the calculation of record following analyses associated with the alternative source term and EPU. For additional information on the relationship between analysis performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

This event involves the postulation of a spectrum of piping breaks inside containment varying in size, type, and location. The break type includes steam and/or liquid process system lines. This event is also coupled with severe natural environmental conditions including earthquake coincidence.

The event has been analyzed quantitatively in Sections 6.3, Emergency Core Cooling Systems; 6.2, Containment Systems; 7.3 and 7.1, Instrumentation and Controls; and 8.3, Onsite Power Systems. Therefore, the following discussion provides only new information not presented in the subject sections. All other information is covered by cross-referencing.

The postulated event represents the envelope evaluation for liquid or steam line failures inside containment.

Note: For additional discussions supporting operation with Feedwater Heater(s) Out of Service (FWHOS), Single Loop Operation (SLO), and operation in the Maximum Extended Operating Domain (MEOD), refer to Appendices 15B, 15C and 15D, respectively.

15.6.5.1 Identification of Causes and Frequency Classification

15.6.5.1.1 Identification of Causes

There are no realistic, identifiable events which would result in a pipe break inside the containment of the magnitude required to cause a loss-of-coolant accident coincident with safe shutdown earthquake plus SACF criteria requirements. The subject piping is designed of high quality, to strict emergency code and standard criteria, and for severe seismic and environmental conditions. However, since such an accident provides an upper limit estimate to the resultant effects for this category of pipe breaks, it is evaluated without the causes being identified.

15.6.5.1.2 Frequency Classification

This event is certainly categorized as a limiting fault.

15.6.5.2 Sequence of Events and Systems Operation

15.6.5.2.1 Sequence of Events

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

15.6.5.2.1.1 Deleted

15.6.5.2.2 Systems Operations

Accidents that could result in the release of radioactive fission products directly into the containment are the results of postulated nuclear system primary coolant pressure boundary pipe breaks. Possibilities for all pipe breaks sizes and locations are examined in Sections 6.2 and 6.3, including the severance of small process system lines, the main steam lines upstream of the flow restrictors, and the recirculation loop pipelines. The most severe nuclear system effects and the greatest release of radioactive material to the containment result from a complete circumferential break of one of the two recirculation loop pipelines. The minimum required functions of any reactor and plant protection system are discussed in Sections 6.2, 6.3, 7.2, 7.3, and 8.3, and Appendix 15A.

15.6.5.2.3 The Effect of Single Failures and Operator Errors

Single failures and operator errors have been adequately considered in the analysis of the entire spectrum of primary system breaks. The consequences of a LOCA with considerations for SACF and SOE occurrence are shown to be fully accommodated without the loss of any required safety function. See Appendix 15A for further details.

15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Model

The analytical methods and associated assumptions which are used in evaluating the consequences of this accident are considered to provide ultra-conservative assessment of the expected consequences of this very improbable event.

The details of these calculations, their justification, and bases for the models are developed in Sections 6.2, 6.3, 7.2, 7.6, 8.3, and Appendix 15A.

15.6.5.3.2 Input Parameters and Initial Conditions

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

A sensitivity study and discussion of the axial power shapes used in LOCA analyses is given in Reference 2.

15.6.5.3.3 Results

Results of this event are given in detail in Sections 6.2 and 6.3. The temperature and pressure transients resulting as a consequence of this accident are insufficient to cause perforation of the fuel cladding. Therefore, no fuel damage results from this accident. The containment integrity is maintained. Post accident tracking instrumentation and control is assured. Continued long term core and containment cooling is demonstrated. Radiological input is minimized and within limits. Continued operator control and surveillance is examined and guaranteed.

15.6.5.3.4 Consideration of Uncertainties

This event was conservatively analyzed; see Sections 6.2, 6.3, 7.3, 7.6, 8.3, and Appendix 15A for details.

15.6.5.4 Barrier Performance

The design basis for the containment is to maintain its integrity and experience normal stresses after the instantaneous rupture of the largest single primary system piping within the structure while also accommodating the dynamic effects of the pipe break at the same time an SSE is also occurring. Therefore, any postulated loss-of-coolant accident does not result in exceeding the containment design limit. For details and results of the analyses, see Sections 3.8, 3.9, and 6.2.

15.6.5.5 Radiological Consequences

The following potential pathways for transport of fission products from the primary containment to the environment following a LOCA have been identified:

a. Containment leakage

b. Leakage from the Main Steam Isolation Valves

c. Liquid leakage outside primary containment

These pathways are described in detail in this section. Leakage pathways associated with secondary containment bypass through the instrument and service air piping and water leakage into the spent fuel pool through the Horizontal Fuel Transfer System have also been assessed and found to be very small relative to the three pathways reported above. The results of these minor leakage pathways have been included in the reported LOCA dose results.

The methods, assumptions, and conditions used to evaluate this accident are in accordance with those guidelines set forth in Regulatory Guide 1.183 (Ref. 1) using the alternative source term described in 10CFR 50.67 (Ref. 4).

A schematic of the transport pathways, a, b, and c above is shown in Figure 15.6-2.

15.6.5.5.1 Fission Product Release From Fuel

The core source terms are based in a high-exposure core operating at 4496 MWt and include fission products from plutonium isotopes. These inventories are listed in Table 15.6-9 and are confirmed to be applicable to the GGNS reload fuel types.

The core source terms are released in phases as the core degrades consistent with the guidance in Regulatory Guide 1.183. For the first 2 minutes, as the reactor depressurizes and fuel temperatures begin to rise, no source terms are released from the fuel. During the next 30 minutes, 5 percent of the core inventories of noble gases, halogens, and alkali metals are released as the fuel rods begin to fail, releasing their gap activity. Then, for the next 90 minutes, the fuel melts and relocates to the bottom of the vessel, releasing significant quantities of volatile source terms as well as small fractions of less volatile nuclides. Two hours after the onset of gap release, the core damage is halted with the injection of ECCS into the vessel. Table 15.6-10 lists the source term groups, nuclides, timing, and release fractions for this evaluation.

15.6.5.5.2 Containment Leakage

As the core source terms are released into the drywell, a fraction of them will be carried into the containment via the pool bypass and through the suppression pool. Considering the core source terms are released after the blowdown, no significant flows through the suppression pool would be expected until the reactor is re-flooded. The dose model assumes a flow of 3000 cfm from the drywell into the lower containment region for the first two hours. At two hours, these volumes are assumed to become well-mixed with each other due to the steam released from ECCS injection with no credit for any potential suppression pool scrubbing. Although the drywell purge compressors would tend to drive drywell atmosphere into the containment, no further communication between these nodes is conservatively assumed. Elemental halogens and particulate source terms are removed from the drywell atmosphere by plate-out and natural deposition.

The containment is modeled with two nodes representing the upper portion of the containment (above El. 208'), which is exposed to containment spray, and the lower annulus region, to which the drywell leaks. These volumes communicate at a rate of 2 exchanges of the lower volume per hour or approximately 18,700 cfm. The containment spray system is assumed to be initiated at 30 minutes. This spray will remove elemental and particulate source terms from the sprayed region and increase the mixing rate between the regions to 70,000 cfm as calculated with the methodology in Reference 5. Containment spray is modeled to continue for 24 hours due to the expected radiation levels in containment. Elemental halogens and particulate source terms are removed from the containment atmosphere by plate-out.

During the first 24 hours, both containment nodes leak into secondary containment at a rate of 0.385 percent per day. After 24 hours, containment pressure has decayed to the point that this leak rate drops in half.

The post-accident suppression pool chemistry was assessed considering production of nitric acid from water radiolysis, hydrochloric acid from radiolysis of chloride-bearing cable jackets, hydriodic acid, and cesium hydroxide. The injection of sodium pentaborate solution from either the Standby Liquid Control system or the Condensate Storage Tank was demonstrated to sufficiently buffer the post-accident suppression pool and maintain dissolved iodine in solution. Therefore, no iodine re-evolution is modeled in the radiological evaluation.

The secondary containment is modeled to achieve an adequate negative pressure to prevent exfiltration in 3 minutes after the LOCA. Any source terms reaching the secondary containment before 3 minutes are immediately released to the environment. No credit is taken for holdup or dilution in the auxiliary building. All containment leakage is immediately directed into the enclosure building, where a mixing fraction of 50 percent is applied. The Standby Gas Treatment (SGTS) draws 4000 cfm from this volume through a charcoal bed and HEPA filter with an additional 1 cfm bypass from unidentified sources. The charcoal bed removes 99% of the elemental and organic halogens while the HEPA filter removes 99% of the particulates. The filtered SGTS flow is released to the environment from the SGTS vent on the roof of the auxiliary building.

In the control room, manual isolation of the unfiltered outside air intake is credited at 20 minutes, terminating the 2000 cfm intake flow. At this time, the Standby Fresh Air Supply system is initiated in its recirculation mode. In this mode, 4000 cfm of control room atmosphere is drawn through a HEPA filter, which removes 99% of the particulates, and returned to the control room.

A control room inleakage rate of 2000 cfm is assumed to begin at this point. After 3 days, the Standby Fresh Air Supply system is re-aligned to draw 4000 cfm of fresh air into the control room through the HEPA filter for the remainder of the accident. An additional 10 cfm of unfiltered inleakage is assumed from ingress and egress for the duration of the accident.

The parameters applied in the containment leakage portion of the LOCA radiological analysis are listed in Table 15.6-5. The model is illustrated in Figure 15.6-3.

15.6.5.5.3 MSIV Leakage

The MSIVs are assumed to leak at a total leakage rate of 250 scfh through all four steamlines with no valve exceeding 100 scfh. An MSIV is assumed to fail open with the closed MSIV on this line leaking at the maximum rate. Source terms from this steamline leak directly from the drywell atmosphere to the turbine building from which they are immediately released to the environment via the turbine building vent. After no later than 20 minutes, all MSIV leakage is directed to secondary containment when the MSIV Leakage Control System is manually initiated. The secondary containment and control room models previously described are applied to this leakage.

The additional parameters applied in the MSIV leakage portion of the LOCA radiological analysis are listed in Table 15.6-6. The MSIV leakage model is included in Figure 15.6-3.

15.6.5.5.4 Liquid Leakage

A significant portion of the released activity will eventually be deposited into the suppression pool. This water is recirculated through the secondary containment by the ECCS. Potential leakage of contaminated liquids following a design basis accident, can result in the release of radioisotopes outside the containment. Although the GGNS design provides barriers to such releases in accordance with regulatory requirements, these barriers must be assumed to pass some limited amount of leakage in order to allow for realistic equipment performance characteristics and testing methods.

The suppression pool activity is conservatively modeled by assuming that all soluble source terms are immediately deposited into the pool upon release from the fuel. The concentration of the source terms is maximized by assuming minimum pool volume.

Beginning at 3 minutes, a leakage rate of 1.12* gallons per minute into the secondary containment was evaluated based on potential leakage sources including penetration leakage, system boundary valve leakage, or leakage from Engineered Safety Feature (ESF) components. Ten percent of the halogens released into secondary containment and control room models previously described are applied to this leakage.

The additional parameters applied in the liquid leakage portion of the LOCA radiological analysis are listed in Table 15.6-7. The liquid leakage model is included in Figure 15.6-3.

15.6.5.5.5 Offsite Dose Calculations

Doses are calculated at the site boundary and low population zone (LPZ) using the 5 percent probability level dispersion factors (χ/Q) listed in Table 15.6-12.

15.6.5.5.6 Control Room Habitability

Pertinent data to calculate the dose received by an operator are listed in Table 15.6-13. Control room χ/Q values are shown in Table 15.6-12.

15.6.5.5.7 Results

The calculated offsite doses due to leakage of containment atmosphere, MSIV leakage, and liquid leakage for the design basis analysis are presented in Table 15.6-14 and are well within the guidelines of Standard Review Plan 15.0.1.

The calculated doses to the control room personnel are presented in Table 15.6-14 and are within the guidelines of 10 CFR 50.67.

15.6.6 Feedwater Line Break-Outside Containment

The reload fuel vendor has determined that the feedwater line break outside containment event is not a limiting event for the current reload cycle. Therefore, this subsection describes the original analysis performed by the NSSS vendor for the initial cycle which remains the current licensing basis for GGNS. For additional information on the relationship between analysis

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*The current LOCA radiological analysis conservatively applies a leakage rate higher than this value.

performed by the NSSS vendor for the initial cycle and the analysis by the reload vendor for the current cycle, refer to Section 15.0.

In order to evaluate large liquid process line pipe breaks outside containment, the failure of a feedwater line is assumed to evaluate the response of the plant design to this postulated event. The postulated break of the feedwater line, representing the largest liquid line outside the containment, provides the envelope evaluation relative to this type of occurrence. The break is assumed to be instantaneous, circumferential, the downstream of the outermost isolation valve. Refer to Figure 15.6-4.

A more limiting event from a core performance evaluation standpoint (feedwater line break inside containment) has been quantitatively analyzed in Section 6.3, Emergency Core Cooling Systems. Therefore, the following discussion provides only new information not presented in Section 6.3. All other information is covered by cross-referencing to appropriate Chapter 6 sections.

15.6.6.1 Identification of Causes and Frequency Classification

15.6.6.1.1 Identification of Causes

A feedwater line break is assumed without the cause being identified. The subject piping is designed to high quality, to strict emergency codes and standards, and to severe seismic environmental requirements.

15.6.6.1.2 Frequency Classification

15.6.6.2 Sequence of Events and Systems Operation

15.6.6.2.1 Sequence of Events

The sequence of events is shown in Table 15.6-15.

15.6.6.2.1.1 Deleted

15.6.6.2.2 Systems Operations

It is assumed that the normally operating plant instrument and controls are functioning. Credit is taken for the actuation of the reactor protection system (safety/relief valves, ECCS, and control rod drive) and plant protection system (RHR heat exchangers) are assumed to function properly to assure a safe shutdown.

The ESF systems and RCIC/HPCS systems are assumed to operate normally.

15.6.6.2.3 The Effect of Single Failures and Operator Errors

The feedwater line outside the containment is a special case of the general loss-of-coolant accident break spectrum considered in detail in Section 6.3. The general single-failure analysis for loss-of-coolant accidents is discussed in detail in subsection 6.3.3.3. For the feedwater line break outside the containment, since the break is isolatable, either the RCIC or the HPCS can provide adequate flow to the vessel to maintain core cooling and prevent fuel rod clad failure. A single failure of either the HPCS or the RCIC would still provide sufficient flow to keep the core covered with water. See Section 6.3 and Appendix 15A for detailed description of analysis.

15.6.6.3 Core and System Performance

15.6.6.3.1 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 of the feedwater piping lines external to the containment. The accident is postulated to occur at the input parameters and initial conditions as given in Table 6.3-2.

15.6.6.3.2 Qualitative Results

The feedwater line break outside the containment is less limiting than either the steam line breaks outside the containment (analysis presented in Sections 6.3 and/or 15.6.4), the feedwater line break inside the containment (analysis presented in subsections 6.3.3 and 15.6.5). It certainly is far less limiting than the design basis accident (the recirculation line break analysis presented in subsections 6.3.3 and 15.6.5).

The reactor vessel is isolated on low-low water level and the RCIC and the HPCS together restore the reactor water level to the normal elevation. The fuel is covered throughout the transient and there are no pressure or temperature transients sufficient to cause fuel damage.

15.6.6.3.3 Consideration of Uncertainties

This event was conservatively analyzed and uncertainties were adequately considered (see Section 6.3 for details).

15.6.6.4 Barrier Performance

This accident is beyond the reactor coolant pressure boundary. It does not result in failure of any fuel.

15.6.6.5 Radiological Consequences

The activity release for this event is much less than the release evaluated in subsection 15.6.4.5 (Main Steam Line Break). The consequences in subsection 15.6.4.5 envelope the consequences of this event.

15.6.7 References

1. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000.

2. NEDO-20566, Section II.A.4.C.4, Page II-188.

3. Deleted

4. 10CFR50.67, "Accident Source Terms".

5. NUREG/CR-0304, "Mixing of Radiolytic Hydrogen Generated Within a Containment Compartment Following a LOCA," G.J. Wilcott, Jr. and Richard G. Gido, July 1978.

TABLE 15.6-1: SEQUENCE OF EVENTS FOR STEAM LINE BREAK OUTSIDE CONTAINMENT

Time-sec	Event
0	Guillotine break of one main steam line outside primary containment.
~0.5	High steam line flow signal initiates closure of main steam line isolation valve.
<1.0	Reactor begins scram.
≤5.5	Main steam isolation valves fully closed.
~10	Safety/relief valves open on high vessel pressure. The valves open and close to maintain vessel pressure at approximately 1000 psi.
~30	RCIC and HPCS would initiate on low water level, L2 (RCIC considered unavailable, HPCS assumed single failure and therefore not available).
~90	Reactor water level above core begins to drop slowly due to loss of steam through the safety valves. Reactor pressure still at approximately 1000 psi.
~310	Low reactor water level initiates automatic ADS logic starting high drywell pressure bypass timer.
~1015	Time exceeds time delay from both the ADS initiation timer and high drywell pressure bypass timer. ADS automatically initiated. Vessel depressurizes rapidly.
~1170	Low pressure ECCS systems initiated with reactor fuel partially uncovered.
~1270	Core effectively reflooded and clad temperature heatup terminated. No fuel rod failure.

TABLE 15.6-2: STEAM LINE BREAK ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS

		Design Basis Assumptions
I.	Data and assumptions used to estimate radioactive source from postulated accidents	
	A. Fuel damaged	None
	B. Release of activity by nuclide	Table 15.6-3
	C. Iodine fractions	
	(1) Elemental	0.0485
	(2) Organic	0.0015
	(3) Particulate	0.9500
	D. Reactor coolant activity before the accident	15.6.4.5.1
II.	Data and assumptions used to estimate activity released	
	A. Isolation Valve Closure Time (sec)	5
III.	Data and assumptions used to estimate control room activity	
	A. Inleakage rate (cfm)	2010
	B. Volume (cu.ft.)	2.53E+05
	C. Control Room Fresh Air System	
	(1) Initiation Time (min)	20
	(2) Recirculation Flow Rate (cfm)	4000
	(3) Filter Efficiency (%)	
	(i) Elemental iodine	0

TABLE 15.6-2: STEAM LINE BREAK ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS (CONTINUED)

		Design Basis Assumptions
	(ii) Organic iodine	0
	(iii) Particulates	99
IV.	Dispersion Data (s/cu.m.)	
	A. Site Boundary (696m)	6.50E-04
	B. Control Room	2.20E-03

TABLE 15.6-3: STEAM LINE BREAK ACCIDENT (DESIGN BASIS ANALYSIS) ACTIVITY RELEASE TO ENVIRONMENT (CURIES)

Equilibrium Iodine Case

Isotope	Elemental	Organic	Particulate
I-131	2.11E-01	6.53E-03	4.13E+00
I-132	1.89E+00	5.84E-02	3.70E+01
I-133	1.42E+00	4.38E-02	2.77E+01
I-134	3.22E+00	9.98E-02	6.32E+01
I-135	2.00E+00	6.21E-02	3.93E+01

Iodine Spiking Case

Isotope	Elemental	Organic	Particulate
I-131	4.22E+00	1.31E-01	8.27E+01
I-132	3.72E+01	1.15E+00	7.30E+02
I-133	2.73E+01	8.45E-01	5.35E+02
I-134	6.69E+01	2.07E+00	1.31E+03
I-135	3.97E+01	1.22E+00	7.78E+02

Noble Gase Release (Applicable to Both Iodine Cases)

Isotope	
Kr-83m	7.43E-02
Kr-85m	1.26E-01
Kr-85	5.03E-04
Kr-87	4.15E-01
Kr-88	4.15E-01
Kr-89	2.64E+00
Xe-131m	4.15E-04
Xe-133m	6.17E-03
Xe-133	1.76E-01
Xe-135m	5.54E-01
Xe-135	4.78E-01
Xe-137	3.27E+00
Xe-138	1.89E+00

TABLE 15.6-4: STEAM LINE BREAK ACCIDENT (DESIGN BASIS ANALYSIS) RADIOLOGICAL RESULTS OF A PUFF RELEASE

	TOTAL EFFECTIVE DOSE EQUIVALENT (TEDE) (REM)	
	Equilibrium Iodine	**Iodine Spiking**
Exclusion Area (696 m)	1.30E-01	2.58E+00
Control Room	<1.53E-01	<3.01E+00

TABLE 15.6-5: CONTAINMENT LEAKAGE DOSE MODEL PARAMETERS

	Design Basis Assumptions
I. Data and assumption used to estimate radioactive source term from postulated accidents	
A. Power level	4496 MWt
B. Core source terms	See Table 15.6-9
C. Release fractions	See Table 15.6-10
D. Halogen chemical species	
(1) Organic	0.15%
(2) Elemental	4.85%
(3) Particulate	95%
II. Data and assumptions used to estimate activity released from containment pathway	
A. Node volumes (ft³)	
(1) Drywell	2.7E5
(2) Sprayed Containment	8.4E5
(3) Unsprayed Containment	5.6E5
(4) Secondary Containment (based on 50% mixing efficiency)	3.0E5
B. Flows between nodes (cfm)	
(1) Drywell to Unsprayed Containment	
(a) 0 hours - 2 hours	3.0E3
(b) 2 hours	Well mixed
(2) Unsprayed Containment to Drywell	
(a) 0 hours - 2 hours	0
(b) 2 hours	Well mixed
(3) Unsprayed Containment to Sprayed Containment	
(a) 0 hours - 30 min	1.87E5
(b) 30 min - 24 hours	7.0E5
(c) 24 hours - 30 days	1.87E5
(4) Sprayed Containment to Unsprayed Containment	
(a) 0 hours - 30 min	1.87E5
(b) 30 min - 24 hours	7.0E5

TABLE 15.6-5: CONTAINMENT LEAKAGE DOSE MODEL PARAMETERS (Continued)

	Design Basis Assumptions
(c) 24 hours - 30 days	1.87E5
C. Primary containment leak rate (%/day)	
(1) 0-24 hours	0.385
(2) 24 hours	0.1925
D. Spray Removal	
(1) Aerosols	8.56 per hour
Time to reach DF of 50	3 hours
(2) Elemental	20 per hour
Time to reach DF of 200	2.8 hours
E. Elemental plateout removal rate (per hour)	
(1) Drywell (0-7 hours)	0.866
(2) Lower Containment (0-2.8 hours)	1.092
(3) Upper Containment (0-2.8 hours)	0.682
F. Aerosol natural deposition removal rate in drywell (per hour)	
0 - 0.5 hours	0.8021
0.5 - 2.0 hours	0.3207
2.0 - 5.0 hours	1.109
5.0 - 8.3 hours	0.6617
8.3 - 12 hours	0.5821
12 - 19.4 hours	0.5514
19.4 - 24 hours	0.5361
24 hours - 30 days	0.0
G. Secondary Containment	
(1) SGTS Flow	4000 cfm
(2) Filter Train Removal Efficiency	
(a) Organic Iodine	99%
(b) Elemental Iodine	99%

TABLE 15.6-5: CONTAINMENT LEAKAGE DOSE MODEL PARAMETERS (Continued)

	Design Basis Assumptions
(c) Particulates	99%
(3) Bypass Flow	1 cfm
III. Dispersion Data	See Table 15.6-12
IV. Breathing Rates (m^3/s)	
(1) 0 - 8 hrs	3.5E4
(2) 8 - 24 hrs	1.8E4
(3) 1 day - 30 days	2.3E4
V. Dose Conversion Factors	Federal Guidance Reports 11 and 12

TABLE 15.6-5A: Deleted

TABLE 15.6-6: MSIV LEAKAGE ADDITIONAL DOSE MODEL PARAMETERS

	Design Basis Assumptions
I. Data and assumptions used to estimate activity released from MSIV pathway	
A. Leakage Rates	
(1) MSIV on steamline with stuck-open MSIV	
(a) 0 hours - 24 hours	100 scfh
(b) 24 hours - 30 days	50 scfh
(2) Total for all remaining steamlines	
(a) 0 hours - 24 hours	150 scfh
(b) 24 hours - 30 days	75 scfh
B. Release pathway for leakage past outboard MSIV	
(1) 0 - 20 minutes	To environment
(2) 20 minutes - 30 days	To secondary containment

TABLE 15.6-7: LIQUID LEAKAGE ADDITIONAL DOSE MODEL PARAMETERS

	Design Basis Assumptions
I. Data and assumptions used to estimate radioactive source term from postulated accidents	
A. Power level	4496 MWt
B. Core source terms	See Table 15.6-9
C. Release fractions	See Table 15.6-10
I. Data and assumptions used to estimate activity released from containment pathway	
A. Suppression pool volume	1.71E5 ft^3
B. Flash fraction	10%
C. Leakage rates (gpm)	
(1) 0 - 3 minutes	0
(2) 3 minutes - 30 days	2.24
D. Halogen chemical species	
(1) Organic	3%
(2) Elemental	97%
(3) Particulate	0%

TABLE 15.6-8: Deleted

TABLE 15.6-9: LOSS-OF-COOLANT ACCIDENT CORE INVENTORY BY NUCLIDE

Isotope	Core Inventory (Ci at t=0)	Isotope	Core Inventory (Ci at t=0)	Isotope	Core Inventory (Ci at t=0)
Co-58	6.14E+05	Ru-105	1.40E+08	Xe-135m	4.80E+07
Co-60	1.51E+06	Ru-106	8.03E+07	Cs-134	3.27E+07
Br-82	8.51E+05	Rh-103m	1.77E+08	Cs-136	1.00E+07
Br-83	1.49E+07	Rh-105	1.32E+08	Cs-137	1.86E+07
Br-84	2.56E+07	Rh-106	8.07E+07	Cs-138	2.25E+08
Kr-85	1.69E+06	Sb-125	2.27E+06	Ba-137m	1.76E+07
Kr-85m	3.15E+07	Sb-127	1.37E+07	Ba-139	2.21E+08
Kr-87	5.98E+07	Sb-129	4.03E+07	Ba-140	2.13E+08
Kr-88	8.43E+07	Te-127	1.37E+07	La-140	2.31E+08
Rb-86	3.32E+05	Te-127m	1.84E+06	La-141	2.01E+08
Sr-89	1.14E+08	Te-129	3.97E+07	La-142	1.94E+08
Sr-90	1.35E+07	Te-129m	5.91E+06	Ce-141	2.02E+08
Sr-91	1.43E+08	Te-131	1.08E+08	Ce-143	1.87E+08
Sr-92	1.54E+08	Te-131m	1.79E+07	Ce-144	1.66E+08
Y-90	1.40E+07	Te-132	1.73E+08	Pr-143	1.80E+08
Y-91	1.48E+08	Te-133	1.37E+08	Pr-144	1.67E+08
Y-91m	8.33E+08	Te-133m	8.79E+07	Pr-144m	1.99E+06
Y-92	1.56E+08	Te-134	1.98E+08	Nd-147	8.10E+07
Y-93	1.81E+08	I-131	1.22E+08	Np-239	2.56E+09
Zr-95	2.03E+08	I-132	1.76E+08	Pu-238	6.45E+05
Zr-97	2.03E+08	I-133	2.48E+08	Pu-239	5.40E+04
Nb 95	2.04E+08	I-134	2.71E+08	Pu-240	7.55E+04
Nb 97	2.05E+08	I-135	2.31E+08	Pu-241	2.54E+07
Nb 97m	1.92E+08	Xe-131m	1.37E+06	Am-241	3.01E+04
Mo 99	2.30E+08	Xe-133	2.38E+08	Cm-242	8.20E+06
Tc 99m	2.02E+08	Xe-133m	7.76E+06	Cm-244	5.97E+05
Ru 103	1.96E+08	Xe-135	8.55E+07		

TABLE 15.6-10: SOURCE TERM RELEASE GROUPS AND TIMING PERCENT OF CORE INVENTORY RELEASED

Group	Isotopes	Gap Release (0-30 min.)	In-Vessel Release (30-90 min.)
Noble Gases	Kr, Xe	5	95
Halogens	Br, I	5	25
Alkali Metals	Rb, Cs	5	20
Tellurium Metals	Te, Sb, Se	0	5
Barium, Strontium	Ba, Sr	0	2
Noble Metals	Ru, Rh, Pd, Mo, Tc, Co	0	0.25
Cerium	Ce, Pu, Np	0	0.05
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am	0	0.02

TABLE 15.6-11: Deleted

TABLE 15.6-12: 5% PROBABILITY - LEVEL χ/Q VALUES (SEC/M^3)

	Time Periods (Hrs)				
	0 - 2	2 - 8	8 - 24	24 - 96	96 - 720
Site (696m)	6.50-04	-	-	-	-
LPZ (3219m)	1.45-04	7.10-05	5.00-05	2.30-05	7.60-06

	Time Periods				
	0 - 2 hrs	2 hrs - 8 hrs	8 - 24 hrs	24 - 96 hrs	96 - 720 hrs
Control Room*	8.00-04	4.80-04	2.10-04	1.50-04	1.05-04

*Unit 1 releases only

TABLE 15.6-13: CONTROL ROOM PARAMETERS

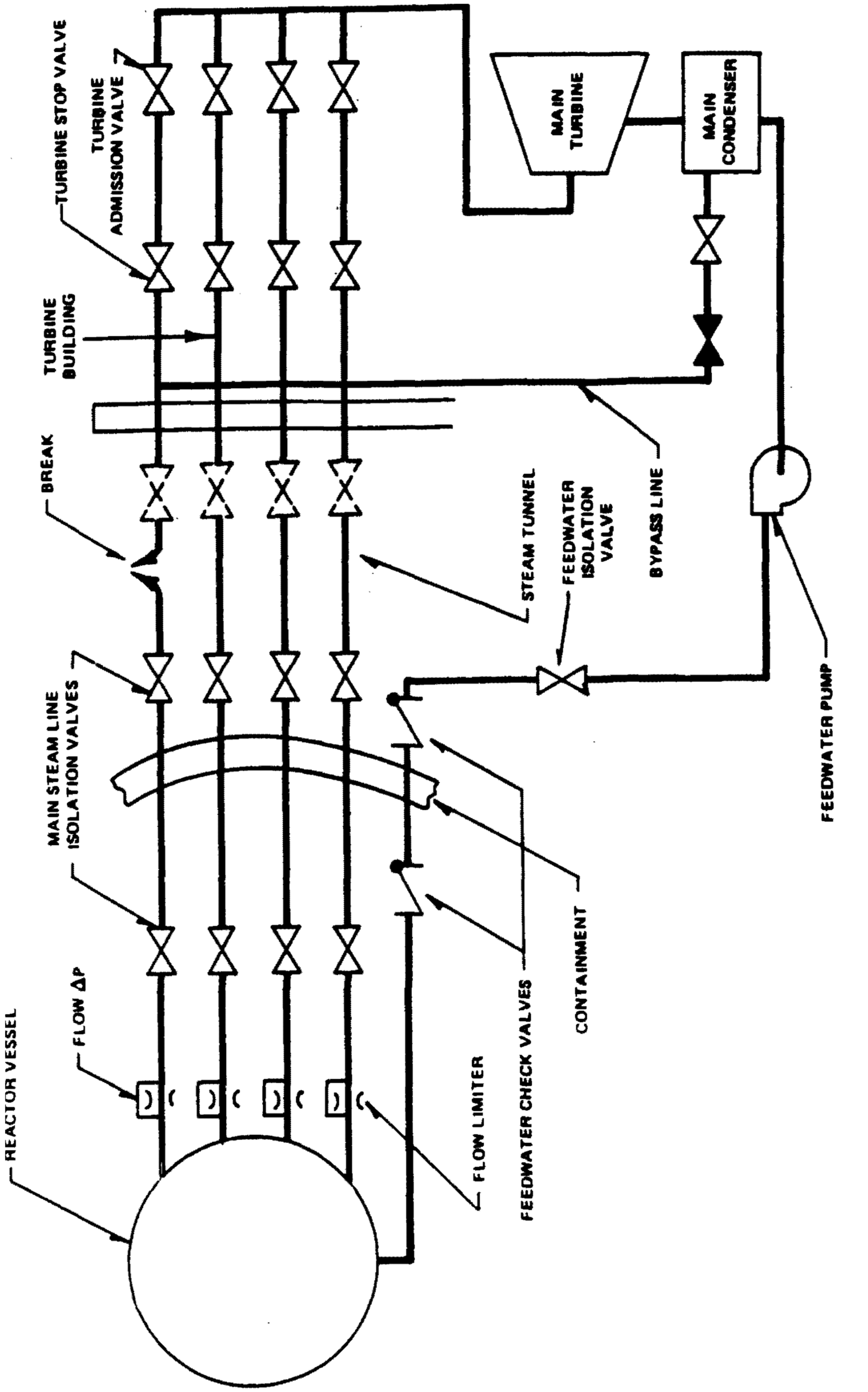
A.	Control Room Volume, ft^3	2.53+05
B.	Intake and Recirculation Filter Efficiencies (%)	
	(1) elemental iodine	0
	(2) organic iodine	0
	(3) particulate iodine	99
C.	Unfiltered Inleakage (cfm)	
	(1) inleakage	2.0+03
	(2) inleakage due to door openings	1.00+01
D.	Filtered Intake (cfm)	
	(1) 0 - 30 days	0.00
E.	Filtered Recirculation Rate (cfm)	
	(1) 20 min - 30 days	4.00+03
F.	Breathing Rates (m^3/sec)	
	(1) 0 - 720 hrs	3.50-04
G.	Occupancy Factors	
	(1) 0 - 8 hrs	1.00+00
	(2) 8 - 24 hrs	1.00+00
	(3) 24 - 96 hrs	6.00-01
	(4) 96 - 720 hrs	4.00-01

TABLE 15.6-14: LOSS-OF-COOLANT-ACCIDENT OFFSITE AND CONTROL ROOM PERSONNEL DOSES

DOSES (Rem)	TEDE
Site boundary (EAB) [696 m] (1.9 - 3.9 hrs)	<10.01
Low Population Zone (LPZ) [3219 m] (0 - 30 days)	<6.37
CONTROL ROOM PERSONNEL (0 - 30 days)	<4.24

TABLE 15.6-15: SEQUENCE OF EVENTS FOR FEEDWATER LINE BREAK OUTSIDE CONTAINMENT (INITIAL CYCLE ANALYSIS REMAINS THE CURRENT ANALYSIS FOR THIS EVENT)

Time-sec	Event
0	One feedwater line breaks.
0+	Feedwater line check valves isolate the reactor from the break.
<30	At low low-water reactor level RCIC would initiate, HPCS would initiate, MSIV closure would initiate, reactor scram would initiate and recirculation pumps would trip.
~2 min	The safety relief valves wculd open and close and maintain the reactor vessel pressure at approximately 1100 psig.
1 to 2 hours	Normal reactor cooldown procedure established.



GRAND GULF NUCLEAR STATION
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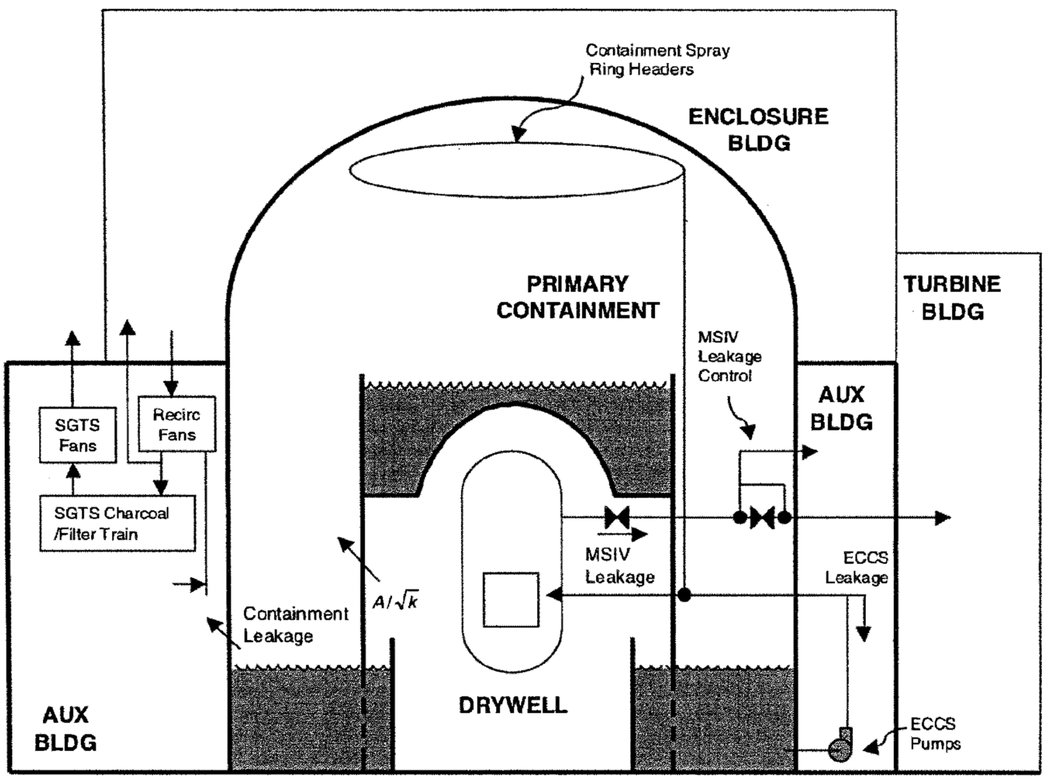
UPDATED FINAL SAFETY ANALYSIS REPORT

STEAM FLOW SCHEMATIC FOR STEAM BREAK OUTSIDE CONTAINMENT

FIGURE 15.6-1

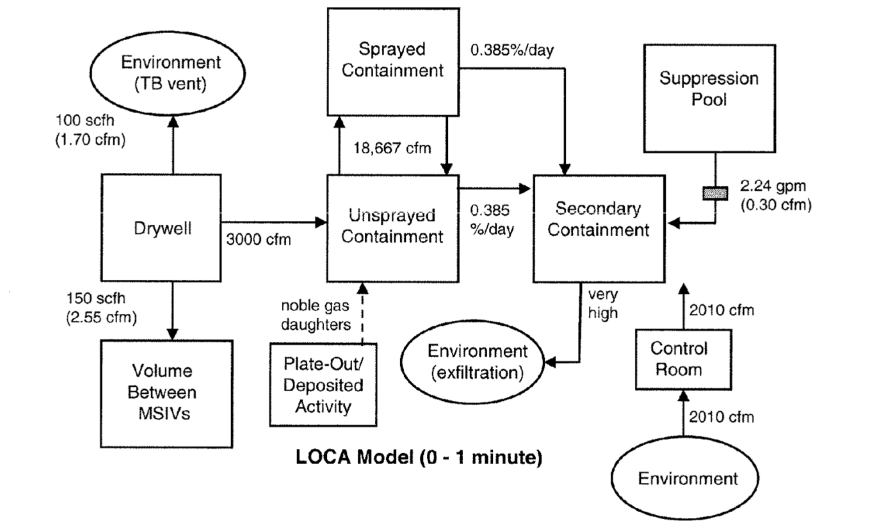
Figure 15.6-2
Post-LOCA Leakage Pathways

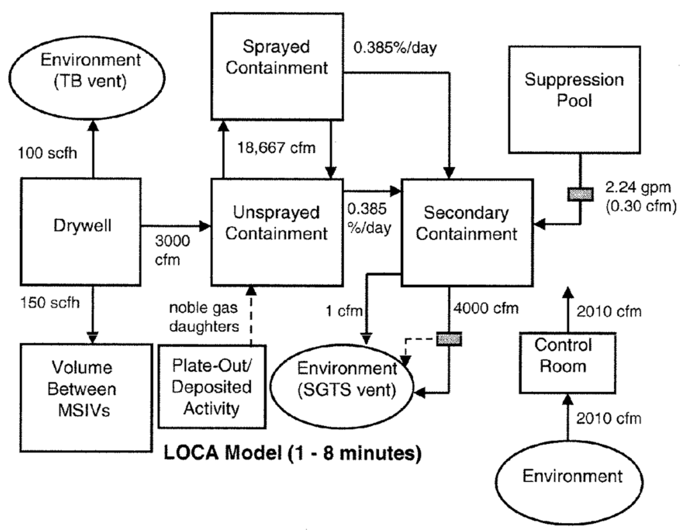


GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	Post-LOCA Leakage Pathways Figure 15.6-2

Figure 15.6-3

Local Analysis Model

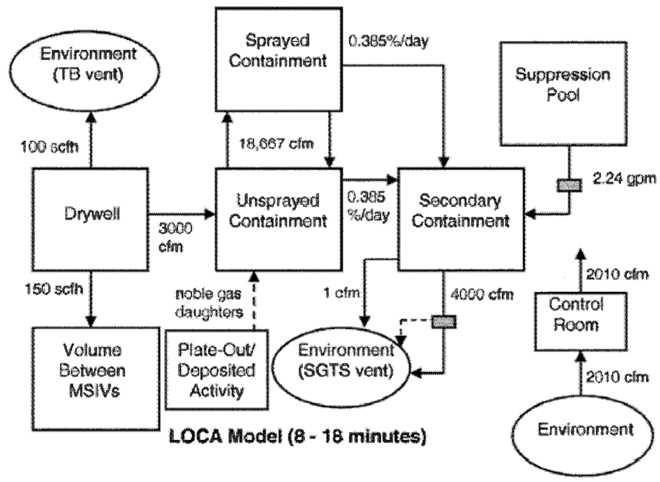




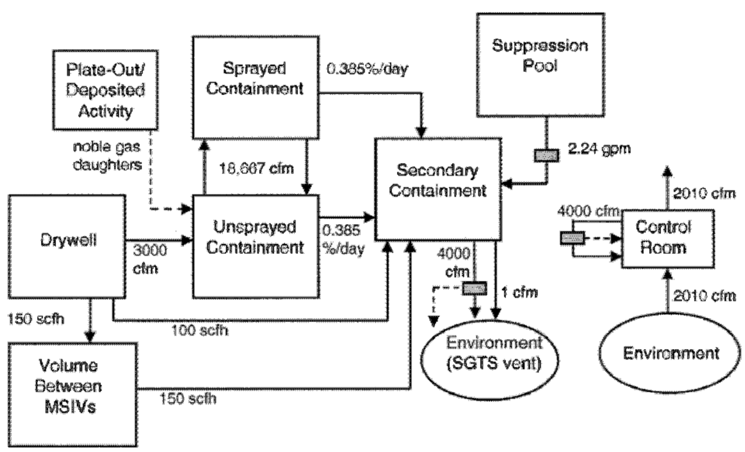
Grand Gulf Nuclear Station Unit 1 Updated Final Safety Analysis Report	LOCA Analytical Model Figure 15.6-3 Sheet 1 of 4

Figure 15.6-3 (cont.)

Local Analysis Model



LOCA Model (8 - 18 minutes)

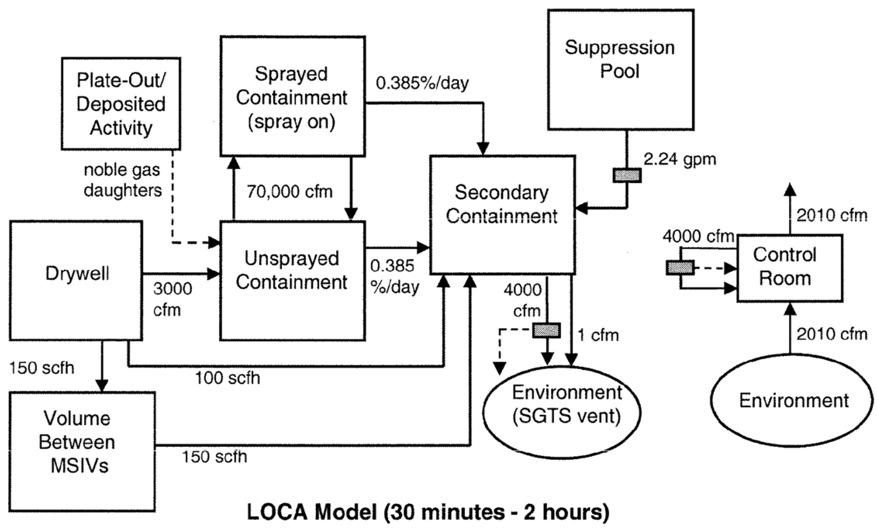


LOCA Model (18 - 30 minutes)

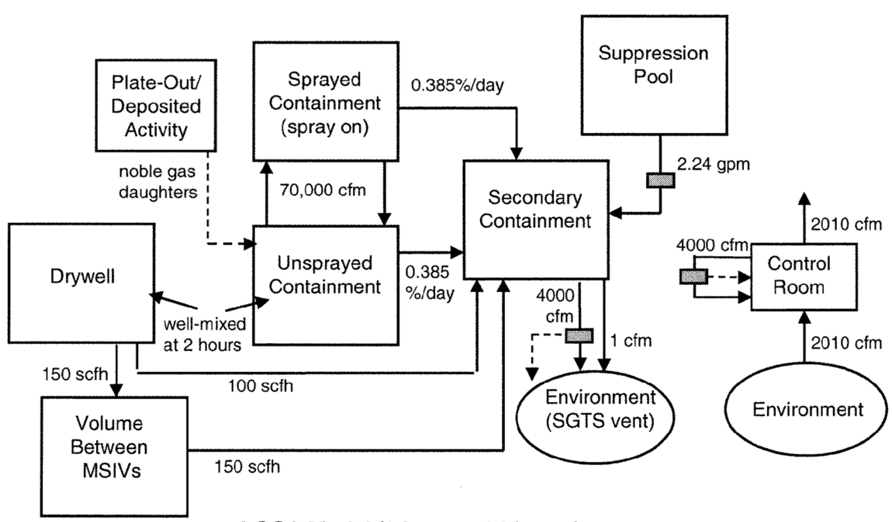
Grand Gulf Nuclear Station Unit 1 Updated Final Safety Analysis Report	LOCA Analytical Model Figure 15.6-3 (cont.) Sheet 2 of 4

Figure 15.6-3 (cont.)

Local Analysis Model



LOCA Model (30 minutes - 2 hours)

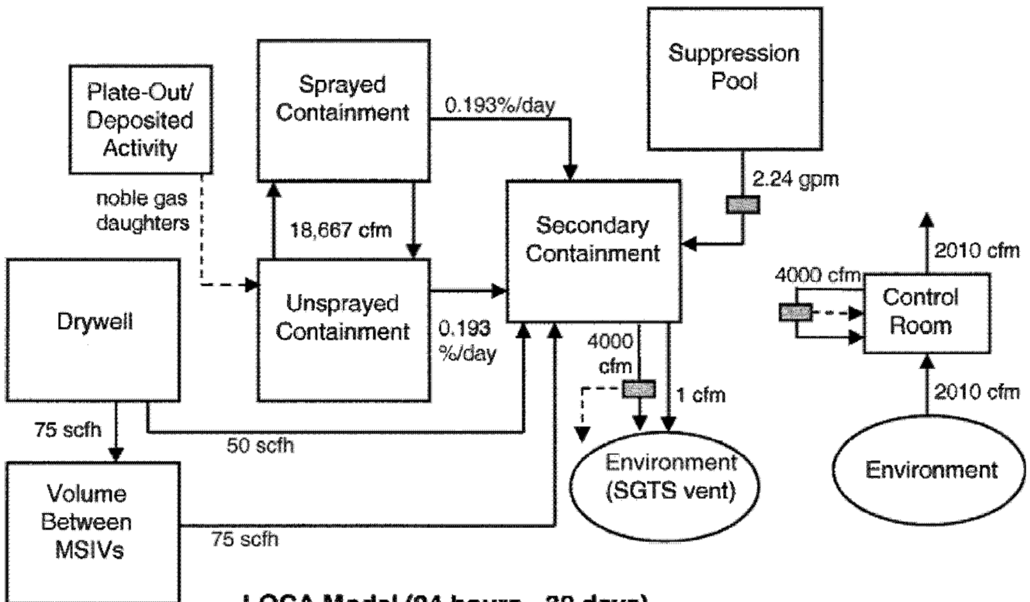


LOCA Model (2 hours - 24 hours)

Grand Gulf Nuclear Station Unit 1 Updated Final Safety Analysis Report	LOCA Analytical Model Figure 15.6-3 (cont.) Sheet 3 of 4

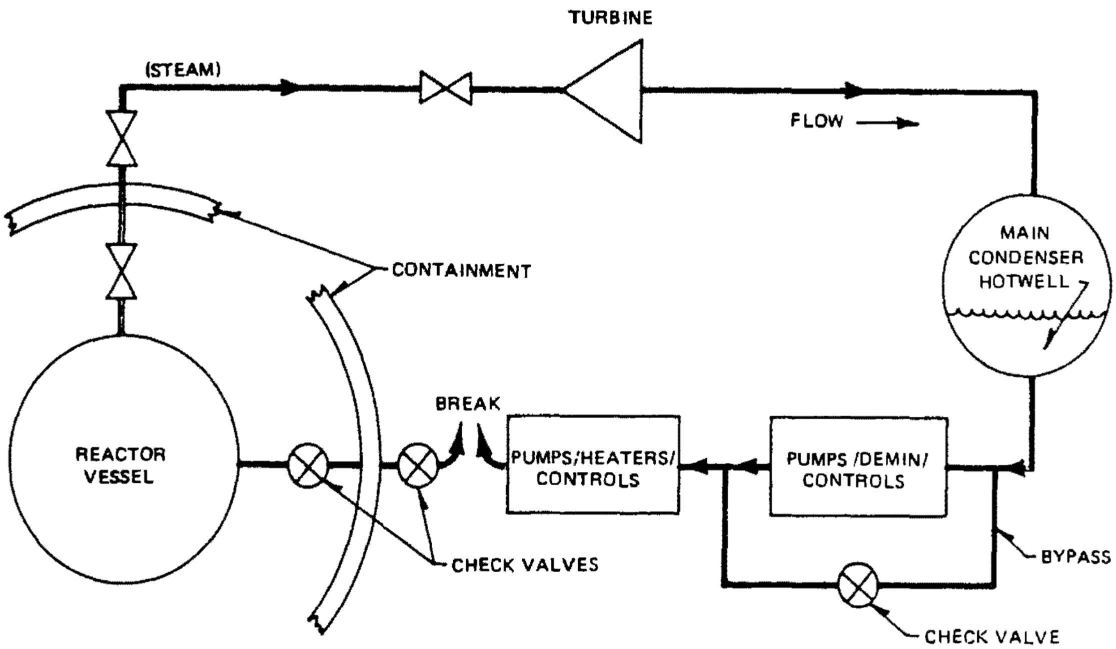
Figure 15.6-3 (cont.)

Local Analysis Model



LOCA Model (24 hours - 30 days)

Grand Gulf Nuclear Station Unit 1 Updated Final Safety Analysis Report	LOCA Analytical Model Figure 15.6-3 (cont.) Sheet 4 of 4



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	LEAKAGE PATH FOR FEEDWATER LINE BREAK OUTSIDE CONTAINMENT FIGURE 15.6-4

15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM AND COMPONENT

This section discusses radioactive releases from the following:

a. Offgas system leak or failure

b. Radioactive liquid waste system leak or failure (Release to the atmosphere)

c. Postulated radioactive releases due to liquid radwaste tank failure (Release to the ground water)

d. Design basis fuel handling accidents

e. Spent fuel cask drop accidents

15.7.1 Offgas System Leak or Failure

Offgas treatment system failure was examined to determine the releases from three major sources:

a. Charcoal adsorber failure

b. Delay line failure

c. Continued operation of the air ejector

15.7.1.1 Identification of Causes and Frequency Classification

15.7.1.1.1 Identification of Causes

Those events which could cause a gross failure in the offgas treatment system are:

a. A seismic occurrence

b. A hydrogen explosion in housing unit

c. Failure of spacially related equipment

Even though the offgas system is designed to NRC Branch Technical Position ETSB 11-1 (Rev. 1) requirements, an event more severe than the design requirements is arbitrarily assumed to occur, resulting in the failure of the offgas system. The seismic failure is the only event which could cause significant system damage.

The equipment and piping are designed to contain any hydrogen-oxygen detonation which has a reasonable probability of occurring. A detonation is not considered as a possible failure mode.

The charcoal filters have a built-in water suppression system to prevent gross fires. See subsection 9.5.1 for a discussion of the fire protection features built into these filters.

The system is reasonably isolated from other systems or components which could cause any serious interaction or failure.

The design basis, description, and performance evaluation of the subject system is given in Section 11.3.

15.7.1.1.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.1.2 Sequence of Events and System Operation

15.7.1.2.1 Sequence of Events

The sequence of events following this failure is shown in Table 15.7-1.

15.7.1.2.2 Identification of Operator Actions

Gross failure of this system may require manual isolation of this system from the main condenser. This isolation results in high condenser pressure and a reactor scram. The operator will monitor the turbine-generator auxiliaries and break vacuum as soon as possible. The operator must notify personnel to evacuate the area immediately and notify radiation protection personnel to survey the area and determine requirements for reentry. The time needed for these actions is about 2 minutes.

15.7.1.2.3 Systems Operation

In analyzing the postulated offgas system failure, no credit is taken for the operation of plant and reactor protection systems, or of engineered safety features. Credit is taken for functioning of normally operating plant instruments and controls and other systems only in assuming the following:

a. Capability to detect the failure itself - indicated by an alarmed increase in radioactivity levels seen by area radiation monitoring system, in an alarmed loss of flow in the offgas system, and in an alarmed increase in activity at the vent release

b. Capability to isolate the system and shutdown the reactor

c. Operational indicator and annunciators in the control room

15.7.1.2.4 The Effect of Single Failures and Operator Errors

After the initial system gross failure, the inability of the operator to actuate a system isolation could affect the analysis. However, the seismic event which is assumed to occur beyond the present plant design basis for non-safety equipment will undoubtedly cause the tripping of turbine or will lead to a load rejection. This will initiate a scram and negate a need for the operator to initiate a reactor shutdown via system isolation.

See Appendix 15A for a further detailed discussion of this subject.

15.7.1.3 Core and System Performance

The postulated failure results in a system isolation necessitating reactor shutdown because of loss of vacuum in the main condenser. This transient has been analyzed in subsection 15.2.5.

15.7.1.4 Barrier Performance

The postulated failure is the rupture of the offgas system pressure boundary. No credit is taken for performance of secondary barriers, except to the extent inherent in the assumed equipment release fractions discussed in subsection 15.7.1.5 below.

15.7.1.5 Radiological Consequences

The design basis analysis is based on NRC Standard Review Plan 15.7.1 and NRC Regulatory Guide 1.98 using the source term described in 10 CFR 100 and 10 CFR 50 GDC 19 (Ref. 6). Specific parametric values used in this evaluation are presented in Table 15.7-2.

15.7.1.5.1 Fission Product Release

The activity in the offgas system is based on the following conditions:

a. 6 scfm air inleakage

b. Continuous release of 399,000 µCi/sec noble gas after 30 minutes decay. A power level of 4496 MWt is used.

It is assumed that SJAE releases are from a break in the delay line and continue for 1 hour following the accident.

All of the noble gases and particulates in the delay line and charcoal vessels are assumed to be released over a period of 2 hours.

15.7.1.5.2 Fission Product Transport to the Environment

The transport pathway consists of direct release from the failed component to the environment. The release of activity to the environment is presented in Table 15.7-3.

15.7.1.5.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.7-4 and are well within the guidelines of 10 CFR 100 (Ref. 5) and 10 CFR 50 GDC 19 (Ref. 6).

15.7.2 Radioactive Liquid Waste System Leak or Failure (Release to the Atmosphere)

15.7.2.1 Identification of Causes and Frequency Classification

Radioactive releases considered include rupture of radwaste tanks, equipment malfunction, and small leaks in the lines transporting liquid radwaste to the system for processing. The most limiting of these failures is defined as an unexpected and uncontrolled release of the radioactive liquid stored in the evaporator bottoms tanks. The radwaste system tanks are non-seismic and are designed and constructed in accordance with Table 3.2-1 and Table 3.2-2. Rupture of these tanks is considered a limiting fault.

Note: [HISTORICAL INFORMATION] [The design information used in the dose calculation concerning the atmospheric release due to rupture of the evaporator bottoms tanks is historical design information. The probability of this accident is zero because the Radwaste Evaporators and the bottoms tanks are abandoned in place. This evaluation remains relevant to the original licensing basis and to the potential future use by GGNS of a high temperature radwaste concentration process. Also, the evaluation is bounding for an atmospheric release due to rupture of any other radioactive liquid waste system tank.]

15.7.2.2 Sequence of Events and Systems Operation

a. Event begins - failure occurs. An evaporator bottoms tank, which has the highest activity level, is assumed to rupture, releasing its entire contents to the radwaste building.

b. Area radiation alarms alert plant personnel.

c. Operator action begins.

The rupture of the evaporator bottoms tank would leave little recourse to the operator. No method of recontaining the gaseous phase discharge is available; isolation of the radwaste area, however, would minimize the results. High radiation alarms, both in the radwaste ventilation exhaust and in the radwaste area would alert the operator to the failure. No credit for any operator action has been taken in evaluating this event.

15.7.2.3 Core and System Performance

This failure is not expected to have any applicable effect on the core or NSSS safety performance.

15.7.2.4 Barrier Performance

This release occurs outside the containment, hence does not involve any barrier integrity aspects.

15.7.2.5 Radiological Consequences

The assumptions used to evaluate the rupture of the evaporator bottoms tank are listed in Table 15.7-5, and the radioactive inventory in the tank is listed in Table 15.7-6.

Offsite doses resulting from the rupture of the evaporator bottoms tank are presented in Table 15.7-7. These doses are based on design basis assumptions. As shown, they are less than 0.5 rem to the whole body and 1.5 rem to the thyroid at the site boundary. As discussed above, this event is historical and the design information used in this analysis is historical design information. However, the offsite dose consequences of this event bound any failure in the radioactive liquid waste system.

15.7.3 Postulated Radioactive Releases Due to Liquid Radwaste Tank Failure

15.7.3.1 Identification of Causes and Frequency Classification

An unspecified event causes the release of the contents of the tank containing the largest quantities of significant radionuclides in the liquid radwaste system. This is a RWCU phase separator decay tank in the radwaste building.

Postulated events that could cause release of the radioactive inventory of a RWCU phase separator decay tank are cracks in the vessel and operator error. The possibility of small cracks and consequent low-level release rates receives primary consideration in system and component design. The RWCU phase separator decay tanks are designed to operate at atmospheric pressure and 150 F maximum temperature, so the possibility of failure is considered small. A liquid radwaste release caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instructions.

Much of the exposition concerning the remote likelihood of a leakage or malfunction accident of the RWCU phase separator decay tanks applies equally to a complete release accident. The probability of a complete rupture or complete malfunction accident is, however, considered lower.

The liquid radwaste tanks are non-seismic and are designed and constructed in accordance with Table 3.2-1 and Table 3.2-2. Rupture of these tanks is considered a limiting fault.

The failure of a RWCU phase separator decay tank is considered a limiting fault.

15.7.3.2 Sequence of Events and System Operation

a. Event begins - RWCU phase separator decay tank fails and the contents are released into the radwaste building.

b. Area radiation alarms alert plant personnel.

c. Operator actions begin.

Should a release of liquid radioactive waste occur, floor drain sump pumps in the floor of the radwaste building will receive a high-water-level signal, activate automatically, and remove the spilled liquid.

In the evaluation of a liquid radwaste tank failure no credit is taken for operator action and it is assumed that liquid leaks from the building into the ground.

15.7.3.3 Core and System Performance

The failure of these liquid radwaste components does not directly affect the NSSS.

15.7.3.4 Barrier Performance

This event does not involve any containment barrier integrity.

15.7.3.5 Radiological Consequences

The radiological analysis and results are presented in subsection 2.4.13.3.

15.7.4 Fuel Handling Accident

The fuel handling accident analysis considers the drop of a fuel handling platform or a fuel assembly onto stored spent fuel bundles. This accident results in the limiting event for a fuel handling accident for two reasons. First, the TRM prohibits the movement of fuel assemblies by any means other than the main hoist of the refueling platform or fuel handling platform. Second, this analysis assumes the worst credible failure of the main hoist. Also, TRM limits the load which can pass over the spent fuel pool so that the weight does not exceed that of one channeled fuel assembly and its associated handling tool (approximately 1140 pounds).

The handling of non-fuel items weighing less than 1140 pounds is addressed under the "light loads" issue. These objects may be lifted over irradiated fuel in the spent fuel racks without secondary containment. In the event these objects are dropped, radiological releases may occur which, without secondary containment, are considered to travel directly to the environment without any iodine removal from SGTS. This issue is currently addressed through administrative controls that provide guidance regarding the maximum allowable impact energies that will result in offsite and control room doses within the limits in NUREG-0800, Standard Review Plan 15.0.1 for a fuel handling accident.

15.7.4.1 Identification of Causes and Frequency Classification

15.7.4.1.1 Identification of Causes

The fuel handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism resulting in the dropping of a raised fuel assembly and the mast onto stored irradiated fuel bundles.

15.7.4.1.2 Frequency Classification

This event has been categorized as a limiting fault.

15.7.4.2 Sequence of Events and Systems Operation

Fuel assemblies can be handled in both the auxiliary building and the upper containment pools. The three types of fuel assemblies that could be involved in an accident are (i) recently irradiated fuel, (ii) non-recently irradiated fuel, and (iii) fresh fuel. Recently irradiated fuel is defined as having a decay time less than that specified in the Technical Specification Bases while non-recently irradiated fuel has decayed for a longer period and consequently has a smaller source term inventory. Fresh fuel has no source term inventory.

15.7.4.2.1 Sequence of Events

The sequence of events which is assumed to occur is as follows:

Events

a. A fuel assembly is being handled by the fuel handling platform over the spent fuel pool or by the refueling platform over the containment racks or reactor core. When the hoist is at its fully-retracted position, the assembly and the mast drop, striking seated irradiated fuel assemblies.

b. All rods in the dropped assembly and a number of rods in the struck assemblies fail, releasing radioactive gases to the pool water.

c. Radioactive gases pass from the water to the air above the drop area.

15.7.4.2.2 System Operation

For accidents involving the drop of a recently irradiated fuel assembly, Technical Specifications ensure the availability of secondary containment and the Standby Gas Treatment System (SGTS). The charcoal adsorbers in this system are credited to remove 99% of the elemental and organic iodine before release to the environment. An unfiltered secondary containment bypass leakage rate of 1 cfm is conservatively assumed from unidentified sources.

For the drop of a recently irradiated fuel assembly in the auxiliary building, a high radiation signal from the monitors in the available ventilation systems is assumed to isolate the normal ventilation system and initiate SGTS. These ventilation systems include the fuel handling area ventilation and fuel pool sweep systems. This leakage path is illustrated in Figure 15.7-1.

For the drop of a recently irradiated fuel assembly in the containment pools, a portion of the released activity is conservatively assumed to be released to the environment through the containment ventilation system. This system is automatically isolated upon high-radiation signals from radiation monitors in the ventilation ductwork and the remainder of the activity is drawn into the auxiliary building and released through SGTS. This leakage path is illustrated in Figure 15.7-2.

For the drop of a fresh or non-recently irradiated fuel assembly, no credit is taken for SGTS operation. For secondary containment, only the gross integrity of the boundary is credited, since open doors and penetrations will be closed quickly.

15.7.4.2.3 The Effects of Single Failures and Operator Errors

The automatic ventilation isolation system, which includes: a) the radiation monitoring detectors, b) isolation valves, and c) the SGTS is designed to single failure criteria and safety requirements.

Refer to Sections 7.6 and 9.4 and to Appendix 15A for further details.

15.7.4.3 Fuel Damage

15.7.4.3.1 Mathematical Model

The analytical methods and associated assumptions used to evaluate the consequences of this accident are considered to provide a conservative assessment of the consequences. This approach is consistent with that described in Reference 2.

The dropped assembly is conservatively modeled to impact the spent fuel racks at a small angle from the vertical, introducing a bending mode of failure for the rods in the dropped assembly. Although the channel is expected to provide some degree of lateral support to the fuel bundle, the dropped assembly is assumed to fail catastrophically, resulting in the failure of all rods in this bundle. Actual bending tests with concentrated point-loads show that each fuel rod absorbs approximately 1 ft-lb prior to cladding failure.

One half of the kinetic energy of the falling fuel bundle assembly is assumed to be dissipated to the struck fuel assemblies. Although the effect of buoyancy is considered, the drag force is conservatively neglected. This impact energy is assumed to be absorbed only in the non-fuel components of the struck bundles. Because a fuel assembly consists of approximately 75 percent fuel by weight, the assumption that no energy is absorbed by the fuel material results in considerable analytical conservatism. Since the struck fuel bundles are assumed to be channeled, the handling of irradiated fuel assemblies over unchanneled fuel is prohibited.

15.7.4.3.2 Input Parameters and Initial Conditions

The assumptions used in the analysis of this accident are listed below:

a. The fuel assembly is dropped from the maximum height allowed by the fuel handling equipment.

b. The calculation of impact energy considers the effects of buoyancy and neglects losses due to drag.

c. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

d. Each failed rod in the struck bundles absorbs only enough energy to cause failure. This approach maximizes the number of failed rods in the struck bundles.

e. The dropped and struck fuel types are selected to produce the highest radiological consequences.

15.7.4.3.3 Results

15.7.4.3.3.1 Energy Available

The fuel assembly and the fuel mast are dropped from the fully-retracted position. Half of the impact energy is assumed to be dissipated in the struck bundles.

15.7.4.3.3.2 Fuel Rod Failures

All the fuel rods in the dropped assembly are assumed to fail on impact due to the imposed bending moment resulting in 86 effective fuel rod failures based on the GNF2 design and 88 based on the GNF3 design. Some of the fuel rods in the struck assemblies will fail in compression. For a drop over the racks, the maximum number of fuel rod failures in the struck assemblies was determined to be 35 while a drop over the core would result in 92 rod failures in the struck assemblies due to the larger drop height. These rod failures are based on the GNF2 design. For the GNF3 design, the number of fuel rod failures in the struck assembly for a drop over the racks and a drop over the core was determined to be 34 and 87 rods, respectively. The dose results are presented for the GNF2 design because it bounds the consequences for a drop of a GNF3 assembly.

15.7.4.4 Barrier Performance

For the drop of a recently irradiated fuel assembly in the containment pools, a small portion of the released activity may be initially released to the environment through the containment ventilation system. During refueling operations inside the containment, 6000 cfm of air is exhausted to the environment through the non-safety grade containment cooling system charcoal filter trains, and 6000 cfm of makeup air is provided by the drywell/containment purge fans. The discharge stream is routed such that the airborne radioactivity is monitored by the containment and drywell ventilation exhaust radiation monitoring system.

The parameters applicable to the above exhaust pathway are reported in Table 15.7-9. No credit is taken for iodine removal in the containment cooling system charcoal filter trains or the containment exhaust charcoal filter train.

Discharge of the stream to the environment is through a 20 inch containment penetration and a 20 inch auxiliary building (secondary containment) penetration. Each penetration is equipped with redundant, seismic Category I, ASME Code, Section III isolation valves. The valves for the containment penetration close automatically upon receiving a high-radiation signal from the above-referenced radiation monitoring equipment. Arrangement of the above equipment is shown schematically in Figures 9.4-11 and 9.4-12. Details of the radiation monitoring system are discussed in subsections 7.6.1.4, 12.3.4, and Section 11.5.

Technical Specifications and the TRM define operability, set points, and surveillance frequencies for the containment and drywell ventilation exhaust radiation monitoring system. Closure times for automatic containment isolation valves are identified in the TRM.

The high radiation signal which isolates the containment ventilation system does not isolate the auxiliary building ventilation system or initiate the SGTS. It does, however, provide computer alarm to the control room. The auxiliary building could then be isolated and the SGTS started through operator action. For a conservative analysis, this action will not be considered.

After the containment ventilation system is isolated, there is a net movement of air through the equipment hatch units into the auxiliary building. Assuming that the activity passes immediately into the auxiliary building (neglecting decay or holdup) maximizes the doses. The activity is pulled into the fuel handling area ventilation system, a high radiation signal isolates normal ventilation, and the SGTS starts. Due to the partial unfiltered release, this containment scenario bounds that case where all of the activity is drawn into the auxiliary building.

15.7.4.5 Radiological Consequences

The fission product inventory in the fuel rods is based on bounding bundle powers and exposures. A one-day decay period is assumed for drops of recently irradiated fuel because the TRM precludes fuel handling within 24 hours following initiation of reactor shutdown. For drops of non-recently irradiated fuel, the minimum decay time in the Technical Specification Bases is applied. The fuel rod source terms applied in this analysis are listed in Table 15.7-11.

The design basis analysis is based on Appendix B to NRC Regulatory Guide 1.183. The dose models used are described in Federal Guidance Reports 11 and 12. Specific values of parameters used in the evaluation are presented in Table 15.7-8.

15.7.4.5.1 Fission Product Transport to the Environment

For cases involving the drop of recently irradiated fuel, secondary containment and the automatic initiation of the SGTS are credited. The ductwork of the fuel storage pool sweep and fuel handling area ventilation systems are designed such that there is no release of unfiltered air to the environment following a fuel handling accident as described in subsection 9.4.2.2. A high radiation signal at the exhaust radiation monitor will automatically close the isolation valves:

1.	Response time of the monitoring system	≤3 sec
2.	Valve closing time	4 sec
3.	Air travel time from the detector to the isolation valve	≥7 sec

The SGTS is automatically started in response to the high radiation signal. At this time, the containment ventilation system isolates. Therefore, the only point of release to the environment is the exhaust of the SGTS. An unfiltered secondary containment bypass leakage rate of 1 cfm is conservatively assumed from unidentified sources. Technical Specifications/TRM defined operability, set points, and surveillance frequencies for the SGTS.

As per Appendix B to Regulatory Guide 1.183, it is assumed that the airborne activity in the fuel storage area is released to the environment over a 2-hour period.

No credit is taken for control room isolation or operation of the Control Room Fresh Air Supply system. Since control room inleakage controls are not required in Mode 5, an infinite amount of inleakage is assumed.

15.7.4.5.2 Results

A variety of different fuel handling accidents can be postulated under the requirements in Technical Specifications and the TRM. However, the worst-case scenario is the drop of a recently irradiated assembly over the vessel without secondary containment. The calculated exposures for this design basis analysis are presented in Table 15.7-10 and are within the guidelines of Standard Review Plan 15.0.1. The LPZ dose is not reported since it is bounded by the site boundary dose.

15.7.5 Spent Fuel Cask Drop Accidents

15.7.5.1 Cask Drop Into Spent Fuel Pool

The spent fuel cask crane is prohibited from traveling over the spent fuel pool or any unprotected safety-related equipment as discussed in subsection 9.1.2.3.3. Thus, an accident resulting from dropping a spent fuel cask or other major load into the spent fuel pool is not credible.

15.7.5.2 Cask Drop to Flat Surface

There is a potential for the drop of a spent fuel cask onto the railroad bed; however, the spent fuel cask crane has been designed to be single-failure proof. Therefore, the radiological consequences of this accident have not been evaluated. Further information on the crane is given in subsection 9.1.4.2.2.3.

15.7.6 References

1. Stancavage, P. P. and Morgan, E. J., “Conservative Radiological Accident Evaluation - The CO/NACO 1 Code,” NEDO-21143, March 1976.

2. GESTAR-II, NEDE-24011-P-A-US, General Electric Standard Application for Reactor Fuel.

3. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July, 2000.

4. 10 CFR 50.67, “Accident Source Term”

5. 10 CFR 100

6. 10 CFR 50 GDC 19

TABLE 15.7-1: SEQUENCE OF EVENTS FOR OFFGAS TREATMENT SYSTEM FAILURE

Approximate Elapsed Time	Events
0 sec	Event begins - system fails
0 sec	Noble gases are released
<1 min	Area radiation alarms alert plant personnel
<1 min	Operator actions begin with a. Initiation of appropriate system isolations b. Manual scram actuation c. Assurance of reactor shutdown cooling

TABLE 15.7-2: OFFGAS SYSTEM FAILURE - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

I.	Data and assumptions used to estimate radioactive source from postulated accidents	
	A. Power level, MWt	4496
	B. Release of activity by nuclide	Table 15.7-3
	C. Duration of release, hours	2
II.	Two-hour dispersion at the site boundary (696 m) χ/Q, sec./m^3	6.50-04
III.	Two-hour dispersion at the low population zone (3219 m) χ/Q, sec./m^3	1.45E-04
IV.	Two-hour dispersion at the control room χ/Q, sec./m^3	7.00E-04

TABLE 15.7-3: OFFGAS SYSTEM FAILURE SYSTEM RUPTURE FISSION PRODUCT RELEASE TO ENVIRONMENT

Isotope	SJAE Release Rate 0-1 Hour (Ci/s)	Holdup Pipe and Charcoal Vessel Combined Release Rate 0-2 hrs 0-2 hrs (Ci/s)
Kr-83m	1.41E-02	1.77E-02
Kr-85m	2.40E-02	7.51E-02
Kr-85	9.58E-05	7.52E-03
Kr-87	7.91E-02	6.67E-02
Kr-88	7.91E-02	1.55E-01
Kr-89	5.03E-01	1.90E-02
Xe-131m	7.91E-05	1.63E-02
Xe-133m	1.17E-03	4.43E-02
Xe-133	3.35E-02	3.04E+00
Xe-135m	1.05E-01	1.84E-02
Xe-135	9.10E-02	5.86E-01
Xe-137	6.23E-01	2.86E-02
Xe-138	3.59E-01	5.83E-02
Rb-88	*	4.96E-02
Rb-89	*	1.06E-03
Ba-137M	*	1.61E-04
Cs-138	*	1.06E-02
Other Isotopes	*	*

TABLE 15.7-4: OFFGAS SYSTEM FAILURE SYSTEM RUPTURE

OFFSITE AND CONTROL ROOM DOSES

	Whole Body Dose (rem)	Thyroid Dose (rem)
Site Boundary (696m)	<1.93	Negligible
Low Population Zone	<0.442	Negligible
Control Room Personnel	<0.143	Negligible

TABLE 15.7-5: RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS*

1. Two-hour dispersion data, χ/Q, sec/m^3

Site boundary (696 m)	1.08-03

2. No credit is taken from the holdup in the radwaste building.

3. The tank activity is based on an I-131 release rate of 700 μCi/sec from the fuel

*The design information used in the liquid waste system leak or failure radiological analysis is historical design information (see Section 15.7.2).

TABLE 15.7-6: RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE ACTIVITY RELEASE TO THE ENVIRONMENT (CURIES)*

Isotopes	Activities Released (Ci)
I-131	1.82
I-132	2.18
I-133	1.30
I-134	1.72
I-135	6.35

*The design information used in the liquid waste system leak or failure radiological analysis is historical design information (see Section 15.7.2).

TABLE 15.7-7: RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE RADIOLOGICAL EFFECTS

Location	Inhalation Thyroid Dose(rem)	Whole Body Dose(rem)
Site boundary (696 m)	< 1.25	negligible

TABLE 15.7-8: FUEL HANDLING ACCIDENT PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS

I. Data and assumptions used to estimate radioactive source from postulated accidents

A.	Bundle Power Level	10.6 MWt
B.	Fuel damaged	Section 15.7.4.3
C.	Plenum activity	8% I-131 10% Kr-85 12% alkali metals 10% other isotopes
D.	Plenum iodine fractions	
	1. Organic	0.15%
	2. Elemental	4.85%
	3. Particulate	95%
E.	Overall Pool Iodine Decontamination Factor	200
	1. Organic Iodine	1
	2. Elemental Iodine	285
	3. Particulates	Infinite

II. Data and assumptions used to estimate activity released

A.	Standby Gas Treatment System filtration efficiency	
	1. Organic iodine	99%
	2. Elemental iodine	99%
B.	Secondary containment bypass	1 cfm

III. Data and assumptions used to calculate control room dose

A.	Control Room Fresh Air System	Not credited
B.	Control Room Inleakage	Infinite

IV. Two-Hour dispersion data, X/Q,

TABLE 15.7-8: FUEL HANDLING ACCIDENT PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS (Continued)

A.	Site boundary	6.50E-04 s/m^3
B.	Control Room	9.0E-04 s/m^3

TABLE 15.7-9: FUEL HANDLING ACCIDENT CONTAINMENT EXHAUST PARAMETERS

A.	Response time of monitoring system	$\leq$ 10 sec.
B.	Travel time of air from monitor to the isolation valve	0.7 sec.
C.	Valve closure time	110 sec.
D.	Exhaust flowrate	6000 cfm
E.	Containment free volume	1.4×10^6 ft^3
F.	Mixing in containment	50%

TABLE 15.7-10: FUEL HANDLING ACCIDENT RADIOLOGICAL EFFECTS

	Total Effective Dose Equivalent (rem TEDE)
Site boundary (696 m)	$\leq$ 3.12
Control Room	$\leq$ 3.14

TABLE 15.7-11: FUEL HANDLING ACCIDENT PARAMETERS
FUEL ROD ACTIVITY (Curies)

ISOTOPE	1 DAY OF DECAY
BR82	2.54E+01
BR83	5.65E-01
I130	7.88E+01
I131	3.25E+03
I132	4.07E+03
I133	3.21E+03
I135	5.25E+02
KR83M	2.18E+00
KR85	3.82E+01
KR85M	3.18E+01
KR88	1.05E+01
XE129M	1.37E-01
XE131M	3.95E+01
XE133	6.35E+03
XE133M	1.85E+02
XE135	1.69E+03
XE135M	8.40E+01

TABLE 15.7-12: DELETED

TABLE 15.7-13: DELETED

TABLE 15.7-14: DELETED

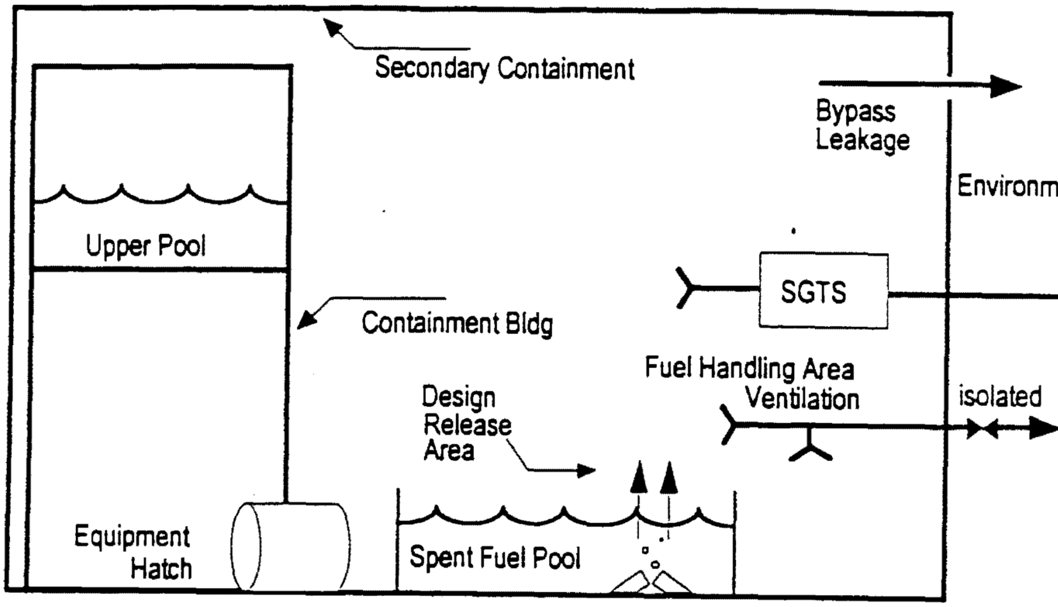


Figure 15.7-1 Fuel Handling Accident Outside Containment Leakage Path

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FUEL HANDLING ACCIDENT OUTSIDE CONTAINMENT LEAKAGE PATH FIGURE 15.7-1

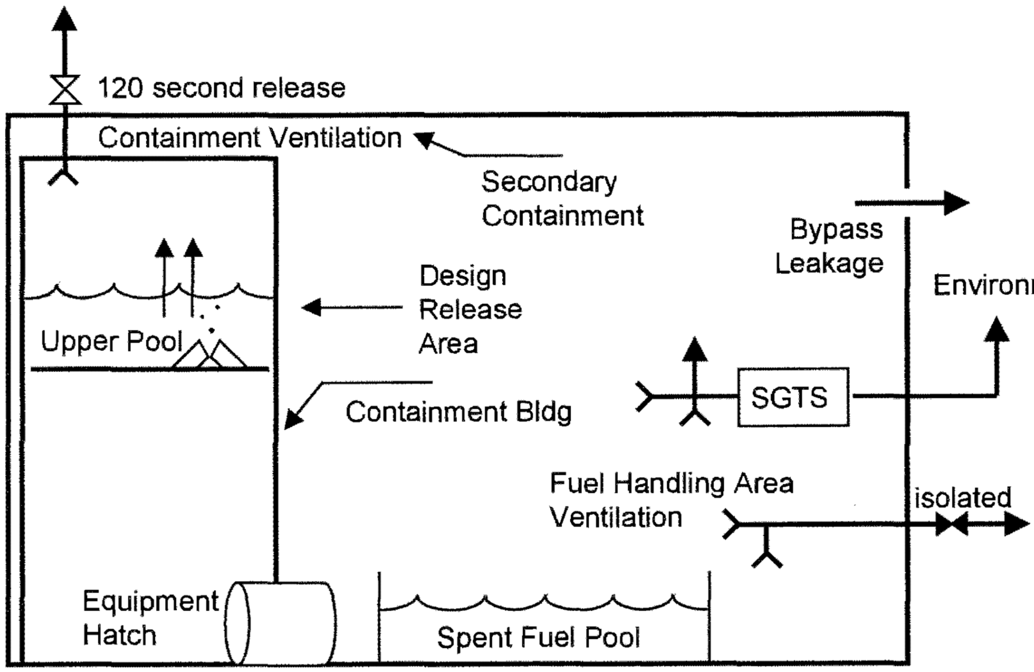


Figure 15.7-2: Possible Leak Paths for Fuel Handling Accident Inside Containment

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	POSSIBLE LEAK PATHS FOR FUEL HANDLING ACCIDENT INSIDE CONTAINMENT FIGURE 15.7-2

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

15.8.1 Capabilities of Present BWR 4/5/6 Design to Accommodate ATWS

[HISTORICAL INFORMATION] [A study was performed at the request of the Nuclear Regulatory Commission to evaluate the consequences of an undefined failure of the scram protection system and report on features which could mitigate the effects of a reactor shutdown from an anticipated transient without control rod drive scram. The GE study identified potential design innovations which could be applied to any BWR to ease the severity of the consequences of failure-to-scram special event. A proposed method for minimizing the effects of failure to scram is described in References 1 and 2.

Several General Electric Topical Reports (Refs. 3, 4 and 5) addressed a regulatory staff proposed position on plants listed as Class I.B (including Grand Gulf) published in WASH-1270, "Technical Reports on Anticipated Transients Without Scram for Water-Cooled Power Reactors." In June and September 1976, GE submitted three studies for response to NRC status reports about ATWS for all BWR 4, 5, and 6 plants (including this application). These GE reports and studies discuss design modifications which would reduce the probability and/or mitigate the consequences of ATWS.

In June of 1984 the NRC issued 10CFR50.62 which required specific changes to reduce the likelihood of failure to shutdown the reactor following anticipated transients and to mitigate the consequences of an ATWS event. Grand Gulf has implemented an alternate rod insertion system, a standby liquid control system equivalent in control capacity to 86 GPM of 13 weight percent sodium pentaborate solution, and additional reactor recirculation pump trips. These changes were implemented as described in Reference 7, which provides NRC concurrence that these changes are in compliance with 10CFR50.62. Grand Gulf was thus shown to be in complete compliance with 10CFR50.62 for a GE fueled core.

The following transients are bounded by assumptions in the referenced GE licensing topical reports and the Appendix 15A coverage.

a. Inadvertent Control Rod Withdrawal

b. Loss of Feedwater

c. Loss of AC Power

d. Loss of Electrical Load

e. Loss of Condenser Vacuum

f. Turbine Trip

g. Closure of Main Steam Line Isolation Valves

The limiting transient was shown to be the MSIV closure event with failure to scram. This transient was analyzed by GE for maximum extended operating domain conditions based on a GE-fueled core. The significant fuel-related factor for this transient is the core average response.

The AREVA NP reload fuel is designed to be compatible with the original and Co-resident GE fuel. The response characteristics of the reload and the GE fuel types are similar and the core-wide behavior of the plant resulting from the limiting ATWS event is relatively unaffected for reload cores (Ref. 6). GE has validated the acceptability of the calculated ATWS response for cores comprised of GE11 and 9x9-5 fuel (Ref. 8). For introduction of Atrium-10 fuel, an ATWS evaluation of peak vessel pressure was made to show that applicable pressure limits are met (Ref. 9). For introduction of GE14, GNF2, and GNF3 fuel designs, evaluations of the limiting ATWS events were performed to show that the applicable acceptance criteria for reactor vessel and containment pressure, suppression pool temperature, and peak cladding temperature are met (Ref. 10, 11 and 15).]

The ATWS events were re-analyzed for the Extended Power Uprate and 24 month fuel cycle (Ref. 12) consistent with the guidelines for extended power uprate for four events: Main Steam Isolation Valve Closure (MSIVC) event, Pressure Regulator Failure - Open (PRFO) event, Loss of Offsite Power (LOOP), and Inadvertent Opening of One Relief Valve (IORV) event. The two limiting events were the MSIVC and PRFO events and the analyses showed that the applicable reactor, core, and containment acceptance criteria continue to be met for operation at the EPU power.

The limiting ATWS events were re-evaluated as part of the implementation of MELLLA + (Ref. 14). The evaluation determined that MELLLA+ results were within all ATWS acceptance criteria: peak vessel pressure (1500 psig), peak cladding temperature (2200°F), peak local cladding oxidation (17%), peak suppression pool temperature (210°F) and peak containment pressure (15.0 psig). The significant change in analyzed parameters was the SLCS initiation delay from the EPU analyzed 120 seconds (Ref. 12) to the MELLLA+ 300 seconds delay. This is based on the analysis criteria that SLCS is initiated at the later of either (1) the time of the high pressure ATWS RPT plus 5 minutes (300 second operator action) or (2) the time at which the suppression pool temperature reaches the boron injection initiation temperature (BIIT) of 110°F plus one minute. The ATWS RPT (300 seconds) satisfies the criteria by being later than the BIIT plus one minute (97 seconds). Borated water enters the vessel 125 seconds later due to transport delay.

This re-evaluations for MELLLA+ and GNF3 also included the evaluation of two other events in order to disposition the effect of core instabilities with a postulated ATWS. These two events were the Turbine Trip With Bypass (TTWBP) event and the Recirculation Pump Trip (RPT) event. The MELLLA+ and GNF3 analyses described in References 14 and 15 credited a set of time critical operator actions for certain postulated events as follows:

1. Initiate reactor level reduction (90 seconds following failure to scram concurrent with no reactor recirculation pumps in service and core thermal power (CTP) greater than 5%). (TTWBP and RPT events with an ATWS)

2. Initiate Standby Liquid Control Injection (300 seconds if CTP greater than 5% or before Suppression Pool Temperature reaches 110 degrees F). (MSIVC, PRFO, TTWBP, and RPT events with an ATWS)

3. Initiate Residual Heat Removal Suppression Pool Cooling (660 seconds). (MSIVC and PRFO events with an ATWS)

15.8.2 References

1. Hatch Unit 1 FSAR, Amendment 10, Appendix L, "Failure-to-Scram Analysis," October 27, 1971.

2. Michelotti, L. A., "Analysis of Anticipated Transients Without Scram," NEDO-10349, March 1971.

3. Claassen, L. B., Eckert, E. C., "Studies of BWR Designs for Mitigation of Anticipated Transients Without Scrams," NEDO-20626, October 1974.

4. Claassen, L. B., Eckert, E. C., "Studies of BWR Designs for Mitigation of Anticipated Transients Without Scrams, Amendment 1," NEDO-20626-1, June 1975.

5. Baysinger, L. W., Eckert, E. C., and Weis, D. G., "Studies of BWR Designs for Mitigation of Anticipated Transients Without Scrams, Amendment 2," NEDO-20626-2, July 1975.

6. MPEX-90/044, "Justification for Cycle 5 MEOD Operation Under ATWS Conditions," Letter N.L. Gamer, ANF, to J. B. Lee, SERI.

7. Claassen, L. B., Earle, R. T., et al, "Anticipated Transients Without Scram - Response to NRC ATWS Rule, 10CFR50.62," NEDE-31096-P-A, February 1987.

8. GEXI-96/00479, "Check for ATWS Compliance-GGNS Cycle 9," R.E. Kingston, November 14, 1996.

9. ANP-2580(P), Revision 0, "Grand Gulf Nuclear Station Cycle 16 Plant Transient Analysis," AREVA NP, December 2006.

10. ECH-NE-08-00031, Rev. 2, "GE14 Fuel Design Cycle-Independent Analyses for Entergy Grand Gulf Nuclear Station," GNH 0000-0072-6563-R2, March, 2011.

11. ECH-NE-10-00021, Rev. 5, "GNF2 Fuel Design Cycle-Independent Analyses for Grand Gulf Nuclear Station", Mar. 2020.

12. GGNS-NE-10-00004, Rev. 4, "GGNS EPU - Anticipated Transients Without Scram."

13. GNRI-2015/00114, Grand Gulf Nuclear Station, Unit 1 - Issuance of Amendment regarding Maximum Extended Load Line Limit Analysis Plus. (TAC No. MF2798), Amendment No. 205, dated August 31, 2015).

14. GGNS-NE-12-00025, GE Hitachi Nuclear Energy, Project Task Report, Entergy Operations, INC., Grand Gulf Nuclear Station MELLLA+, Task T0902: Anticipated Transient Without Scram, dated August 2012.

15. ECH-NE-20-00006, Rev. 0, "GGNS Fuel Design Cycle-Independent Analyses for Grand Gulf Nuclear Station", Mar. 2020

APPENDIX 15A PLANT NUCLEAR SAFETY OPERATIONAL ANALYSIS (NSOA) - (A SYSTEM-LEVEL/QUALITATIVE TYPE PLANT FMEA)

15A.1 OBJECTIVES

15A.1.1 General Objectives

The main general objectives of the Nuclear Safety Operational Analysis (NSOA) are cited below along with the mission of each objective.

a. Essential Protective Sequences ...to identify and demonstrate that all the essential protection sequences needed to accommodate the plant normal operations, anticipated and abnormal operational transients, and design basis accidents are available and adequate

b. Design Basis Adequacy ...to identify and demonstrate that all the safety design basis of the various structures, systems or components, needed to satisfy the plant essential protection sequences are appropriate, available and adequate

c. System-Level/Qualitative Type FMEA ...to provide a system level/qualitative-type failure modes and effects analysis (FMEA) of essential protective sequences to show compliance with the single active component failure (SACF) or single operator error (SOE) criteria

d. NSOA Criteria Relative to Plant Safety Analysis ...to identify the systems, equipment, or components' operational conditions or requirements essential to satisfy the nuclear safety operational criteria utilized in the Chapter 15 plant events

e. Technical Specification Operational Basis ...to establish limiting operating conditions, testing, and surveillance bases relative to plant technical specification operational requirements

15A.1.2 Specific Objectives

The specific objectives of the plant-wide Nuclear Safety Operational Analysis (NSOA) are cited below:

a. Essential Protective Sequences

Each event considered in the plant safety analysis (Chapter 15) is further examined and analyzed.

Essential protective sequences are identified. The appropriateness of each sequence is discussed for all operating modes. Each protective sequence path is evaluated for SACF.

b. Design Basis Adequacy

Each event protective sequence involves specific structures, systems or components performing safety or power generation functions. There are also interrelationships between primary systems and secondary or auxiliary equipment in providing these functions. The individual design bases (identified throughout the SAR for each structure, system, or component) are brought together in this section. The entire plant safety analysis is evaluated here. The necessary equipment working together in satisfying plantwide design bases by performing its individual system design bases are illustrated.

c. System-Level/Qualitative Type FMEA

A plant-wide system level qualitative-type FMEA is presented here. Each event protective sequence entry is evaluated relative to SACF or SOE criteria. Safety classification aspects and interrelationships between system are also considered. The system-level SACF or SOE is certainly a valid conservative "Worst-case" envelope evaluation. Discounting any less severe evaluations than SACF or SOE such as by quantitative analysis is not claimed in this section although certainly it would assure less limiting results than shown.

d. NSOA Criteria Relative to Plant Safety Analysis

The plant safety analysis performed in Chapter 15 is further examined relative to the systematic classification of plant events by frequency of occurrence, radiological impact, unacceptable results, and allowable limits of the safety criteria for the various event classifications: normal (planned) operation, anticipated (expected) and abnormal (unexpected) operational transients, and design basis accidents are described and established.

e. Technical Specifications Operational Basis

Evaluations presented in this section provide the basis for justifications of more realistic and engineered technical specifications including system or equipment surveillance requirements, allowable down times, etc.

15A.2 APPROACH TO OPERATIONAL NUCLEAR SAFETY

15A.2.1 General Philosophy

Nuclear safety means different things to different people. To derive a consistent set of nuclear safety requirements for the operation of a nuclear power plant, nuclear safety must first be defined; otherwise, almost any proposed operational restriction or plant system could be cast in a "nuclear safety" light.

Nuclear safety requirements that impose restrictions on the power output of a nuclear utility plant necessarily represent a judgment (conscious or unconscious) between the benefits and potential hazards of nuclear generated power. In the limit, theoretically the safest nuclear power plant would be one not even allowed to approach criticality, but no public benefit would be derived from the plant. On the other hand, a nuclear plant allowed to operate with none of its safety systems operable would theoretically represent an undue risk to the public. It is one of the objectives of this NSOA to derive nuclear safety operational requirements and analyses for the plant that are based on specified measures of real nuclear safety aspects. These measures involve both broad and deep specific safety considerations. These also represent conscious, reasoned judgments of the relationship between public risk and benefit.

The specified measures of safety used in this analysis are referred to as "unacceptable results oriented." They are analytically determinable limits on the consequences of different classifications of plant events. The nuclear safety operational analysis is thus an "event-consequence" oriented evaluation.

15A.2.2 Specific Philosophy

In this appendix the following specific philosophical observations are utilized to develop the NSOA.

a. Scope and Classification Of Plant Events

The scope and classification of the situations analyzed will include:

1. Normal (Planned) Operations

2. Anticipated (Expected) Operational Transients

3. Abnormal (Unexpected) Operational Transients

4. Design Basis (Postulated) Accidents

5. Special (Hypothetical) Events

Refer to Tables 15A.2-1 through 15A.2-5 for specific event/classifications.

The events referenced and classified above represent all the plant situations considered applicable to safety evaluation.

b. Safety and Power Generation Aspects

It is very important to recognize the difference between safety and power generation aspects. Safety considerations directly involve the health and safety of the offsite general public. Matters identified with "safety" classification are governed by very precise regulatory requirements. Safety functions include:

1. The accommodation of abnormal operational transients and postulated design basis accidents

2. The maintenance of containment integrity, when necessary

3. The assurance of ECCS, when necessary

4. The continuance of RCPB integrity, when necessary

Safety is related to offsite dose limits, infrequent and low probability occurrences, SACF criteria, worst-case operating conditions and initial assumptions, automatic (10 minute) corrective action, significant unacceptable dose and environmental effects, and the involvement of other coincident (mechanistic or non-mechanistic) plant and environmental situations.

Power generation considerations are directly related to continued plant power generation operation, equipment operational matters, component availability aspects, and indirectly related to long-term offsite general public effects.

Matters identified with "power generation" classification are also covered by regulatory guidelines. Power generation functions include:

1. The accommodation of planned operations and anticipated operational transients

2. The minimization of radiological releases to appropriate levels

3. The assurance of safe and orderly reactor shutdown, when necessary, and/or return to power generation operation

4. The continuance of plant equipment design conditions to ensure long term reliable operation

Power generation is related to 10 CFR 20 and 10 CFR 50 Appendix I dose limits, moderate and high probability occurrences, nominal operating conditions and initial assumptions, allowable immediate operator manual actions, and insignificant unacceptable dose and environmental effects.

c. Frequency of Events

Consideration of the frequency of the initial (or initiating) event is reasonably straight-forward. Added considerations of further component failures or operator errors certainly complicates the classification grouping and thus the related limits or acceptable consequences. The events in this appendix are initially grouped per initiating frequency occurrence. The imposition of further failures will necessitate at a later date further reclassification. This classification will certainly result in the event being listed in a less restrictive category.

The time intervals considered in the Chapter 15 transient analyses only cover short term events which include the most severe point from a thermal and pressure margin concern. Long term events such as Reactor Vessel Isolation, Initial Core Cooling, etc., are not included in the simulation. Should these events occur, they will follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred, and are expected to be less severe than those already experienced by the system. The main purpose of the sequence of event table is to show the proper sequence assumed in the analysis and does not include the long term response.

The introduction of SACF or SCF or SOE in planned operation/anticipated and abnormal operational transient evaluations has not been previously considered a design basis or evaluation prerequisite. It is entertained here for plant capability demonstration purposes.

d. Conservative Analysis - Margins

The unacceptable results established in this appendix relative to the public health and safety aspects are in themselves in conformance to regulatory requirements per se. They are also in conformance with regulations by large margins even though the events, their assumptions, conditions of evaluation, coincident situations, the limits, etc., are equally conservative in themselves by large margins. Further introduction of large margin operational requirements is not reasonable or justifiable. The results of this NSOA directly lead to envelope technical specifications.

The utilization of margins allowance to introduce further limiting restrictions is unwise, unreasonable, and countersafety oriented.

Restrictive operations on hypothetical limits established by further operational limits (e.g. set point margins) lead to disrespect for true safety aspects.

e. Safety Function Definition

Consideration of the frequency of the need for a safety function should be very carefully weighed and examined in order to truly assess real design basis, operational, and availability requirements.

First of all, the essential protective sequences shown for an event in this Appendix are the minimum required to be available to satisfy the SACF or SOE evaluation aspects of the event and yet meet all safety functional objectives. Many more protective "success paths" exist with the event than are shown.

Secondly, not all the events involve equal natural, environmental or plant conditional assumptions (e.g. LOCA and SSE are associated with Event 39. In Event 36, CRDA is not assumed to be associated with any SSE or OBE occurrence). Therefore, seismic safety function requirements are inappropriate for Event 36, although most safety function equipment associated with the protective sequence are capable of more limiting events, Event 39. The probability of Event 36 is far less than Event 39 occurrence-wise and certainly evaluation-assumption-wise.

Third, containment may be a safety function for some event (when uncontained radiological effects would be unacceptable) but for others, it certainly may not be applicable (e.g. during refueling). The requirement to maintain the containment in post-accident recovery is only appropriate, when needed, to limit doses to less than 10 CFR 50.67. After radiological sources are depleted with time, further containment is unnecessary. Thus the time domain and need for a function is taken into account and considered when evaluating the events in this appendix.

Fourth, the use of low frequency, high priority ESF equipment, limiting unacceptable result events for high probability, minor unacceptable result events should not be misunderstood to require similar pedigree equipment requirements on other supplement motorsafety components.

The interpretation of the use of ESF-SACF capable systems for anticipated operational transient protective sequences should not lead to the assumption that these equipment requirements (seismic, redundancy, diversity, testable, IEEE, etc.) are appropriately required for this event or associated with the event.

Although certain events assume the operation of certain nonsafety-grade systems to provide a realistic transient signature, failures of these systems would not make these events more thermally or pressure limiting than the limiting events already presented in the FSAR. In fact, many of the events shown which have a Level 9 turbine trip (a nonsafety-grade trip) would be less severe if the Level 9 trip was assumed not to function. Also, the loss of feedwater event would be no more severe without the recirculation runback-again, this system is simulated to demonstrate expected plant response. In summary, the thermal and pressure limits are not compromised by the simulation of nonsafety-grade systems.

The impacts of ΔCPR and peak vessel pressure without taking credit for these non-safety grade systems and components is discussed as follows:

[HISTORICAL INFORMATION] [As a concern of the credibility for level 9 turbine trip and turbine bypass system, GE and the NRC met on November 20 and 21, 1978 for a comprehensive review of all such transients and, as a result of that meeting, determined that the most limiting event which takes credit for level 9 turbine trip and turbine bypass system is the feedwater controller failure.] Results indicate a ΔCPR increase of approximately 0.02 and 0.08 for this transient without a level 9 turbine trip and a functioning turbine bypass system, respectively. The BWR/6, however, has a safety grade level 8 scram which precludes the use of the level 9 turbine trip for scram. The impact of ΔCPR with no credit for recirculation runback is negligible because the NSSS already reached its lowest MCPR before recirculation runback occurs.

For the peak vessel pressure concern, sensitivity studies show that the peak vessel pressures for the transients in Table 15A.2-11 without credit for non-safety grade systems and components is bounded by that of the worst overpressure protection case, i.e., MSIV closure with flux scram.

f. Envelope and Actual Event Analysis

The event analyses presented in Chapter 15 when examined from the frequency standpoint would lead to the conclusion that each year a spectrum of the events are occurring as defined. Study of the plant occurrences certainly verifies that the protective sequences cited are conservative, and in most cases never needed. Experience, of course, has been confined to planned operation, anticipated operational transients, and a very small number of abnormal operational transients situations. Operator action is very valuable and repeatedly demonstrated yet ignored as a protective sequence. Consideration and credit of this success path certainly should be allowed and recognized for operational transients.

15A.2.3 Consistency of the Analysis

One evaluation objective of this analysis is consistency. Therefore, it is worthwhile to investigate possible inconsistencies in the selection of nuclear safety operational requirements (and technical specifications); then it will be seen in the presented NSOA that such inconsistencies are avoided.

Figure 15A.2-1 illustrates three inconsistencies. Panel A shows the possible inconsistency resulting from operational requirements being placed on separated levels of protection for one event. If the second and sixth levels of protection are important enough to warrant operational requirements, then so are the third, fourth, and fifth levels. Panel B shows the possible inconsistency resulting from operational requirements being arbitrarily placed on some action thought to be important to safety. In the case shown, scram represents different protection levels for two similar events in one category; if the fourth level of protection for Event B is important enough to warrant an operational requirement, then so is the fourth level for Event A. Thus, to simply place operational requirements on all equipment needed for some action (scram, isolation, etc.) could be inconsistent and certainly unreasonable if different protection levels are represented. Panel C shows the possible inconsistency resulting from operational requirements being placed on some arbitrary level of protection for any and all postulated events. Here the inconsistency is not recognizing and accounting for different event categories based on cause or expected frequency of occurrence.

Inconsistencies of the types illustrated in Figure 15A.2-1 are avoided in the NSOA by directing the analysis to "event-consequences" oriented aspects. Analytical inconsistencies are avoided by analytically treating all the events of a category under the same set of functional rules. Thus, it is valid to compare the results of the analyses of the events in any one category and invalid to compare events of different category to the other category respective, limits.

15A.2.4 Comprehensiveness of the Analysis

One evaluation objective of this analysis is to be comprehensive. Therefore, the analysis must be sufficiently comprehensive in method that (1) all plant hardware is considered; and, (2) that the full range of plant operating conditions are considered. The tendency to be preoccupied with "worst cases" (those that appear to give the most severe consequences) is recognized; however, the protection sequences essential to lesser cases may be different (more or less restrictive) from the worst case sequence. To assure that operational and design basis requirements are defined and appropriate for all equipment essential to attaining acceptable consequences, all essential protection sequences must be identified for each of the plant safety events examinations. Only in this way is a comprehensive level of safety attained. Thus, the NSOA is also "protection sequence"-oriented to achieve comprehensiveness.

15A.2.5 Systematic Approach of the Analysis

In summary, the systematic method utilized in this analysis contributes to both (a) the consistency and (b) comprehensiveness of the analysis mentioned above. It also represents a sound and disciplined approach to a very important yet practical engineering problem. The desired characteristics representative of a systematic approach to selecting BWR operational requirements are listed as follows:

a. Specified measures of safety-unacceptable results

b. Consideration of all potential planned operations

c. Systematic event selection

d. Common treatment analysis (FMEA, SACF, SOE) of all events of any one type

e. Systematic identification of plant actions and systems essential to avoiding unacceptable results

f. Emergence of operational requirements and limits from system analysis

Figure 15A.2-2 illustrates the systematic process by which the operational and design basis nuclear safety requirements and technical specifications are derived. The process involves the evaluation of carefully selected plant events relative to the unacceptable results (specified measures of safety). Those limits, actions, systems, and components found to be essential to achieving acceptable consequences are the subjects of operational requirements.

It is important to note that the analysis of each of the transients is based on the single-failure criterion associated with abnormal transients (i.e., abnormal transients are defined as events which occur as a result of equipment malfunctions as a result of a single, active component failure or operator error). Following this single failure, the resulting transient is simulated in a conservative fashion to show the response of primary system variables and how the various plant systems would interact and function. In the above transients, the consideration of any additional failures is not considered appropriate within the realm of anticipated transient definition.

15A.2.6 Relationship of Nuclear Safety Operational Analysis to Safety Analyses of Chapter 15

One of the main objectives of the operational analyses is to identify all essential protection sequences and to establish the detailed equipment conditions essential to satisfying the nuclear safety operational criteria. The spectrum of events examined in Chapter 15 represent a complete set of plant safety considerations. The main objective of the earlier analyses of Chapter 15, is, of course, to provide detailed "worst case" (limiting or envelope) analysis of the plant events. The "worst cases" are correspondingly analyzed and treated likewise in this Appendix.

The detailed discussion relative to each of the events covered in Chapter 15 will not be repeated in this appendix. Please refer back to the specific section in Chapter 15 as cross-correlated in Tables 15A.2-1 thru 15A.2-5.

Tables 15A.2-1 thru 15A.2-5 provide cross-correlation between the NSOA event, its protection sequence diagram, and its specific safety evaluation identification earlier in Chapter 15.

Nonsafety-grade equipment is not used in the FSAR analyses to mitigate accidents. However, when the assumption of a nonsafety-grade equipment's use results in more severe consequences during an accident than its malfunction, the nonsafety-grade equipment is assumed to perform its intended function. Nonsafety-grade systems or components are used to mitigate less severe events as shown in NSOA Figure 15A.2-3. Table 15A.2-11, Nonsafety-Grade Systems/Components Assumed in FSAR Analyses, summarizes the specific nonsafety-grade systems or components assumed in specific FSAR events, and cross-references to specific FSAR sections.

The combination of NSOA figures and Table 15A.2-11 identifies the employment of nonsafety-grade system or components assumed in evaluating FSAR events, either to mitigate less severe transients or as a conservative assumption for accidents.

15A.2.7 Relationship Between NSOA and Operational Requirements, Technical Specifications, Design Basis, and SACF Aspects

By definition, "an operational requirement" is a requirement or restriction (limit) on either the value of a plant variable or the operability condition associated with a plant system. Such requirements must be observed during all modes of plant operation (not just at full power) to assure that the plant is operated safely (to avoid the unacceptable results). There are two kinds of operational requirements for plant hardware:

a. Limiting condition for operation: the required condition for a system while the reactor is operating in a specified state

b. Surveillance requirements: the nature and frequency of tests required to assure that the system is capable of performing its essential functions

Operational requirements are systematically selected for one of two basic reasons:

a. To assure that unacceptable results are avoided or mitigated following specified plant events by examining and challenging the system, component, and equipment design basis

b. To assure the existence of a single failure proof success path to acceptable consequences should a transient or accident occur by confirming SACF or SOE criteria conformance

The operational requirements that emerge from the NSOA are frequently complex hardware requirements applicable only under certain carefully specified plant conditions. Although these complex operational requirements are the true safety requirements, they frequently are too complicated for direct use as a clear technical specification. As shown in Figure 15A.2-2, the complex operational requirements are conservatively simplified as a final step in the process so that a practical set of technical specifications and operating procedures may be obtained.

The individual structures, systems, components which perform a safety function are required to do so under design basis conditions including environmental consideration and under single active component failure assumptions. The NSOA confirms the previous examination of the individual equipment (See "Evaluations" subsection) requirement conformance analyses.

15A.2.8 Unacceptable Results Criteria

Tables 15A.2-6 through 15A.2-10 identify the unacceptable results associated with different event categories. In order to prevent or mitigate them, they are recognized as the major bases for identifying system operational requirements as well as the bases for all other safety analyses vs. criteria throughout the SAR.

15A.2.9 General Nuclear Safety Operational Criteria

The following general nuclear safety operational criteria are used to select operational requirements:

Applicability	Nuclear Safety Operational Criteria
Planned operation anticipated, abnormal operational transients, design basis accidents, and additional separate plant capability events	The plant shall be operated so as to avoid unacceptable results.
Anticipated and abnormal operational transients and design accidents	The plant shall be operated in such a way that no Single Active Component Failure (SACF) can prevent the safety actions essential to avoiding the unacceptable results associated with anticipated or abnormal operational transients or design basis accidents. However, this requirement is not applicable during structure, system or component repair if the availability of the safety action is maintained either by restricting the allowable repair time or by more frequently testing a redundant structure, system, or component.

The specific unacceptable results associated with the different categories of plant operation and events are dictated by:

a. Frequence of occurrence (probability)

b. Allowable limits (per the probability) - related to radiological, structural, environmental, etc., aspects

c. Coincidence of other related or unrelated disturbances

d. Time domain of event and consequences consideration

TABLE 15A.2-1: PLANNED (NORMAL) OPERATION

Cross-Correlation References

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.
1	Refueling - Initial - Reload	15.A 6-3,4,5,6	9.1
2	Achieving Criticality	15.A 6-3,4,5,6	4.6
3	Heat-Up	15.A 6-3,4,5,6	4.4
4	Power Operation - Generation - Steady State - Daily Load and Reduction Recovery - Grid Frequency Control Response - Control Rod Sequence Exchanges - Power Generation Surveillance Testing • Turbine Stop Valve Surveillance Tests • Turbine Control Valve Surveillance Tests • MSIV Surveillance Tests	15.A 6-3,4,5,6	10.2, 8.2, 4.6
5	Achieving Shutdown	15.A 6-3,4,5,6	4.6
6	Cooldown	15.A 6-3,4,5,6	5.4.7

TABLE 15A.2-2: ANTICIPATED (EXPECTED) OPERATIONAL TRANSIENTS

Cross-Correlation References

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.
7	Manual or Inadvertent SCRAM	15A.6-7	7.2
8	Loss of Plant Instrument Service Air Systems	15A.6-8	9.3.1
9	Inadvertent Start-Up of HPCS Pump	15A.6-9	15.5.1
10	Inadvertent Start-Up of Idle Recirculation Loop Pump	15A.6-10	15.4.4
11	Recirculation Loop Flow Control Failure with Increasing Flow	15A.6-11	15.4.5
12	Recirculation Loop Flow Control Failure with Decreasing Flow	15A.6-12	15.3.2
13	Recirculation Loop Pump Trip - With One Pump - With Two Pumps	15A.6-13	15.3.1
14	Inadvertant MSIV Closure - With One Valve - With Four Valves	15A.6-14a 15A.6-14b	15.2.4
15	Inadvertant Operation of One Safety/Relief Valve - Opening/Closing - Stuck Open	15A.6-15	15.6.1

TABLE 15A.2-2: ANTICIPATED (EXPECTED) OPERATIONAL TRANSIENTS (Continued)

Cross-Correlation References

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.
16	Continuous Control Rod Withdrawal Error - During Start-Up - During Refueling	15A.6-16	15.4.1
17	Continuous Control Rod Withdrawal Rod Error at Power	15A.6-17	15.4.2
18	RHRS - Shutdown Cooling Failure Loss of Cooling	15A.6-18	15.2.9
19	RHRS - Shutdown Cooling Failure Increased Cooling	15A.6-19	15.1.6
20	Loss of All Feedwater Flow	15A.6-20	15.2.7
21	Loss of Feedwater Heater	15A.6-21	15.1.1
22	Feedwater Controller Failure Maximum Demand - Low Power	15A.6-22	15.1.2
23	Pressure Control Failure -Open	15A.6-23	15.1-3
24	Pressure Control Failure -Closed	15A.6-24	15.2.1
25	Main Turbine Trip With By-Pass System Operational	15A.6-25	15.2.3
26	Loss of Main Condenser Vacuum	15A.6-26	15.2.5
27	Main Generator Trip (Load Rejection) With By-Pass System Operational	15A.6-27	15.2.2

TABLE 15A.2-2: ANTICIPATED (EXPECTED) OPERATIONAL TRANSIENTS (Continued)

Cross-Correlation References

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.
28	Loss of Plant Normal On-Site AC POWER - Auxiliary Transformer Failure	15A.6-28	15.2.6
29	Loss of Plant Normal Off-Site AC POWER - Grid Connection Failure	15A.6-29	15.2.6

TABLE 15A.2-3: ABNORMAL (UNEXPECTED) OPERATIONAL TRANSIENTS

Cross-Correlation References

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.
30	Main Generator Trip (Load Rejection) with By-Pass System Failure	15A.6-30	15.2.2
31	Main Turbine Trip With By-Pass System Failure	15A.6-31	15.2.3
32	Inadvertant Loading and Operation of a Fuel Assembly In An Improper Position	15A.6-32	15.4.7
33	Recirculation Loop Pump Seizure	15A.6-33	15.3.3
34	Recirculation Loop Pump Shaft Failure	15A.6-34	15.3.4

TABLE 15A.2-4:DESIGN BASIS (POSTULATED) ACCIDENTS

Cross-Correlation References

NSOA EVENT NO.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.
35	Control Rod Drop Accident	15A.6-35	15.4.9
36	Fuel Handling Accident	15A.6-36	15.7.4
37	Loss-of-Coolant Accident Resulting From Spectrum of Postulated Piping Breaks Within the RPCB Inside Containment	15A.6-37	15.6.5
38	Small, Large, Steam and liquid Piping Breaks outside Containment	15A.6-38	15.6.4
39	Instrument Line Break Outside Containment	15A.6-38	15.6.2
40	Feedwater Line Break Outside Containment	15A.6-38	15.6.6
41	Gaseous Radwaste System Leak or failure	15A.6.39	15.7.1
42	Augmented offgas Treatment System Failure	15A.6-40	15.7.1
43	Liquid Radwaste System Leak or Failure	15A.6-41	15.7.2
44	Liquid Radwaste System Storage Tank Failure	15A.6-42	15.7.3

TABLE 15A.2-5: SPECIAL (PLANT CAPABILITY) EVENTS
Cross-Correlation References

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.
45	Shipping Cask Drop - Solid Radwaste - Spent Fuel - New Fuel	No Figure	15.7.5
46	Reactor Shutdown From Anticipated Transient Without SCRAM(ATWS)	15A.6-43	15.8
47	Reactor Shutdown From Outside Control Room	15A.6-44	7.4.1.4
48	Reactor Shutdown Without Control Rods	15A.6-45	9.3.5

TABLE 15A.2-6:PLANT EVENT CATEGORY: PLANNED (NORMAL) OPERATION UNACCEPTABLE RESULTS CRITERIA

Unacceptable Results

1-1. Release of radioactive material to the environs that exceeds the limits of either 10 CRF 20 or 10 CFR 50, Appendix I.

1.2. Fuel failure to such an extent that were the freed fission products released to the environs via the normal discharge paths for radioactive material, the limits of 10 CFR 20 would be exceeded.

1-3. Nuclear system stress in excess of that allowed for planned operation by applicable industry codes.

1-4. Existence of a plant condition not considered by plant safety analyses.

TABLE 15A.2-7:PLANT EVENT CATEGORY: ANTICIPATED (EXPECTED) OPERATIONAL TRANSIENTS UNACCEPATABLE RESULTS CRITERIA

Unacceptable Results

2-1. Release of radioactive material to the environs that exceeds the limits of 10 CFR 20.

2.2. Any fuel failure calculated as a direct result of the transient analyses.

2-3. Nuclear system stress exceeding that allowed for transients by applicable industry codes.

2-4. Containment stresses exceeding that allowed for transients by applicable industry codes when containment is required.

TABLE 15A.2-8: PLANT EVENT CATEGORY: ABNORMAL (UNEXPECTED) OPERATIONAL TRANSIENTS

UNACCEPTABLE RESULTS CRITERIA

	Unacceptable Results
3-1.	Radioactive material release exceeding the guideline values of 1/10 of 10 CFR 50.67.
*3-2.	Failure of the fuel barrier as a result of exceeding mechanical or thermal limits.
3-3.	Nuclear system stresses exceeding that allowed for transients by applicable industry codes.
3-4.	Containment stresses exceeding that allowed for accidents by applicable industry codes when containment is required.

* Failure of the fuel barrier means gross core-wide fuel cladding perforations.

TABLE 15A.2-9: PLANT EVENT CATEGORY: DESIGN BASIS (POSTULATED) ACCIDENTS UNACCEPTABLE RESULTS CRITERIA

	Unacceptable Results
4-1.	Radioactive material release exceeding the guideline values of 10 CFR 50.67.
**4-2.	Failure of the fuel barrier as a result of exceeding mechanical or thermal limits.
4-3.	Nuclear system stresses exceeding that allowed for accidents by applicable industry codes.
4-4.	Containment stresses exceeding that allowed for accidents by applicable industry codes when containment is required.
4-5.	Overexposure to radiation of plant control room personnel.

**Failure of the fuel barrier includes fuel cladding fragmentation (loss-of-coolant accident) and excessive fuel enthalpy (control rod drop accident).

TABLE 15A.2-10:PLANT EVENT CATEGORY: SPECIAL (PLANT CAPABILITY) EVENTS
UNACCEPATABLE RESULTS CONSIDERATIONS

<u>Special Events Considered</u>

A. Reactor shutdown from outside control room

B. Reactor shutdown without control rods

C. Reactor shutdown with anticipated transient Without scram (ATWS)

<u>Capability Demonstration</u>

5-1. Ability to shut down reactor by manipulating controls and equipment outside the control room.

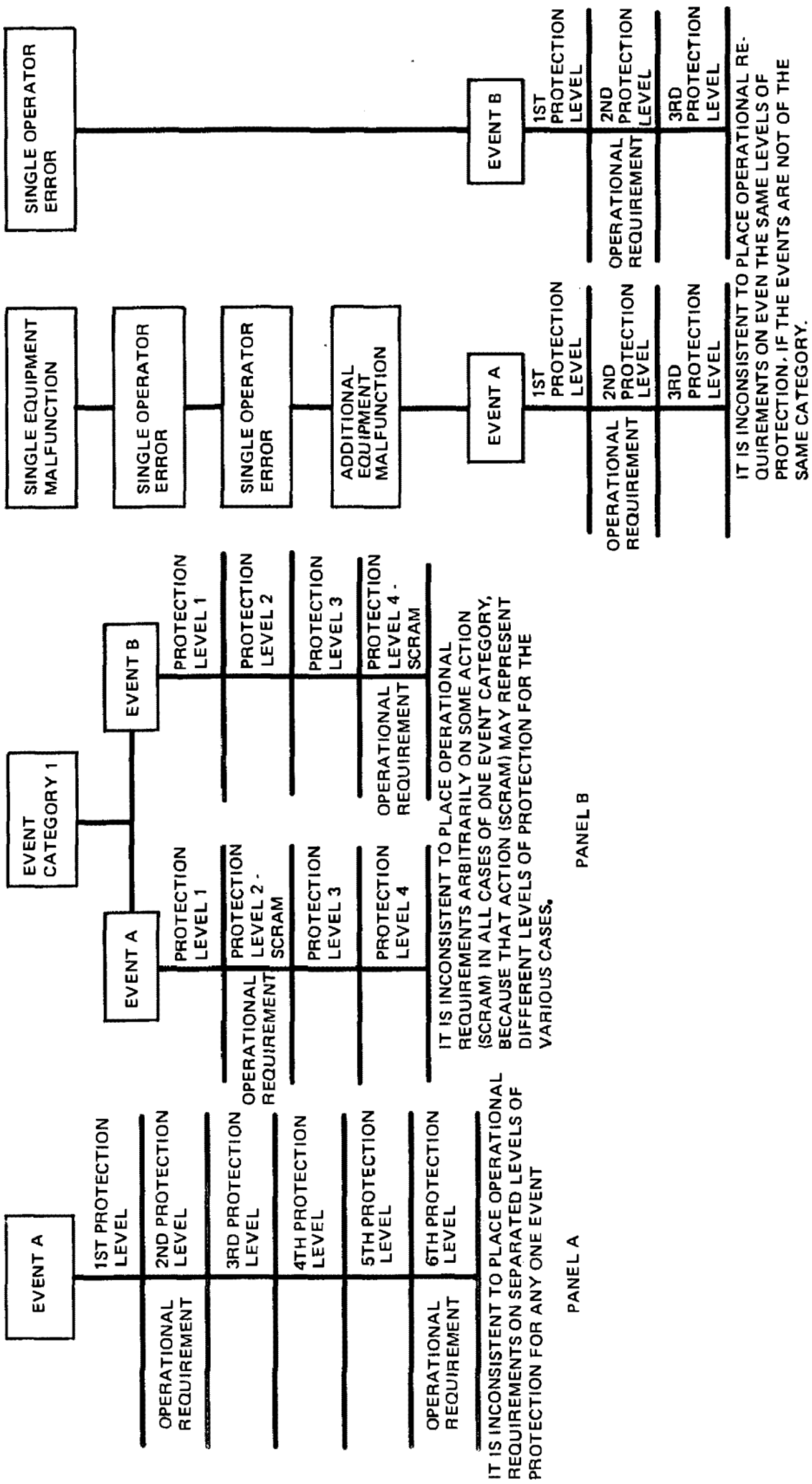
5.2. Ability to bring the reactor to the cold shutdown condition from outside the control room.

5-3. Ability to shut down the reactor independent of control rods.

TABLE 15A.2-11: NONSAFETY-GRADE SYSTEMS/COMPONENTS ASSUMED IN FSAR ANALYSES

Moderate Frequency Events

FSAR Subsection	Transient	Nonsafety-Grade System or Component
15.1.2	Feedwater controller failure, max demand	Level 9 turbine trip, turbine bypass
15.2.2	Load rejection	Turbine bypass
15.2.5	Loss of condenser vacuum	Turbine bypass
15.2.6	Loss of ac power	Level 9 turbine trip, turbine bypass
15.2.7	Loss of all feedwater flow	Recirculation runback
15.3.1	Trip of both recirculation pumps	Level 9 turbine trip, turbine bypass
15.3.2	Recirculation control failure - decreasing flow	Level 9 turbine trip, turbine bypass
15.4.5	Recirculation control failure - increasing flow	Level 9 turbine trip, turbine bypass
15.5.1	Inadvertent startup of HPCS	Level 9 turbine trip, turbine bypass
	Limiting Fault Event	
15.3.3	Recirculation pump seizure	Level 9 turbine trip, turbine bypass



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

POSSIBLE INCONSISTENCIES IN THE SELECTION OF NUCLEAR SAFETY OPERATIONAL REQUIREMENTS
FIGURE 15A.2-1

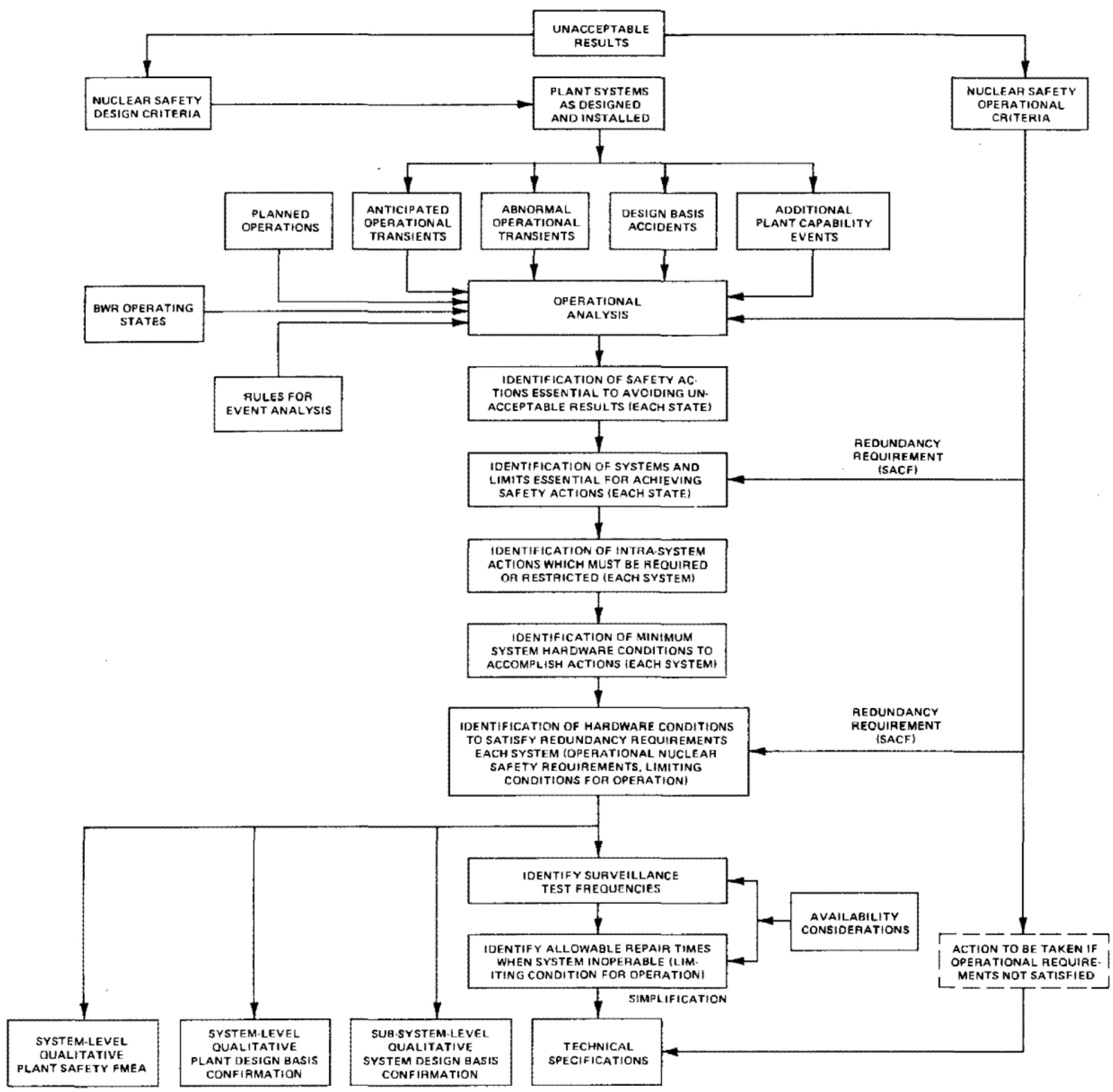
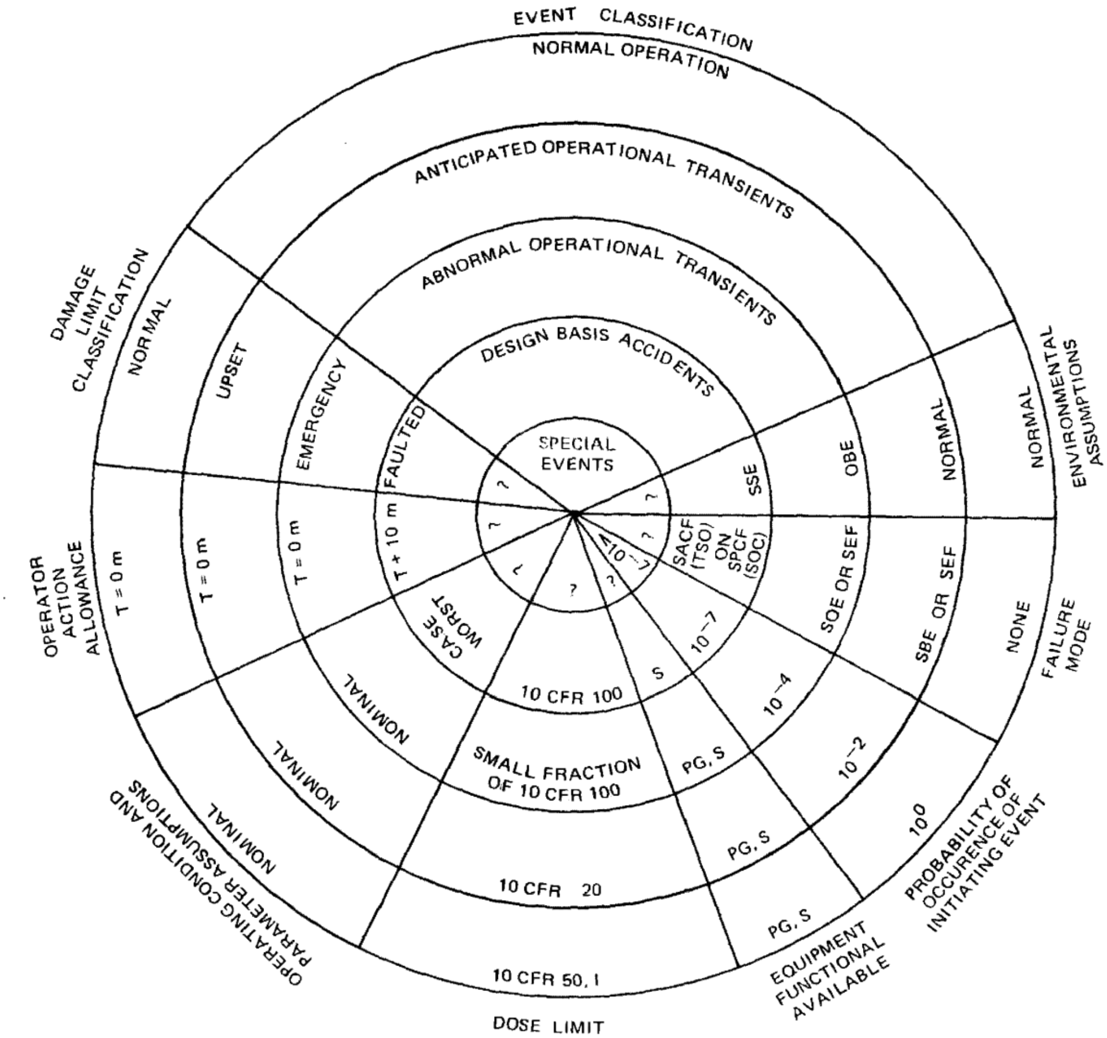


FIGURE 15A.2-2 - BLOCK DIAGRAM OF METHOD USED TO DERIVE NUCLEAR SAFETY OPERATIONAL REQUIRMENTS SYSTEM AND SUBSYSTEM LEVEL QUALITATIVE FMEA AND DESIGN BASIS CONFIRMATION AUDITS AND TECHNICAL SPECIFICATIONS



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 **UPDATED FINAL SAFETY ANALYSIS REPORT**
SIMPLIFIED NSOA CLASSIFICATION INTERRELATIONSHIPS FIGURE 15A.2-3

15A.3 METHOD OF ANALYSIS

15A.3.1 General Approach

The NSOA is performed assuming that the plant design has been established. The end products of the analysis are the nuclear safety operational requirements and the restrictions on plant hardware and its operation that must be observed (1) to satisfy the nuclear safety operational criteria, and (2) to show compliance of the plant safety and power generation systems with plant wide requirements. Figure 15A.2-2 shows the process used in the analysis. The following inputs are required for the analysis of specific plant events:

a. Applicable unacceptable results (subsection 15A.2.7)

b. Applicable nuclear safety operational criteria (subsection 15A.2.8)

c. Definition of BWR operating states (subsection 15A.3.2)

d. Event selection criteria (subsection 15A.3.3)

e. Rules for event analysis (subsection 15A.3.5)

With this information, each selected event is evaluated to determine systematically, the actions, the systems, and the limits essential to avoiding the defined unacceptable results. The essential plant components and limits so identified are considered to be in agreement with and subject to nuclear operational, design basis requirements and technical specification restrictions.

15A.3.2 BWR Operating States

Four BWR operating states in which the reactor can exist are defined in Table 15A.3-1. The main objective in selecting operating states is to divide the BWR operating spectrum into sets of initial conditions to facilitate consideration of various events in each state.

Each operating state includes a wide spectrum of values for important plant parameters. Within each state, these parameters are considered over their entire range to determine the limits on their values necessary to satisfy the nuclear safety operational criteria. Such limitations are presented in the subsections of the safety analysis report that describe the systems associated with the parameter limit. The plant parameters to be considered in this manner include the following:

Reactor coolant temperature

Reactor vessel water level

Reactor vessel pressure

Reactor vessel water quality

Reactor coolant forced circulation flow rate

Reactor power level (thermal and neutron flux)

Core neutron flux distribution

Feedwater temperature

Containment temperature and pressure

Suppression pool water temperature and level

Spent fuel pool water temperature and level

15A.3.3 Selection of Events for Analysis

15A.3.3.1 Planned Operations

"Planned Operation" refers to normal plant operation under predetermined conditions in the absence of significant abnormalities. Operations subsequent to an incident (transient, accident, or additional plant capability event) are not considered planned operations until the actions taken or equipment used in the plant are identical to those that would be used had the incident not occurred. As defined, the planned operations can be considered as a chronological sequence: refueling outage achieving criticality heatup power operation achieving shutdown cooldown refueling outage.

The planned operations are defined below.

a. Refueling outage: Includes all the planned operations associated with a normal refueling outage except those tests in which the reactor is taken critical and returned to the shutdown condition. The following planned operations are included in refueling outage:

   1. Planned, physical movement of core components (fuel, control rods, etc.)

   2. Refueling test operations (except criticality and shutdown margin tests)

   3. Planned maintenance

   4. Required inspection

b. Achieving criticality: Includes all the plant actions normally accomplished in bringing the plant from a condition in which all control rods are fully inserted to a condition in which nuclear criticality is achieved and maintained

c. Heatup: Begins where achieving criticality ends and includes all plant actions normally accomplished in approaching nuclear system rated temperature and pressure by using nuclear power (reactor critical).

   Heatup extends through warmup and synchronization of the main turbine-generator

d. Power operation: Begins where heatup ends and includes continued plant operation at power levels in excess of heatup power

e. Achieving shutdown: Begins where the main generator is unloaded and includes all plant actions normally accomplished in achieving nuclear shutdown (more than one rod subcritical) following power operation

f. Cooldown: Begins where achieving shutdown ends and includes all plant actions normal to the continued removal of decay heat and the reduction of nuclear system temperature and pressure

The exact point at which some of the planned operations end and others begin cannot be precisely determined. It is shown in later sections of this appendix that such precision is not required, since the protection requirements are adequately defined in passing from one state to the next. Dependence of several planned operations on the one rod subcritical condition provides an exact point on either side of which protection (especially scram) requirements differ. Thus, where a precise boundary between planned operations is needed, the definitions provide the needed precision.

Together, the BWR operating states and the planned operations define the full spectrum of conditions from which transients, accidents, and special events are initiated. The BWR operating states define only the physical condition (pressure, temperature, etc.) of the reactor; the planned operations define what the plant is doing. The separation of physical conditions from the operation being performed is deliberate and facilitates careful consideration of all possible initial conditions from which incidents may occur.

15A.3.3.2 Anticipated (Expected) Operational Transients

To select anticipated operational transients, eight nuclear system parameter variations are considered as potential initiating causes of threats to the fuel and the reactor coolant pressure boundary. The parameter variations are as follows:

a. Nuclear system pressure increase

b. Reactor vessel water (moderator) temperature decrease

c. Positive reactivity insertion

d. Reactor vessel coolant inventory decrease

e. Reactor core coolant flow decrease

f. Reactor core coolant flow increase

g. Core coolant temperature increase

h. Excess of coolant inventory

These parameter variations, if uncontrolled, could result in damage to the reactor fuel or reactor coolant pressure boundary, or both. A nuclear system pressure increase threatens to rupture the reactor coolant pressure boundary from internal pressure. A pressure increase also collapses voids in the moderator, causing an insertion of positive reactivity that threatens fuel damage as a result of overheating. A reactor vessel water (moderator) temperature decrease results in an insertion of positive reactivity as density increases. This could lead to fuel overheating. Positive reactivity insertions are possible from causes other than nuclear system pressure or moderator temperature changes. Such reactivity insertions threaten fuel damage caused by overheating. Both a reactor vessel coolant inventory decrease and a reduction in coolant flow through the core threaten to overheat the fuel as the coolant becomes unable to adequately remove the heat generated in the core. An increase in coolant flow through the core reduces the void content of the moderator, resulting in an increased fission rate. A core coolant temperature increase threatens the integrity of the fuel; such a variation could be the result of a heat exchanger malfunction during operation in the shutdown cooling mode. An excess of coolant inventory could be the result of malfunctioning water level control equipment; such a malfunction can result in a turbine trip, which causes an increase in nuclear system pressure and an increased fission rate.

The eight parameter variations listed above include all effects within the nuclear system caused by anticipated operational transients that threaten the integrity of the reactor fuel or reactor coolant pressure boundary. Variation of any one parameter may cause a change in another listed parameter; however, for analysis purposes, threats to barrier integrity are evaluated by groups according to the parameter variation originating the threat. For example, positive reactivity insertions resulting from sudden pressure increases are evaluated in the group of threats stemming from nuclear system pressure increases.

Anticipated operational transients are defined as transients resulting from single equipment failures or single operator errors that can be reasonably expected (moderate probability of occurrence - once per day to once in 20 years) during any mode of plant operation. Examples of this range of probability of single operational failures or operator errors are:

a. Opening or closing any single valve (a check valve is not assumed to close against normal flow)

b. Starting or stopping any single component

c. Malfunction or maloperation of any single control device

d. Any single electrical failure

e. Any single operator error

An operator error is defined as an active deviation from written operating procedures or nuclear plant standard operating practices. A single operator error is the set of actions that is a direct consequence of a single reasonably expected erroneous decision. The set of actions is limited as follows:

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

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

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

Examples of single operator errors are as follows:

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

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

c. An incorrect calibration of an average power range monitor

d. Manual isolation of the main steam lines caused by operator misinterpretation of an alarm or indication

The five types of a single operator error or a single equipment malfunction are applied to various plant systems with a consideration for a variety of plant conditions to discover events directly resulting in any undesired parameter variations listed. Once discovered, each event is evaluated for the threat it poses to the integrity of the radioactive material barriers.

15A.3.3.3 Abnormal (Unexpected) Operational Transient

To select abnormal operational transients, eight nuclear system parameter variations are considered as potential initiating causes of gross core-wide fuel failures and threats to the reactor coolant pressure boundary. The parameter variations are as follows:

a. Nuclear system pressure increase

b. Reactor vessel water (moderator) temperature decrease

c. Positive reactivity insertion

d. Reactor vessel coolant inventory decrease

e. Reactor core coolant flow decrease

f. Reactor core coolant flow increase

g. Core coolant temperature increase

h. Excess of coolant inventory

These parameter variations, if uncontrolled, could result in gross core-wide reactor fuel failure or damage to the reactor coolant pressure boundary, or both.

The eight parameter variations listed above include all effects within the nuclear system caused by abnormal operational transients that threaten gross core-wide reactor fuel integrity or seriously affect reactor coolant pressure boundary. Variation of any one parameter may cause a change in another listed parameter; however, for analysis purposes, threats to barrier integrity are evaluated by groups according to the parameter variation originating the threat. For example, positive reactivity insertions resulting from sudden pressure increases are evaluated in the group of threats stemming from nuclear system pressure increases.

Abnormal operational transients are defined as incidents resulting from single or multiple equipment failures and/or single or multiple operator errors that are not reasonably expected (less than one event in 20 years to one in 100 years) during any mode of plant operation. Examples of single or multiple operational failures and/or single or multiple operator errors are:

a. Catastrophic failure of major power generation equipment components

b. Multiple electrical failures

c. Multiple operator errors

d. Combinations of an equipment and an operator error

Operator error is defined as an active deviation from written operating procedures or nuclear plant standard operating practices. A multiple operator error is the set of actions that is a direct consequence of several unexpected erroneous decisions.

Examples of multiple operator errors are as follows:

a. Inadvertent loading and operating a fuel assembly in an improper position

b. The movement of a control rod during refueling operations. The various types of single errors and/or single malfunctions are applied to various plant systems with a consideration for a variety of plant conditions to discover events directly resulting in any undesired parameter variations listed. Once discovered, each event is evaluated for the threat it poses to the integrity of the various radioactive material barriers.

15A.3.3.4 Accidents

Accidents are defined as hypothesized events that affect one or more of the radioactive material barriers and that are not expected during plant operations. These are plant events, equipment failures, or combinations of initial conditions which are of extremely low probability (once in 100 years to once in 10,000 years). The postulated accident types considered are as follows:

a. Mechanical failure of a single component leading to the release of radioactive material from one or more barriers. The components referred to here are not those that act as radioactive material barriers. Examples of mechanical failure are breakage of the coupling between a control rod drive and the control rod.

b. Arbitrary rupture of any single pipe up to and including complete severance of the largest pipe in the reactor coolant pressure boundary. This kind of accident is considered only under conditions in which the nuclear system is pressurized.

For purposes of analysis, accidents are categorized as those events that result in releasing radioactive material:

a. From the fuel with the reactor coolant pressure boundary, containment and auxiliary buildings initially intact (Event 35)

b. Directly to the containment(Event 37)

c. Directly to the auxiliary or turbine buildings with the containment initially intact (Events 35, 38, 39, 40, 45)

d. Directly to the auxiliary buildings with the containment not intact (Events 36, 45)

e. Directly to the fuel handling area (Event 36, 45)

f. Directly to the turbine building (Events 41, 42)

g. Directly to the environs (Events 43, 44)

The effects of various accident types are investigated, with a consideration for the full spectrum of plant conditions, to examine events that result in the release of radioactive material. The accidents resulting in potential radiation exposures greater than any other accident considered under the same general accident assumptions are designated design basis accidents.

15A.3.3.5 Additional Special Plant Capability Events

A number of additional events are evaluated to demonstrate plant capabilities relative to special arbitrary nuclear safety criteria. These special events involve very, very low probability occurrence situations. As an example, the adequacy of the redundant reactivity control system is demonstrated by evaluating the special event: "reactor shut-down without control rods." Another similar example, the capability to perform a safe shut-down from outside the control room is demonstrated by evaluating the special event "reactor shut-down from outside the control room."

15A.3.4 Applicability of Events to Operating States

The first step in performing an operational analysis for a given "incident" (transient, accident, or special event) is to determine in which operating states the incident can occur. An incident is considered applicable within an operating state if the incident can be initiated from the physical conditions that characterize the operating state. Applicability of the "planned operations" to the operating states follows from the definitions of planned operations. A planned operation is considered applicable within an operating state if the planned operation can be conducted when the reactor exists under the physical conditions defining the operating state.

15A.3.5 Rules for Event Analysis

The following functional rules are followed in performing SACF, operational and design basis analyses for the various plant events:

a. An action, system, or limit is considered essential only if it is essential to avoiding an unacceptable result or satisfying the nuclear safety operational criteria.

b. The full range of initial conditions (as defined in paragraph 15A.3.5.(c)) is considered for each event analyzed so that all essential protection sequences are identified. Consideration is not limited to "worst cases" because lesser cases sometimes require more restrictive actions or systems different from the "worst cases".

c. The initial conditions for transients, accidents, and additional plant capability events are limited to conditions that would exist during planned operations in the applicable operating state.

d. For planned operations, consideration is made only for actions, limits, and systems essential to avoiding the unacceptable results during operation in that state (as opposed to transients, accidents, and additional plant capability events, which are followed through to completion). Planned operations are treated differently from other events because the transfer from one state to another during planned operations is deliberate. For events other than planned operations, the transfer from one state to another may be unavoidable.

e. Limits are derived only for those essential parameters that are continuously monitored by the operator. Parameter limits associated with the required performance of an essential system are considered to be included in the requirement for the operability of the system. Limits on frequently monitored process parameters are called "envelope limits," and limits on parameters associated with the operability of a safety system are called "operability limits." Systems associated with the control of the envelope parameters are considered nonessential if it is possible to place the plant in a safe condition without using the system in question.

f. For transients, accidents and additional plant capability events, consideration is made for the entire duration of the event and aftermath until some planned operation is resumed. Planned operation is considered resumed when the procedures being followed or equipment being used are identical to those used during any one of the defined planned operations.

g. Credit for operator action is taken on a case-by-case basis depending on the conditions that would exist at the time operator action would be required. Because transients, accidents, and additional plant capability events are considered through the entire duration of the event until planned operation is resumed, manual operation of certain systems is sometimes required following the more rapid or automatic portions of the event. Credit for operator action is taken only when the operator can reasonably be expected to accomplish the required action under the existing conditions.

h. For transients, accidents, and additional plant capability events, only those actions, limits, and systems are considered essential for which there arises a unique requirement as a result of the event. For instance, if a system that was operating prior to the event (during planned operation) is to be employed in the same manner following the event and if the event did not affect the operation of the system, then the system would not appear on the protection sequence diagram.

i. The operational analyses identifies all the support or auxiliary systems essential to the functioning of the front-line safety systems. Safety system auxiliaries whose failure results in safe failure of the front-line safety systems are considered nonessential.

j. A system or action that plays a unique role in the response to a transient, accident, or additional plant capability event is considered essential unless the effects of the system or action are not included in the detailed analysis of the event.

15A.3.6 Steps in an Operational Analysis

All information needed to perform an operational analysis for each plant event has been presented (Figure 15A.2-2). The procedure followed in performing an operational analysis for a given event (selected according to the event selection criteria) is as follows:

a. Determine the BWR operating states in which the event is applicable.

b. Identify all the essential protection sequences (safety actions and front-line safety systems) for the event in each applicable operating state.

c. Identify all the safety system auxiliaries essential to the functioning of the front-line safety systems.

The above three steps are performed in later sections of this appendix.

To derive the operational requirements and technical specifications for the individual components of a system included in any essential protection sequence, the following steps are taken:

a. Identify all the essential actions within the system (intrasystem actions) necessary for the system to function to the degree necessary to avoid the unacceptable results.

b. Identify the minimum hardware conditions necessary for the system to accomplish the minimum intrasystem actions.

c. If the single-failure criterion applies, identify the additional hardware conditions necessary to achieve the plant safety actions (scram, pressure relief, isolation, cooling, etc.) in spite of single failures. This step gives the nuclear safety operational requirements for the plant components so identified.

d. Identify surveillance requirements and allowable repair times for the essential plant hardware (subsection 15A.5.2).

e. Simplify the operational requirements determined in steps c. and d. so that technical specifications may be obtained that encompass the true operational requirements and are easily used by plant operations and management personnel.

TABLE 15A.3-1: BWR OPERATING STATES

	States			
Conditions	**A**	**B**	**C**	**D**
Reactor vessel head off	X*	X*		
Reactor vessel head on			X	X
Shutdown	X		X	
Not shutdown		X		X

Definition

Shutdown: K_{eff} sufficiently less than 1.0 that the full withdrawal of any one control rod could not produce criticality under the most restrictive potential conditions of temperature, pressure, core age, and fission product concentrations

*Because the reactor vessel head is off in states A and B, pressure is atmospheric.

15A.4 DISPLAY OF OPERATIONAL ANALYSIS RESULTS

15A.4.1 General

To identify and establish fully the requirements, restrictions, and limitations that are to be observed during plant operation, plant systems and components are related to the needs for their actions in satisfying the nuclear safety operational criteria. This Appendix displays these relationships in a series of block diagrams.

Table 15A.3-1 indicates in which operating states each event is applicable. For each event, a block diagram is presented showing the conditions and systems required to achieve each essential safety action. The block diagrams show only those systems necessary to provide the safety actions such that the nuclear safety operational and design basis criteria are satisfied. The total plant capability to provide a safety action is generally not shown, only the minimum capability essential to satisfying the operational criteria. The BWR design is based on the bounding transient and assumed safety system failures. The NSOAs are intended to provide assurance that the bounding case is truly the limiting case. Therefore, only in appropriate cases are some of the nonsafety systems shown in the diagrams. It is very important to understand that only enough protective equipment is cited in the diagram to provide the necessary action. Many events can utilize many more paths to success than are shown. These operational analyses involve the minimum equipment needed to prevent or avert an unacceptable result. Thus, the diagrams depict all essential protection sequences for each event with the least amount of protective equipment needed. Once all of these protection sequences are identified in block diagram form, system requirements are derived by considering all events in which the particular system is employed. The analysis considers the following conceptual aspects:

a. The BWR operating state

b. Types of operations or events that are possible within the operating state

c. Relationships of certain safety actions to the unacceptable results and to specific types of operations and events

d. Relationships of certain systems to safety actions and to specific types of operations and events

e. Supporting or auxiliary systems essential to the operation of the front-line safety systems

f. Functional redundancy (The single-failure criterion applied at the safety action level. This is, in effect, a qualitative/ system level/FMEA-type analysis.)

Each block in the sequence diagrams represents a finding of essentiality for the safety action, system, or limit under consideration. Essentiality in this context means that the safety action, system, or limit is essential to satisfying the nuclear safety operational criteria. Essentiality is determined through an analysis in which the safety action, system, or limit being considered is completely disregarded in the analyses of the applicable operations or events. If the nuclear safety operational criteria are satisfied without the safety action, system, or limit, then the safety action, system, or limit is not essential, and no operational nuclear safety requirement would be indicated. When the disregarding of a safety action, system, or limit results in violating one or more nuclear safety operational criteria, the safety action, system, or limit is considered essential, and the resulting operational nuclear safety requirements can be related to specific criteria and unacceptable results.

15A.4.2 Protection Sequence and Safety System Auxiliary Diagrams

Block diagrams illustrate essential protection sequences for each event requiring unique safety actions. These protection sequence diagrams show only the required front-line safety systems. The format and conventions used for these diagrams are shown in Figure 15A.4-1.

A special note is provided on Figure 15A.4-1. The NSOA in this document reflects a combined but individual examination of the most recent three BWR product line plant designs together. The several small differences between the individual designs and their combined presentations are easily identified by special symbol notations. Where a difference is not noted, the designs are functionally identical.

The auxiliary systems essential to the correct functioning of front-line safety systems are shown on safety system auxiliary diagrams. The format used for these diagrams is shown in Figure 15A.4-2. The diagram indicates that auxiliary systems A, B, and C are required for proper operation of front-line safety system X.

Total plant requirements for an auxiliary system or the relationships of a particular auxiliary system to all other safety systems (frontline and auxiliary) within an operating state are shown on the commonality of auxiliary diagrams. The format used for these diagrams is shown in Figure 15A.4-3. The convention employed in Figure 15A.4-3 indicates that auxiliary system A is required:

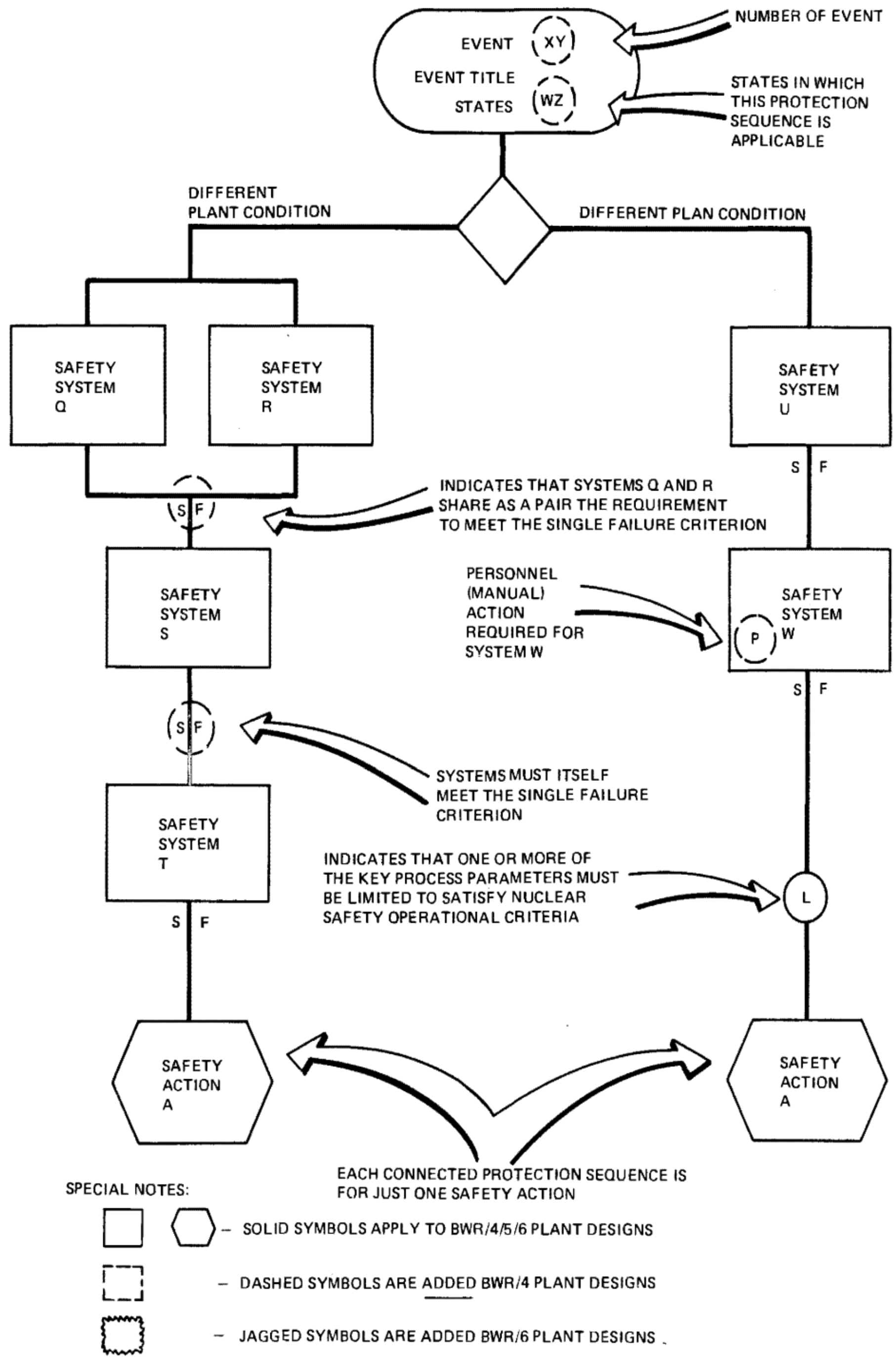
a. To be single-failure proof relative to system γ in state A-events X, Y; state B-events X, Y; state C events X, Y, Z; state D-events X, Y, Z

b. To be single-failure proof relative to the parallel combination of systems α and β in state A-events U, V, W; state B-events V, W; state C-events U, V, W, X; state D-events U, V, W, X

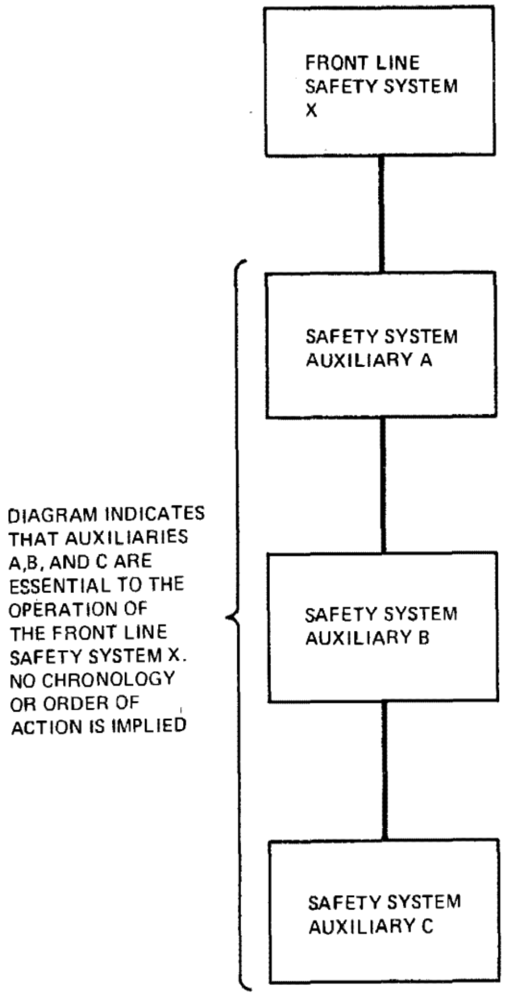
c. To be single-failure proof relative to the parallel combination of system π and [system ε in series with the parallel combination of systems ξ and ψ] in state C-events Y, W; state D-events Y, W, Z. As noted, system ε is part of the combination but does not require auxiliary system A for its proper operation.

d. For system δ in state B-events Q, R; state D-events Q, R, S

With these three types of diagrams, it is possible to determine for each system the detailed functional requirements and conditions to be observed regarding system hardware in each operating state. The detailed conditions to be observed regarding system hardware include such nuclear safety operational requirements as test frequencies and the number of components that must be operable.



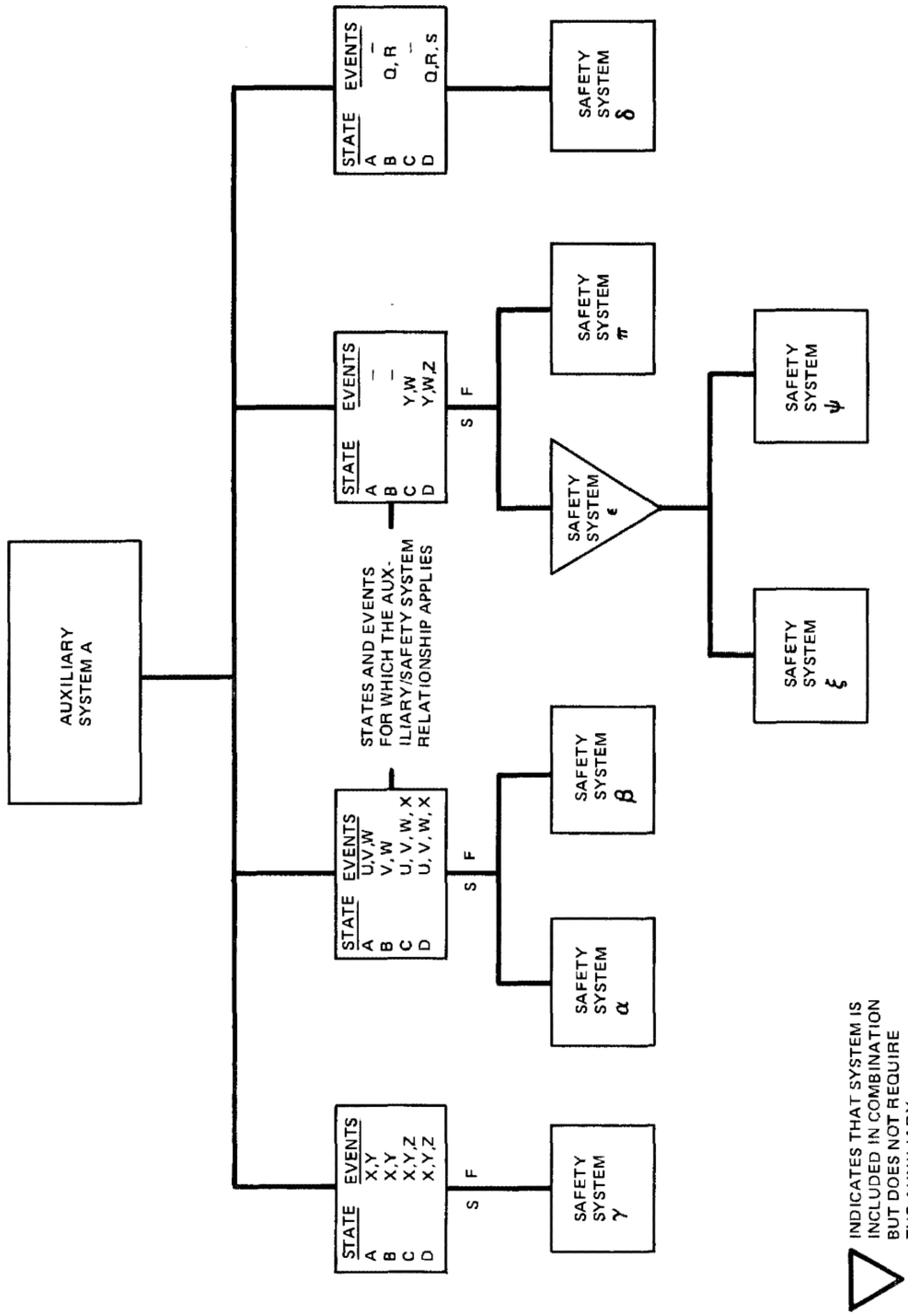
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	FORMAT FOR PROTECTION SEQUENCE DIAGRAMS FIGURE 15A.4-1



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

FORMAT FOR SAFETY SYSTEM
AUXILIARY DIAGRAMS

FIGURE 15A.4-2



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UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

FORMAT FOR COMMONALITY OF AUXILIARY DIAGRAMS

FIGURE 15A.4-3

15A.5 BASES FOR SELECTING SURVEILLANCE TEST FREQUENCIES

15A.5.1 Normal Surveillance Test Frequencies

After the essential nuclear safety systems and engineered safeguards have been identified by applying the nuclear safety operational criteria, surveillance requirements are selected for these systems. In this selection process, the various systems are considered in terms of relative availability, test capability, plant conditions necessary for testing, and engineering experience with the system type. The surveillance test frequency selected represents the application of engineering judgment integrating all of these considerations. However, the selected frequencies are conservative with respect to the surveillance requirements needed to maintain the availability of the system in excess of the design goal.

15A.5.2 Allowable Repair Times

Allowable repair times are selected by computation using availability analysis methods (Ref. 1 of Section 15A.9) for redundant standby systems. The resulting maximum average allowable repair times assure that a system's long-term availability, including allowance for repair, is not reduced below the theoretical availability that would be achieved if repairs could be made in zero time.

15A.5.3 Repair Time Rule

A safety system can be repaired while the reactor is in operation if the repair time is equal to or less than the maximum allowable average repair time. If repair is not complete when the allowable repair time expires, the plant must be placed in its safest mode (with respect to the protection lost).

To maintain the validity of the assumptions used to establish the above repair time rule, the following restrictions must be observed:

a. The allowable repair time will be used to restore failed safety-related equipment and perform necessary maintenance to ensure that the equipment remains operable. Using this time will be kept to a minimum. Other maintenance will be scheduled when the equipment is not needed.

b. When a failure is discovered by test, all the redundant components should be tested to establish that they are good at the beginning of the repair time for the failed component and do not suffer from the same failure mode discovered in the failed component.

   If there are multiple failures (which exceed the limits of the technical specifications) of the same mode, the repair time allowance does not apply and the plant must be placed in a condition in which the actions of the safety system are not essential to avoiding the unacceptable safety results.

c. At the conclusion of the repair, the repaired component must be retested and placed in service. The redundant components must also be retested, not only to validate the assumptions, but to assure that the repair did not inadvertently invalidate a good component.

d. Once the need for repair of a failed component is discovered, repairs should proceed as quickly as possible consistent with good craftsmanship.

Alternatively, if a system is expected to be out of repair for an extended time, the availability of the remaining systems can be maintained at the prefailure level by testing them more often. This technique is fully developed in Reference 1 of Section 15A.9

15A.6 OPERATIONAL ANALYSES

Results of the operational analyses for a BWR 6 plant are discussed in the following paragraphs and displayed on Figures 15A.6-1 through 15A.6-46. Tables 15A.6-1 through 15A.6-5 indicate the BWR operating states in which each of the approximate 50 events is applicable.

15A.6.1 Safety System Auxiliaries

Figures 15A.6-1 and 15A.6-2 show the safety system auxiliaries essential to the functioning of each front-line safety system. Commonality of auxiliary diagrams are shown in Figures 15A.6-46 through 15A.6-51.

15A.6.2 Planned (Normal) Operations

15A.6.2.1 General

Requirements for the planned operations normally involve limits (L) on certain key process variables and restrictions (R) on certain plant equipment. The control block diagrams for each operating state (Figures 15A.6-3 through 15A.6-6) show only those controls necessary to avoid unacceptable safety results 1-1 through 1-4. Refer to Table 15A.2-6 for unacceptable results criteria.

Following is a description of the planned operations (Events 1 through 6), as they pertain to each of the four operating states. The description of each operating state contains a definition of that state, a list of the planned operations that apply to that state, and a list of the safety actions that are required to avoid the unacceptable safety results.

15A.6.2.2 Event Definitions

Event 1 - Refueling Outage

Refueling outage includes all the planned operations associated with a normal refueling outage except those tests in which the reactor is made critical and returned to the shutdown condition. The following planned operations are included in refueling outage:

a. Planned, physical movement of core components (fuel, control rods, etc.)

b. Refueling test operations (except criticality and shutdown margin tests)

c. Planned maintenance

d. Required inspection

Event 2 - Achieving Criticality

Achieving criticality includes all the plant actions normally accomplished in bringing the plant from a condition in which all control rods are fully inserted to a condition in which nuclear criticality is achieved and maintained.

Event 3 - Reactor Heatup

Heatup begins where achieving criticality ends and includes all plant actions normally accomplished in approaching nuclear system rated temperature and pressure by using nuclear power (reactor critical). Heatup extends through warmup and synchronization of the main turbine generator.

Event 4 - Power Operation - Electric Generation

Power operation begins where heatup ends and continued plant operation at power levels in excess of heatup power or steady state operation begins. It also includes plant maneuvers such as:

a. Daily electrical load reduction and recoveries

b. Electrical grid frequency control adjustment

c. Control rod/reactor fuel/core management movements

d. Power generation surveillance testing involving:

    1. Turbine stop valve closing

    2. Turbine control valve adjustments

    3. MSIV exercising

Event 5 - Achieving Reactor Shutdown

Achieving shutdown begins where the main generator is unloaded and includes all plant actions normally accomplished in achieving nuclear shutdown (more than one rod subcritical) after power operation.

Event 6 - Reactor Cooldown

Cooldown begins where achieving shutdown ends and includes all plant actions normal to the continued removal of decay heat and the reduction of nuclear system temperature and pressure.

15A.6.2.3 Required Safety Actions/Related Unacceptable Results

The following paragraphs describe the safety actions for planned operations. Each description includes a selection of the operating states that apply to the safety action, the plant system affected by limits or restrictions, and the unacceptable result that is avoided. The four operating states are defined in Table 15A.3-1. The unacceptable results criteria are tabulated in Table 15A.2-6.

15A.6.2.3.1 Radioactive Material Release Control

Radioactive materials may be released to the environs in any operating state; therefore, radioactive material release control is required in all operating states. Because of the significance of preventing excessive release of radioactive materials to the environs, this is the only safety action for which monitoring systems are explicitly shown. The offgas vent radiation monitoring system provides indication for gaseous release through the radwaste building vent. Gaseous releases through other vents are monitored by the ventilation monitoring system. The process liquid radiation monitors are specifically not required, because all liquid wastes are monitored by batch sampling before a procedural controlled release. Limits are expressed on the offgas vent system, liquid radwaste system, and solid radwaste system so that the planned releases of radioactive materials comply with the limits given in 10 CFR 20, 10 CFR 50, and 10 CFR 71 (related unacceptable safety result 1-1).

15A.6.2.3.2 Core Coolant Flow Rate Control

In State D, when above approximately 10 percent NB rated power, the core coolant flow rate must be maintained above certain minimums (i.e., limited) to maintain the integrity of the fuel cladding (1-2) and assure the validity of the plant safety analysis (1-4).

15A.6.2.3.3 Core Power Level Control

The plant safety analyses of accidental positive reactivity additions have assumed as an initial condition that the neutron source level is above a specified minimum. Because a significant positive reactivity addition can only occur when the reactor is less than one rod subcritical, the assumed minimum source level need be observed only in States B and D. The minimum source level assumed in the analyses has been related to the counts/sec readings on the source range monitors (SRM); thus, this minimum power level limit on the fuel is expressed as a required SRM count level. Observing the limit assures validity of the plant safety analysis (1-4). Maximum core power limits are also expressed for operating States B and D to maintain fuel integrity (1-2) and remain below the maximum power levels assumed in the plant safety analysis (1-4).

15A.6.2.3.4 Core Neutron Flux Distribution Control

Core neutron flux distribution must be limited in State D, otherwise core power peaking could result in fuel failure (1-2). Additional limits are expressed in this state, because the core neutron flux distribution must be maintained within the envelope of conditions considered by plant safety analysis (1-4).

15A.6.2.3.3 Core Power Level Control

The plant safety analyses of accidental positive reactivity additions have assumed as an initial condition that the neutron source level is above a specified minimum. Because a significant positive reactivity addition can only occur when the reactor is less than one rod subcritical, the assumed minimum source level need be observed only in States B and D. The minimum source level assumed in the analyses has been related to the counts/sec readings on the source range monitors (SRM); thus, this minimum power level limit on the fuel is expressed as a required SRM count level. Observing the limit assures validity of the plant safety analysis (1-4). Maximum core power limits are also expressed for operating States B and D to maintain fuel integrity (1-2) and remain below the maximum power levels assumed in the plant safety analysis (1-4).

15A.6.2.3.4 Core Neutron Flux Distribution Control

Core neutron flux distribution must be limited in State D, otherwise core power peaking could result in fuel failure (1-2). Additional limits are expressed in this state, because the core neutron flux distribution must be maintained within the envelope of conditions considered by plant safety analysis (1-4).

15A.6.2.3.5 Reactor Vessel Water Level Control

In any operating state, the reactor vessel water level could, unless controlled, drop to a level that will not provide adequate core cooling; therefore, reactor vessel water level control applies to all operating states. Observation of the reactor vessel water level limits protects against fuel failure (1-2) and assures the validity of the plant safety analysis (1-4).

15A.6.2.3.6 Reactor Vessel Pressure Control

Reactor vessel pressure control is not needed in States A and B because vessel pressure cannot be increased above atmospheric pressure. In State C, a limit is expressed on the reactor vessel to assure that it is not hydrostatically tested until the temperature is above the NDT temperature plus 60 F; this prevents excessive stress (1-3). Also, in States C and D a limit is expressed on the residual heat removal system to assure that it is not operated in the shutdown cooling mode when the reactor vessel pressure is greater than approximately 150 psig; this prevents excessive stress (1-3). In States C and D, a limit on the reactor vessel pressure is necessitated by the plant safety analysis (1-4).

15A.6.2.3.7 Nuclear System Temperature Control

In operating States C and D, a limit is expressed on the reactor vessel to prevent the reactor vessel head bolting studs from being in tension when the temperature is less than 70 F to avoid excessive stress (1-3) on the reactor vessel flange. This limit does not apply in States A and B because the head will not be bolted in place during criticality tests or during refueling. In all operating states, a limit is expressed on the reactor vessel to prevent an excessive rate of change of the reactor vessel temperature to avoid excessive stress (1-3). In States C and D, where it is planned operation to use the feedwater system, a limit is placed on the reactor fuel so that the feedwater temperature is maintained within the envelope of conditions considered by the plant safety analysis (1-4). For State D, a limit is observed on the temperature difference between the recirculation system and the reactor vessel to prevent the starting of the recirculation pumps. This operating restriction and limit prevents excessive stress in the reactor vessel (1-3).

15A.6.2.3.8 Nuclear System Water Quality Control

In all operating states, water of improper chemical quality could produce excessive stress as a result of chemical corrosion (1-3). Therefore, a limit is placed on reactor coolant chemical quality in all operating states. For all operating states where the nuclear system can be pressurized (States C and D), an additional limit on reactor coolant activity assures the validity of the analysis of the main steam line break accident (1-4).

15A.6.2.3.9 Nuclear System Leakage Control

Because excessive nuclear system leakage could occur only while the reactor vessel is pressurized, limits are applied only to the reactor vessel in States C and D. Observing these limits prevents vessel damage due to excessive stress (1-3) and assures the validity of the plant safety analysis (1-4).

15A.6.2.3.10 Core Reactivity Control

In State A during refueling outage, a limit on core loading (fuel) to assure that core reactivity is maintained within the envelope of conditions considered by the plant safety analysis (1-4). In all states, limits are imposed on the control rod drive system to assure adequate control of core reactivity so that core reactivity remains within the envelope of conditions considered by the plant safety analysis (1-4).

15A.6.2.3.11 Control Rod Worth Control

Any time the reactor is not shut down and is generating less than 10 percent power (States B and D), a limit is imposed on the control rod pattern to assure that control rod worth is maintained within the envelope of conditions considered by the analysis of the control rod drop accident (1-4).

15A.6.2.3.12 Refueling Restriction

By definition, planned operation event 1 (refueling outage) applies only to State A. Observing the restrictions on the reactor fuel and on the operation of the control rod drive system within the specified limit maintains plant conditions within the envelope considered by the plant safety analysis (1-4).

15A.6.2.3.13 Containment Pressure and Temperature Control

In States C and D, limits are imposed on the containment and the suppression pool to maintain temperature and pressure within the envelope considered by plant safety analysis (1-4). These limits assure an environment in which instruments and equipment can operate correctly within the containment. Limits on the pressure suppression pool apply to the water temperature and water level to assure that it has the capability of absorbing the energy discharged during a safety/relief valve blowdown.

15A.6.2.3.14 Stored Fuel Shielding, Cooling, and Reactivity Control

Because both new and spent fuel will be stored during all operating states, stored fuel shielding, cooling, and reactivity control apply to all operating states. Limits are imposed on the spent fuel storage pool storage positions, water level, fuel handling procedures, and water temperature. Observing the limits on fuel storage positions assures that spent fuel reactivity remains within the envelope of conditions considered by the plant safety analysis (1-4). Observing the limits on water level assures shielding in order to maintain conditions within the envelope of conditions considered by the plant safety analysis (1-4) and provides the fuel cooling necessary to avoid fuel damage (1-2). Observing the limit on water temperature avoids excessive fuel pool stress (1-3). A limit is imposed on the new fuel storage arrangement to assure that the fuel storage geometry is maintained within the envelope of reactivity conditions considered by the plant safety analysis (1-4).

15A.6.2.4 Operational Safety Evaluations

State A

In State A the reactor is in a shutdown condition, the vessel head is off, and the vessel is at atmospheric pressure. The applicable events for planned operations are refueling outage, achieving criticality, and cooldown (Events 1, 2, and 6, respectively).

Figure 15A.6-3 shows the necessary safety actions for planned operations, the corresponding plant systems, and the event for which these actions are necessary. As indicated in the diagram, the required safety actions are as follows:

Safety Actions

Radioactive material release control

Reactor vessel water level control

Nuclear system temperature control

Nuclear system water quality control

Core reactivity control

Refueling restrictions

Stored fuel shielding, cooling, and reactivity control

State B

In State B the reactor vessel head is off, the reactor is not shutdown, and the vessel is at atmospheric pressure. Applicable planned operations are achieving criticality and achieving shutdown (Events 2 and 5, respectively).

Figure 15A.6-4 relates the necessary safety actions for planned operations, the plant systems, and the event for which the safety actions are necessary. The required safety actions for planned operation in State B are as follows:

Safety Actions

Radioactive material release control

Core power level control

Reactor vessel water level control

Nuclear system temperature control

Nuclear system water quality control

Core reactivity control

Rod worth control

Stored fuel shielding, cooling, and reactivity control

State C

In State C the reactor vessel head is on and the reactor is shut down. Applicable planned operations are achieving criticality and cooldown (Events 2 and 6, respectively).

Sequence diagrams relating essential safety actions for planned operations, plant systems, and applicable events are shown in Figure 15A.6-5. The required safety actions for planned operation in State C are as follows:

Safety Actions

Radioactive material release control

Reactor vessel water level control

Reactor vessel pressure control

Nuclear system temperature control

Nuclear system water quality control

Nuclear system leakage control

Core reactivity control

Containment pressure and temperature control

Stored fuel shielding, cooling, and reactivity control

State D

In State D the reactor vessel head is on and the reactor is not shutdown. Applicable planned operations are achieving criticality, heatup, power operation and achieving shutdown (Events 2, 3, 4, and 5, respectively).

Figure 15A.6-6 relates essential safety actions for planned operations, corresponding plant systems, and events for which the safety actions are necessary. The required safety actions for planned operation in State D are as follows:

Safety Actions

Radioactive material release control

Core coolant flow rate control

Core power level control

Core neutron flux distribution control

Reactor vessel water level control

Reactor vessel pressure control

Nuclear system temperature control

Nuclear system water quality control

Nuclear system leakage control

Core reactivity control

Rod worth control

Containment pressure and temperature control

Stored fuel shielding, cooling, and reactivity control

15A.6.3 Anticipated (Expected) Operational Transients

15A.6.3.1 General

The safety requirements and protection sequences for anticipated operational transients are described in the following paragraphs for Events 7 through 29. The protection sequence block diagrams show the sequence of front-line safety systems. (Refer to Figures 15A.6-7 through 15A.6-29.) The auxiliaries for the front-line safety systems are indicated in the auxiliary diagrams (Figures 15A.6-1 and 15A.6-2) and the commonality of auxiliary diagrams (Figures 15A.6-46 through 15A.6-51).

15A.6.3.2 Required Safety Actions/Related Unacceptable Result

The following list relates safety actions for anticipated operational transients that mitigate or prevent the unacceptable safety results.

Safety Action	Related Unacceptable Result Criteria	Reason Action Required
Scram and/or RPT	2-2 2-3	To prevent fuel damage and to limit nuclear system pressure rise
Pressure relief	2-3	To prevent excessive nuclear system pressure rise
Core and Containment cooling	2-2	To prevent fuel and containment damage in the event that normal cooling is interrupted

Safety Action	Related Unacceptable Result Criteria	Reason Action Required
Reactor vessel isolation	2-2	To prevent fuel damage by reducing the outflow of steam and water from the reactor vessel, thereby limiting the decrease in reactor vessel water level
Restore ac power	2-2	To prevent fuel damage by restoring ac power to systems essential to other safety actions
Prohibit rod motion	2-2	To prevent exceeding fuel limits during transients
Containment isolation	2-4	To minimize radiological effects

15A.6.3.3 Event Definitions & Operational Safety Evaluations

Event 7 - Manual & Inadvertent SCRAM

The deliberate manual or inadvertent automatic scram due to single operator error is an event which can occur under any operating conditions. Although assumed to occur here for examination purposes, multi-operator error or action is necessary to initiate such an event.

While all the safety criteria apply, no unique safety actions are required to control the planned-operation-like event after effects of the subject initiation actions. In all operating states, the safety criteria are therefore met through the design basis of the plant systems. Figure 15A.6-7 identifies the protection sequences for this event.

Event 8 - Loss-of-Plant Instrument Air System

Loss of all plant instrument/service air system causes reactor shutdown and the closure of isolation valves. Although these actions occur, they are not a requirement to prevent unacceptable results in themselves. Multi-equipment failures would be necessary in order to cause the deterioration of the subject system to the point that the components supplied with instrument or service air would cease to operate "normally" and/or "fail-safe". The results in actions are identical to the Event 14 described later.

Isolation of the main steam lines can result in a transient for which some degree of protection is required only in operating States C and D. In operating States A and B, the main steam lines are continuously isolated.

Isolation of all main steam lines is most severe and rapid in operating State D during power operation.

Figure 15A.6-8 shows how scram is accomplished by main steam line isolation through the actions of the reactor protection system and the control rod drive system. The nuclear system pressure relief system provides pressure relief. Pressure relief, combined with loss of feedwater flow, causes reactor vessel water level to fall. Either high-pressure core cooling system supplies water to maintain water level and to protect the core until normal steam flow (or other planned operation) is established.

Adequate reserve instrument air supplies are maintained exclusively for the continual operation of the safety/relief valves until reactor shutdown is accomplished.

Event 9 - Inadvertent HPCS Pump (or any NSSS Pump) Start (Moderator Temperature Decrease)

An inadvertent pump start (temperature decrease) is defined as an unintentional start of any nuclear system pump that adds sufficient cold water to the reactor coolant inventory to cause a measurable decrease in moderator temperature. This event is considered in all operating states because it can potentially occur under any operating condition. Since the HPCS pump operates over nearly the entire range of the operating states and delivers by far the greatest amount of cold water to the vessel, the following analysis will describe its inadvertent operation rather than other NSSS pumps (e.g., RCIC, RHR, LPCS).

While all the safety criteria apply, no unique safety actions are required to control the adverse effects of such a pump start (i.e., pressure increase and temperature decrease in States A and C). In these operating states, the safety criteria are met through the basic design of the plant systems, and no safety action is specified. In States B and D, where the reactor is not shutdown, the operator or the plant normal control system can control any power changes in the normal manner of power control.

Figure 15A.6-9 illustrates the protection sequence for the subject event. Single failures to the normal plant control system pressure regulator or the feedwater controller systems will result in further protection sequences. These are shown in Events 22 and 23. The single failure (SF) aspects of their protection sequences will, of course, not be required.

Event 10 - Startup of Idle Recirculation Pump

The cold-loop startup of an idle recirculation pump can occur in any state and is most severe and rapid for those operating states in which the reactor may be critical (States B and D). When the transient occurs in the range of 10 to 60 percent power operation, no safety action response is required. Reactor power is normally limited to approximately 60 percent design power because of core flow limitations while operating with one recirculation loop working. Above about 60 percent power, a high neutron flux scram is initiated. Figure 15A.6-10 shows the protective sequence for this event.

Event 11 - Recirculation Flow Control Failure (Increasing Flow)

A recirculation flow control failure causing increased flow is applicable in States C and D. In State D, the accompanying increase in power level is accommodated through a reactor scram. As shown in Figure 15A.6-11, the scram safety action is accomplished through the combined actions of the neutron monitoring, reactor protection, and control rod drive systems.

Event 12 - Recirculation Flow Control Failure (Decreasing Flow)

This recirculation flow control malfunction causes a decrease in core coolant flow. This event is not applicable to States A and B because the reactor vessel head is off and the recirculation pumps normally would not be in use.

For the M/G set flow control mode, failures of one or the master flow controller will result in a transient equivalent to one or two recirculation pump trips, respectively; it is shown on Figure 15A.6-12. For the flow control valve control mode the fast closure of one or two control valves results in the protective sequence of Figure 15A.6-12.

Event 13 - Trip of One or Both Recirculation Pumps

The trip of one recirculation pump produces a milder transient than does the simultaneous trip of two recirculation pumps.

The transient resulting from this two-loop trip is not severe enough to require any unique safety action. The transient is compensated for by the inherent nuclear stability of the reactor. This event is not applicable in States A and B because the reactor vessel head is off and the recirculation pumps normally would not be in use. The trip could occur in States C and D; however, the reactor can accommodate the transient with no unique safety action requirement. Figure 15A.6-13 provides the protection sequence for the event for one or both pump trip actuations.

In fact, this event now constitutes an acceptable operational technique to reduce or minimize the effects of other event conditions. To this end, an engineered recirculation pump trip capability is included in the plant operational design to reduce pressure and thermo-hydraulic transient effects. Operating States C and D are involved in this event.

Tripping a single recirculation pump requires no protection system operation.

A two pump trip results in a high water level trip of the main turbine which further causes a stop valve closure and its subsequent scram actuation. Main steam lines isolation soon occurs and is followed by RCIC/HPCS systems initiation on low water level. Soon relief valve actuation will follow.

Event 14 - Isolation of One or All Main Steam Lines

Isolation of the main steam lines can result in a transient for which some degree of protection is required only in operating States C and D. In operating States A and B, the main steam lines are continuously isolated.

Isolation of all main steam lines is most severe and rapid in operating State D during power operation.

Figure 15A.6-14a shows how scram is accomplished by main steam line isolation through the actions of the reactor protection system and the control rod drive system. The nuclear system pressure relief system provides relief. Pressure relief, combined with loss of feedwater flow, causes reactor vessel water level to fall. Either the HPCS or RCIC System supplies water to maintain water level and to protect the core until normal steam flow (or other planned operation) is established.

Isolation of one main steam line causes a significant transient only in State D during high power operation. Scram is the only unique action required to avoid fuel damage and nuclear system overpressure. Because the feedwater system and main condenser remain in operation following the event, no unique requirement arises for core cooling.

As shown in Figure 15A.6-14b, the scram safety action is accomplished through the combined actions of the neutron monitoring, reactor protection, and control rod drive systems.

Event 15 - Inadvertent Opening of the Safety/Relief Valve

The inadvertent opening of a safety/relief valve is possible in any operating state. The protection sequences are shown in Figure 15A.6-15. In States A, B, and C, the water level cannot be lowered far enough to threaten fuel damage; therefore, no safety actions are required.

In State D, there is a slight decrease in reactor pressure following the event. The pressure controller closes the main turbine control valves enough to stabilize pressure at a level slightly below the initial value. There are no unique safety system requirements for this event.

If the event occurs when the feedwater system is not active in State D, a loss in the coolant inventory results in a reactor vessel isolation. The low water level signal initiates reactor vessel isolation. The nuclear system pressure relief system provides pressure relief.

Core cooling is accomplished by the RCIC/HPCS system which is automatically initiated by the incident detection circuitry (IDC). The automatic depressurization system (ADS) or the manual relief valve system remain as the backup depressurization system if needed. After the vessel has depressurized, long term core cooling is accomplished by the LPCI, LPCS, or HPCS, all of which are initiated on low water level by the IDC system or are manually operated. Containment/suppression pool cooling is manually initiated.

Event 16 - Control Rod Withdrawal Error (During Refueling & Startup Operation)

Because a control rod withdrawal error resulting in an increase of positive reactivity can occur under any operating condition, it must be considered in all operating states. For this specific event situation, only State A and B apply.

Refueling

No unique safety action is required in operating State A for the withdrawal of one control rod because the core is more than one control rod subcritical. Withdrawal of more than one control rod is precluded by the protection sequence shown in Figure 15A.6-16. During core alterations, the mode switch is normally in the REFUEL position, which allows the refueling equipment to be positioned over the core and also inhibits control rod withdrawal. This transient, therefore, applies only to operating State A.

No safety action is required because the total worth (positive reactivity) of one fuel assembly or control rod is not adequate to cause criticality. Moreover, mechanical design of the control rod assembly prevents physical removal without removing the adjacent fuel assemblies.

Startup

During low power operation (State B), the neutron monitoring system via the RPS will initiate scram if necessary. Refer to Figure 15A.6-16.

Event 17 - Control Rod Withdrawal Error (During Power Operation)

Because a control rod withdrawal error resulting in an increase of positive reactivity can occur under any operating condition, it must be considered in all operating states. For this specific event situation, only States C and D apply.

During power operation (Power Range) (State D), a number of plant protective devices of various designs prohibit the control rod motion before critical levels are reached. Refer to Figure 15A.6-17. While in State C no protective action is needed.

Systems in the power range (0 to 100 percent NBR) first of all prevent the selection of an out-of-sequenced rod movement for RWM and RSCS (Banked Position or Notch Group), or the RPCS provides out-of-sequenced rod selection. Secondly, the movement of the rod is monitored and limited within acceptable intervals (either/or) by neutronic effects or actual rod motion, (notch counting). RCIS provides movement surveillance. Of course, always beyond these rod motion control limits are the fuel/core scram protection systems. While in State C no protective action is needed.

Event 18 - Loss of Shutdown Cooling

The loss of RHR-shutdown cooling can occur only during the low pressure portion of a normal reactor shutdown and cooldown.

As shown in Figure 15A.6-18, for most single failures that could result in primary loss of shutdown cooling capabilities, no unique safety actions are required; in these cases, shutdown cooling is simply reestablished using redundant shutdown cooling equipment. In the cases where the RHR-shutdown cooling suction line becomes inoperative, a unique arrangement for cooling arises. In States A and B, in which the reactor vessel head is off, the LPCI, LPCS or HPCS can be used to maintain reactor vessel water level. In States C and D, in which the reactor vessel head is on and the system can be pressurized, the automatic depressurization system (ADS) or manual operation of relief valves in conjunction with any of the ECCS and the RHR suppression pool cooling mode (both manually operated) can be used to maintain water level and remove decay heat. Containment/Suppression pool cooling is actuated. Core and containment decay heat are removed by the RHR containment cooling system. The alternate shutdown cooling mode discussed in 5.4.7.1.5 can also be used.

Event 19 - RHR Shutdown Cooling Malfunction (Moderator Temperature Decrease)

An RHR shutdown cooling malfunction causing a moderator temperature decrease must be considered in all operating states. However, this event is not considered in States C and D if nuclear system pressure is too high to permit operation of the shutdown cooling (RHR). Refer to Figure 15A.6-19. No unique safety actions are required to avoid the unacceptable safety results for transients as a result of a reactor coolant temperature decrease induced by misoperation of the shutdown cooling heat exchangers.

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

Event 20 - Loss of All Feedwater Flow

A loss of feedwater flow results in a net decrease in the coolant inventory available for core cooling. A loss of feedwater flow can occur in States C and D. Appropriate responses to this transient include a reactor scram on low water level and maintenance of reactor vessel water level.

As shown in Figure 15A.6-20, the reactor protection and control rod drive systems effect a scram on low water level. The containment and reactor vessel isolation control system (CRVICS) and the main steam line isolation valves act to isolate the reactor vessel. After the main steam isolation valves close, decay heat slowly raises system pressure to the lowest relief valve setting. Pressure is relieved by the nuclear system pressure relief system. Initial core cooling is necessary to restore and maintain water level. Either the RCIC/HPCS system can maintain adequate water level. For long term shutdown and extended core coolings, containment/suppression pool cooling systems are manually initiated.

The requirements for operating State C is the same as for State D.

Event 21 - Loss of a Feedwater Heater

Loss of a feedwater heater must be considered with regard to the nuclear safety operational criteria only in operating State D because significant feedwater heating does not occur in any other operating state.

A loss of feedwater heating causes a transient that requires no protective actions when the reactor is initially on automatic recirculation flow control. If the reactor is on manual flow control, however, the neutron flux increase associated with this event will reach the scram setting. As shown in Figure 15A.6-21, the scram safety action is accomplished through actions of the neutron monitoring, reactor protection, and control rod drive systems. Water level will initiate a turbine trip and isolation will soon follow.

Event 22 - Feedwater Controller Failure - Maximum Demand

A feedwater controller failure, causing an excess of coolant inventory in the reactor vessel, is possible in all operating states. Feedwater controller failures considered are those that would give failures of automatic flow control, manual flow control, or feedwater bypass valve control. In operating States A and B, no safety actions are required since the vessel head is removed and the moderator temperature is low. In operating State D, any adverse responses by the reactor caused by cooling of the moderator can be mitigated by a scram. As shown in Figure 15A.6-22, the accomplishment of the scram safety action is satisfied through the combined actions of the neutron monitoring, reactor protection, and control rod drive systems. Pressure relief is required in States C and D and is achieved through the operation of the nuclear system pressure relief system. Initial restoration of the core water level is by the RCIC/HPCS systems. Prolonged isolation may require extended core cooling and containment/ suppression pool cooling.

Event 23 - Pressure Controller Failure (Open Direction)

A pressure controller failure in the open direction, causing the opening of a turbine control or bypass valve applies only in operating States C and D, because in other states the pressure controller is not in operation. A pressure controller failure is most severe and rapid in operating State D at low power.

The various protection sequences giving the safety actions are shown in Figure 15A.6-23. Depending on plant conditions existing prior to the event, scram will be initiated either on main steam line isolation, main turbine trip, reactor vessel high pressure, or reactor vessel low water level. The sequence resulting in reactor vessel isolation also depends on initial conditions. With the mode switch in "Run," isolation is initiated when main steam line pressure decreases to approximately 800 psig. After isolation is completed, decay heat will cause reactor vessel pressure to increase until limited by the operation of the relief valves. Core cooling following isolation can be provided by either the RCIC or HPCS. Shortly after reactor vessel isolation, normal core cooling can be reestablished via the main condenser and feedwater systems or if prolonged isolation is necessary, extended core and containment cooling will be manually actuated.

Event 24A - One Pressure Controller Channel Failure - Closed

A pressure control failure in the closed direction (or downscale), causing the closing of turbine control valves, applies only in operating States C and D, because in other states the pressure controller is not in operation.

A single pressure controller channel failure downscale would result in little or no effect on the plant operation. The other two pressure control channels would provide turbine-reactor control. The total failure of the pressure controller is reported in Event 24B.

The various protection sequences giving the safety actions are shown in Figure 15A.6-24. Upon failure of one pressure controller channel downscale, normally the backup channels will maintain the plant in the present status upon the initial channel downscale failure.

Event 25 - Main Turbine Trips (With Bypass System Operation)

A main turbine trip can occur only in operating State D (during heatup or power operation). A turbine trip during heatup is not as severe as a trip at full power because the initial power level is low (35.4 percent), thus minimizing the effects of the transient and enabling return to planned operations via the by-pass system operation. For a turbine trip above 35.4 percent power, a scram will occur via turbine stop valve closure as will a recirculation pump trip (RPT). Subsequent relief valve actuation will occur. Eventual main steam line isolation and RCIC/HPCS system initiation will result from low water level. Figure 15A.6-25 depicts the protection sequences required for main turbine trips. Main turbine trip and main generator trip are similar anticipated operational transients and, although main turbine trip is a more severe transient than main generator trip due to the rapid closure of the turbine stop valves, the required safety actions are the same.

Event 26 - Loss of Main Condenser Vacuum (Turbine Trip)

A loss of vacuum in the main turbine condenser can occur any time steam pressure is available and the condenser is in use; it is applicable to operating States C and D. This nuclear system pressure increase transient is the most severe of the pressure increase transients. However, scram protection in State C is not needed since the reactor is not coupled to the turbine system.

For State D above 35.4 percent power, loss of condenser vacuum will initiate a turbine trip with its attendant stop valve closures (which leads to scram) and a recirculation pump trip (RPT). Loss of condenser vacuum will also initiate isolation, pressure relief valve actuation, and RCIC/HPCS initial core cooling. A scram is initiated by MSIV closure to prevent fuel damage and is accomplished with the actions of the reactor protection system and control rod drive system. Below 35.4 percent power (State D) scram is initiated by a high neutron flux signal. Figure 15A.6-26 shows the protection sequences. Decay heat will necessitate extended core and containment cooling. When the nuclear system depressurizes sufficiently, the low pressure core cooling systems provide core cooling until a planned operation via RHR shutdown cooling is achieved.

Event 27 - Main Generator Trip (With Bypass System Operation)

A main generator trip with by-pass system operation can occur only in operating State D (during heatup or power operation). Fast closure of the main turbine control valves is initiated whenever an electrical grid disturbance occurs which results in significant loss of electrical load on the generator. The turbine control valves are required to close as rapidly as possible to prevent excessive overspeed of the main turbine - generator rotor. Closure of the turbine control valves will cause a sudden reduction in steam flow which results in an increase in system pressure. Above 35.4 percent power scram will occur as a result of fast control valve closure. Turbine tripping will actuate the Recirculation Pump Trip (RPT). Subsequently main steam line isolation will result, pressure relief and initial core cooling by RCIC/HPCS will take place. Prolonged shutdown of the turbine-generator unit will necessitate extended core and containment cooling. A generator trip during heatup (<35.4 percent) is not severe because the turbine by-pass system can accommodate the decoupling of the reactor and the turbine-generator unit, thus minimizing the effects of the transient and enabling return to planned operations. Figure 15A.6-27 depicts the protection sequences required for a main generator trip. Main generator trip and main turbine trip are similar anticipated operational transients. Although the main generator trip is a less severe transient than a turbine trip due to the rapid closure of the turbine stop valves, the required safety actions for both are the same sequence.

Event 28 - Loss of Normal Onsite Power - Service Transformer Failure

There is a variety of possible plant electrical component failures which could affect the reactor system. The total loss of onsite ac power is the most severe. The loss of a service transformer results in a sequence of events similar to that resulting from a loss of feedwater flow. The most severe situation occurs in State D during power operation. Figure 15A.6-28 shows normal onsite power in the States A, B, C, and D.

The reactor protection and control rod drive systems effect a scram on main turbine trip or loss of reactor protection system power sources. The turbine trip will actuate a recirculation pump trip (RPT). The containment and reactor vessel isolation control system CRVICS and the main steam line isolation valves act to isolate the reactor vessel. After the main steam line isolation valves (MSIV) close, decay heat slowly raises system pressure to the lowest relief valve setting. Pressure is relieved by the nuclear system pressure relief system. With continued isolation decay heat may cause increase in nuclear system pressure, eventually lifting relief valves and allowing reactor vessel water level to decrease. The core containment cooling sequences shown in Figure 15A.6-28 denote the short- and long-term actions for achieving adequate cooling.

Event 29 - Loss of Offsite Power - Grid Loss

There is a variety of possible plant-network electrical component failures which can affect reactor operation. The total loss of offsite ac power is the most severe. The loss of both house and offsite auxiliary power sources results in a sequence of events similar to that resulting from a loss of feedwater flow (see Event 20). The most severe case occurs in State D during power operation. Figure 15A.6-29 shows the safety actions required for a total loss of offsite power in all States A, B, C, and D.

The reactor protection and control rod drive systems affect a scram from main turbine trip or loss of reactor protection system power sources. The turbine trip will initiate recirculation pump trip (RPT). The containment and reactor vessel isolation control system (CRVICS) and the main steam line isolation valves (MSIV) act to isolate the reactor vessel. After the main steam line isolation valves close, decay heat slowly raises system pressure to the lowest relief valve setting. Pressure is relieved by the nuclear system pressure relief system. After the reactor is isolated and feedwater flow has been lost, decay heat will cause an increase in nuclear system pressure, eventually lifting relief valves and allowing reactor vessel water level to decrease. The core and containment cooling sequence shown in Figure 15A.6-29 shows the short- and long-term sequences for achieving adequate cooling.

15A.6.4 Abnormal (Unexpected) Operational Transients

15A.6.4.1 General

The safety requirements and protection sequences for abnormal operational transients are described in the following paragraphs for Events 30 through 34. The protection sequence block diagrams show the sequence of front-line safety systems (refer to Figures 15A.6-30 through 15A.6-34). The auxiliaries for the front-line safety systems are indicated in the auxiliary diagrams (Figures 15A.6-1 and 15A.6-2) and the commonality of auxiliary diagrams (Figures 15A.6-46 through 15A.6-51).

15A.6.4.2 Required Safety Actions/Related Unacceptable Results

The following list relates the safety actions for abnormal operational transients to mitigate or prevent the unacceptable safety results cited in Table 15A.2-8.

Safety Action	Related Unacceptable Result	Reason Action Required
Scram and/or RPT	3-2 3-3	To limit gross core-wide fuel damage and to limit nuclear system pressure rise
Pressure relief	3-3	To prevent excessive nuclear system pressure rise
Core and Containment cooling	3-2 3-4	To limit further fuel and containment cooling damage in the event that normal cooling is interrupted

Safety Action	Related Unacceptable Result	Reason Action Required
Reactor vessel isolation	3-2	To limit further fuel damage by reducing the outflow of steam and water from the reactor vessel, thereby limiting the decrease in reactor vessel water level
Restore ac power	3-2	To limit initial fuel damage by restoring ac power to systems essential to other safety actions
Containment isolation	3-4	To limit radiological effects

15A.6.4.3 Event Definition & Operational Safety Evaluation

Event 24B - Pressure Controller Downscale Failure

A pressure controller downscale failure causing the closure of all four turbine control valves applies in operating States C and D, because in other states, the pressure controller is not in operation. The various protection sequences giving the safety actions are shown in Figure 15A.6-24. This event will result in a reactor scram on high neutron flux or high pressure, system isolation, and subsequent extended isolation core cooling system actuations. Prolonged isolation will require core and containment cooling and possibly some radiological effluent control.

Event 30 - Main Generator Trip (Without Bypass System Operation)

A main generator trip without bypass system operation can occur only in operating State D (during heatup or power operation). A generator trip during heatup without by-pass operation results in the same situation as at power operation case. Figure 15A.6-30 depicts the protection sequences required for a main generator trip. The event is basically the same as that described in Event 27 at power levels above 35.4 percent full power. A scram, RPT, isolation, relief valve, and RCIC/HPCS operation will immediately result in prolonged shutdown, which will follow the same pattern as Event 27.

The load rejection and turbine trip are similar abnormal operational transients and, although main generator trip is a less severe transient than a turbine trip due to the rapid closure of the turbine stop valves, the required safety actions are the same.

Event 31 - Main Turbine Trip (Without Bypass System Operation)

A main turbine trip without bypass can occur only in operating State D (during heatup or power operation). Figure 15A.6-31 depicts the protection sequences required for main turbine trips. Plant operation with bypass system operation above or below 35.4 percent power, due to bypass system failure, will result in the same transient effects: a scram, a RPT, an isolation, subsequent relief valve actuation, and immediate RCIC/HPCS actuation.

After prolonged shutdown, similar extended core and containment cooling will be required as noted previously in Event 25.

Main turbine trip and load rejections are similar abnormal operational transients and, although main turbine trip is a more severe transient than main generator trip due to the rapid closure of the turbine stop valves, the required safety actions are the same.

Event 32 - Inadvertent Loading and Operation with Fuel Assembly in Improper Position

Operation with a fuel assembly in the improper position can occur in all operating states. No protection sequences are necessary relative to this event. Results of worst fuel bundle loading error will not cause fuel cladding integrity damage. It requires three independent equipment/operator errors to allow this situation to develop. See Figure 15A.6-32 for the event sequence.

Event 33 - Recirculation Loop Pump Seizure

A recirculation loop pump seizure event considers the instantaneous stoppage of the pump motor shaft of one recirculation loop pump. The case involving operation at design power in State D. A main turbine trip will occur as vessel level swell exceeds the turbine trip setpoint. This results in a trip scram and a RPT when the turbine stop valves close. Relief valve opening will occur to control pressure level and temperatures.

RCIC or HPCS systems will maintain vessel water level. Prolonged isolation will require core and containment cooling and possibly some radiological effluent control.

The protection sequence for this event is given in Figure 15A.6-33.

Event 34 - Recirculation Loop Pump Shaft Break

A recirculation loop pump shaft break event considers the degraded, delayed stoppage of the pump motor shaft of one recirculation loop pump. The case involving operation at design power in State D. A main turbine trip will occur as vessel level swell exceeds the turbine trip setpoint. This results in a trip scram and a RPT when the turbine stop valves close. Relief valve opening will occur to control pressure level and temperatures. RCIC or HPCS systems will maintain vessel water level. Prolonged isolation will require core and containment cooling and possibly some radiological effluent control.

The protection sequence for this event is given in Figure 15A.6-34.

15A.6.5 Design Basis (Postulated) Accidents

15A.6.5.1 General

The safety requirements and protection sequences for accidents are described in the following paragraphs for Events 35 through 44. The protection sequence block diagrams show the safety actions and the sequence of front-line safety systems used for the accidents (refer to Figures 15A.6-35 through 15A.6-42). The auxiliaries for the front-line safety systems are indicated in the auxiliary diagrams (Figures 15A.6-1 and 15A.6-2) and the commonality of auxiliary diagrams (Figures 15A.6-46 through 15A.6-51).

15A.6.5.2 Required Safety Actions/Unacceptable Results

The following list relates the safety actions for design basis accident to mitigate or prevent the unacceptable results cited in Table 15A.2-9.

Safety Action	Related Unacceptable Result	Reason Action Required
Scram	4-2	To prevent fuel cladding failure[1] and to prevent excessive nuclear system pressures
Pressure relief	4-3	To prevent excessive nuclear system pressure.
Core and Cooling	4-2	To prevent fuel cladding failure.
Reactor vessel isolation	4-1	To limit radiological effect to not exceed the guideline values of 10 CFR 50.67
Establish reactor containment	4-1	To limit radiological effects to not exceed the guideline values of 10 CFR 50.67.
Containment cooling	4-4	To prevent excessive pressure in the containment when containment is required.
Stop rod ejection	4-2	To prevent fuel cladding failure.
Restrict loss of reactor coolant (passive)	4-2	To prevent fuel cladding failure.
Control Room isolation	4-5	To prevent overexposure to radiation of plant personnel in the control room.
Limit reactivity	4-2 4-3	To prevent fuel cladding failure and to prevent excessive nuclear system pressure.

[1]Failure of the fuel barrier includes fuel cladding fragmentation (loss-of-coolant accident) and excessive fuel enthalpy (control rod drop accident).

15A.6.5.3 Event Definition and Operational Safety Evaluations

Event 35 - Control Rod Drop Accident (CRDA)

The control rod drop accident (CRDA) results from an assumed failure of the control rod-to-drive mechanism coupling after the control rod (very reactive rod) becomes stuck in its fully inserted position. It is assumed that the control rod drive is then fully withdrawn before the stuck rod falls out of the core. The control rod velocity limiter, an engineered safeguard, limits the control rod drop velocity. The resultant radioactive material release is maintained far below the guideline values of 10 CFR 50.67.

The control rod drop accident is applicable only in operating State D. The control rod drop accident cannot occur in State B because rod coupling integrity is checked on each rod to be withdrawn if more than one rod is to be withdrawn. No safety actions are required in States A or C where the plant is shutdown by more than one rod prior to the accident.

Figure 15A.6-35 presents the different protection sequences for the control rod drop accident. As shown in Figure 15A.6-35, the reactor is automatically scrammed and isolated. For all design basis cases, the neutron monitoring, reactor protection, and control rod drive systems will provide a scram from high neutron flux. The sequences in Figure 15A.6-35 assume that the main steam line radiation monitoring system will initiate the isolation of the reactor vessel and certain containment lines, or that the operator will manually do so. The analysis of Section 15.4.9 does not require MSIV isolation for compliance with radiological dose limits. Any high radiation in the containment areas will initiate closure of other possible pathways to atmosphere, as necessary.

After the reactor has been scrammed and isolated, the pressure relief system allows the steam (produced by decay heat) to be directed to the suppression pool. Initial core cooling is accomplished by either the RCIC or the HPCS or the normal feedwater system. With prolonged isolation, as indicated in Figure 15A.6-35, the reactor operator initiates the RHR/ suppression pool cooling mode and depressurizes the vessel with the automatic depressurization system (ADS) or via normal manual relief valve operation. The LPCI, LPCS, or HPCS maintain the vessel water level and accomplish extended core cooling. Isolation of turbine-condenser fission product releases will also be maintained.

Event 36 - Fuel Handling Accident (FHA)

Because a fuel-handling accident can potentially occur any time when fuel assemblies are being manipulated, either over the reactor core or in a spent fuel pool, this accident is considered in all operating states. Considerations include mechanical fuel damage caused by drop impact and a subsequent release of fission products. The protection sequences pertinent to this accident are shown in Figure 15A.6-36. Containment and/or auxiliary building isolation and standby gas treatment operation are automatically initiated by the respective ventilation radiation monitoring systems.

Figure 15A.6-36 describes the protection sequences for the event.

Event 37 - Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within RPCB Inside Containment (DBA-LOCA)

Pipe breaks inside the containment are considered only when the nuclear system is significantly pressurized (States C and D). The result is a release of steam and water into the containment. Consistent with NSOA criteria, the protection requirements consider all size line breaks including larger liquid recirculation loop piping down to small steam instrument line breaks. The most severe cases are the circumferential break of the largest (liquid) recirculation system pipe and the circumferential break of the largest (steam) main steam line.

As shown in Figure 15A.5-37, in operating State C (reactor shut down, but pressurized), a pipe break accident up to the DBA can be accommodated within the nuclear safety operational criteria through the various operations of the main steam line isolation valves, emergency core cooling systems (HPCS, automatic depressurization system (ADS), LPCI, and LPCS, containment and reactor vessel isolation control system, containment, auxiliary buildings, standby gas treatment system, control room atmospheric control and isolation system, MSIV-LCS, FWLC system, standby service water systems, combustible gas control system, suppression pool makeup system, equipment cooling systems, and the incident detection circuitry. For small pipe breaks inside the containment, pressure relief is effected by the nuclear system pressure relief system, which transfers decay heat to the suppression pool. For large breaks, depressurization takes place through the break itself. In State D (reactor not shut down, but pressurized), the same equipment is required as in State C but, in addition, the reactor protection system and the control rod drive system must operate to scram the reactor. The limiting items, on which the operation of the above equipment is based, are the allowable fuel cladding temperature and the containment pressure capability. The control rod drive housing supports are considered necessary whenever the system is pressurized to prevent excessive control rod movement through the bottom of the reactor pressure vessel following the postulated rupture of one control rod drive housing (a lesser case of the design basis loss-of-coolant accident and a related preventive of a postulated rod ejection accident).

After completion of the automatic action of the above equipment, manual operation of the RHR (suppression pool cooling mode) and ADS (controlled depressurization) is required to maintain containment pressure and fuel cladding temperature within limits during extended core cooling.

Events 38, 39, 40 - Large, Small, Steam, and Liquid Pipe Breaks Outside Containment (SLBA)

Pipe break accidents outside the containment are assumed to occur any time the nuclear system is pressurized (States C and D). This accident is most severe during operation at high power (State D). In State C, this accident becomes a lesser case of the State D sequence.

The protection sequences for the various possible pipe breaks outside the containment are shown in Figures 15A.6-38. The sequences also show that for small breaks (breaks not requiring immediate action) the reactor operator can use a large number of process indications to identify the break and isolate it.

In operating State D (reactor not shut down, but pressurized), scram is accomplished through operation of the reactor protection system and the control rod drive system. Reactor vessel isolation is accomplished through operation of the main steam line isolation valves and the containment and reactor vessel isolation control system.

For a main steam line break, initial core cooling is accomplished by either the HPCS or the automatic depressurization system (ADS) or manual relief valve operation in conjunction with either the LPCI or the LPCS. These systems provide two, three, or four parallel paths to effect initial core cooling, thereby satisfying the single failure criterion. Extended core cooling is accomplished by the single failure proof, parallel combination of LPCS,HPCS, and LPCI. The automatic depressurization system (ADS) or relief valve system operation and the RHR suppression pool cooling mode (both manually operated) are required to maintain containment pressure and fuel cladding temperature within limits during extended core cooling.

Event 41 - Gaseous Radwaste System Leak or Failure

It is assumed that the line leading to the steam jet air ejector fails near the main condenser. This results in activity normally processed by the offgas treatment system being discharged directly to the turbine building and subsequently through the ventilation system to the environment. This failure results in a loss-of-flow signal to the offgas system. This event can be considered only under States C and D.

The reactor operator initiates a normal shutdown of the reactor to reduce the gaseous activity being discharged. A loss of main condenser vacuum will result (timing depending on leak rate) in a main turbine trip and ultimately a reactor shutdown. Refer to Event 26 for reactor protection sequence (see Figure 15A.6-26).

The protective sequences for this event are provided in Figure 15A.6-39.

Event 42 - Augmented Offgas Treatment System Failure

An evaluation of those events which could cause a gross failure in the offgas system has resulted in the identification of a postulated seismic event, more severe than the one for which the system is designed, as the only conceivable event which could cause significant damage.

The detected gross failure of this system will result in manual isolation of this system from the main condenser. The isolation results in high main condenser pressure and ultimately a reactor scram.

The undetected postulated failure soon results in a system isolation necessitating reactor shutdown because of loss of vacuum in the main condenser. This transient has been analyzed in Event 26 (see Figure 15A.6-26).

The protective sequences for this event are provided in Figure 15A.6-40.

Event 43 - Liquid Radwaste System Leak or Failure

Releases which could occur inside and outside of the containment, not covered by Events 35 through 43 will probably include small spills and equipment leaks of radioactive materials inside structures housing the subject process equipment.

Conservative values for leakage have been assumed and evaluated in the plant under routine releases. The offsite dose that results from any small spill which could occur outside containment will be negligible in comparison to the dose resulting from the accountable (expected) plan leakages.

The protective sequences for this event are provided in Figure 15A.6-41.

[HISTORICAL INFORMATION] [Event 44 - Liquid Radwaste System - Storage Tank Failure (This Tank has been abandoned in place)

An unspecified event causes the complete release of the average radioactivity inventory in the subject tank containing the largest quantities of significant radionuclides from the liquid radwaste system. This is assumed to be one of the evaporator bottoms tanks in the radwaste building. The airborne radioactivity released during the accident passes directly to the environment via the radwaste building vent.

The postulated events that could cause release of the radioactive inventory of the evaporator bottoms tank include cracks in the vessels and an operator error. The possibility of small cracks and consequent low-level release rates receives primary consideration in system and component design. The evaporator bottoms tank is designed to operate at atmospheric pressure and 200 F maximum temperature so the possibility of failure is considered small. A liquid radwaste release caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instruction. A positive action interlock system is usually provided to prevent inadvertent opening of a drain valve. Should a release of liquid radioactive wastes occur, floor drain sump pumps in the floor of the radwaste building will receive a high water level alarm, activate automatically, and remove the spilled liquid to a contained storage tank.

The protective sequences for this event are provided in Figure 15A.6-42.]

15A.6.6 Special (Plant Capability) Events

15A.6.6.1 General

Additional special events are postulated to demonstrate that the plant is capable of accommodating off-design occurrences (Events 45 through 48). As such, these events are beyond the safety requirements of the other event categories. The safety actions shown on the sequence diagrams (refer to Figures 15A.6-43 through 15A.6-46) for the additional special events follow directly from the requirements cited in the demonstration of the plant capability.

Auxiliary system support analyses are shown in Figures 15A.6-1, 2, and 15A.6-46 through 15A.6-51.)

15A.6.6.2 Required Safety Action/Unacceptable Results

Safety Action	Related Unacceptable Result	Reason Action Required
Manually initiate all shutdown controls from local panels	5-1 5-2	Local panel control has been provided and is available outside control room. Reactor can be scrammed by opening the reactor scram breakers.
Manually initiate SLCS	5-3	Standby Liquid Control System to control reactivity to cold shutdown is available.

15A.6.6.3 Event Definitions and Operational Safety Evaluation

Event 45 - Spent Fuel Cask Drop

Due to the redundant design of the spent fuel cask handling crane, the cask drop accident is not considered a credible accident.

Event 46 - Reactor Shutdown - ATWS

Reactor shutdown from a plant transient occurrence (e.g., turbine trip) without the use of mechanical control rods is an event currently being evaluated to determine the capability of the plant to be safely shutdown. The event is applicable in any operating state. Figure 15A.6-43 shows the protection sequence for this extremely improbable and demanding event in each operating state. In State A, no sequence is shown because the reactor is already in the condition finally required by definition.

State D is the most limiting case. Upon initiation of the plant transient situation (turbine trip), a scram will be initiated but no control rods are assumed to move. The recirculation pumps will be tripped by the initial turbine trip signal. If the nuclear system becomes isolated from the main condenser, low power neutron heat can be transferred from the reactor to the suppression pool via the relief valves. The incident detection circuitry initiates operation of the HPCS on low water level which maintains reactor vessel water level. The standby liquid control system will be manually initiated and the transition from low power neutron heat to decay heat will occur. The RHR suppression pool cooling mode is used to remove the low power neutron and decay heat from the suppression pool as required. When reactor pressure falls to approximately 100 psig, the RHR shutdown cooling mode is started and continued to cold shutdown. Various single failure analytical exercises can be examined to further show additional capabilities to accommodate further plant system degradations. (Ref. 1 of Section 15A.9).

Event 47 - Reactor Shutdown From Outside Control Room

Reactor shutdown from outside 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 event is applicable in any operating States A, B, C, and D.

Figure 15A.6-44 shows the protection sequences for this event in each operating state. In State A, no sequence is shown because the reactor is already in the condition finally required for the event. In State C, only cooldown is required since the reactor is already shutdown.

A scram from outside the control room can be achieved by opening the ac supply breakers for the reactor protection system. If the nuclear system becomes isolated from the main condenser, decay heat is transferred from the reactor to the suppression pool via the relief valves. The incident detection circuitry initiates operation of the RCIC/HPCS systems on low water level which maintains reactor vessel water level, and the RHRS suppression pool cooling mode is used to remove the decay heat from the suppression pool if required. When reactor pressure falls to approximately 100 psig, the RHR shutdown cooling mode is started.

Event 48 - Reactor Shutdown Without Control Rods

Reactor shutdown, without control rods is an event requiring an alternate method of reactivity control (the standby liquid control system). By definition, this event can occur only when the reactor is not already shutdown. Therefore, this event is considered only in operating States B and D.

The standby liquid control system must operate to avoid unacceptable result criteria 5-3. The design bases for the standby liquid control system result from these operating criteria when applied under the most severe conditions (State D at rated power). As indicated in Figure 15A.6-45, the standby liquid control system is manually initiated and controlled in States B and D.

TABLE 15A.6-1: PLANT EVENTS APPLICABLE IN EACH BWR OPERATING STATE PLANNED (NORMAL) OPERATION

		BWR Operating States			
Types of Operation and Events		**A**	**B**	**C**	**D**
1.	Refueling outage	X			
2.	Achieving criticality	X	X	X	X
3.	Heatup				X
4.	Power operation				X
5.	Achieving shutdown		X		X
6.	Cooldown	X		X	

TABLE 15A.6-2: PLANT EVENTS APPLICABLE IN EACH BWR OPERATING STATE ANTICIPATED (EXPECTED) OPERATIONAL TRANSIENTS

Types of Operation and Events		BWR Operating States A	B	C	D
7.	Manual or Inadvertent SCRAM			X	X
8.	Loss of Plant Instrument Air System	X	X	X	X
9.	Inadvertent Startup of HPCS Pump	X	X	X	X
10	Startup of Idle Recirculation Loop Pump	X	X	X	X
11.	Recirculation Loop Flow Control Failure-Increasing			X	X
12.	Recirculation Loop Flow Control Failure-Decreasing			X	X
13.	Recirculation Loop Pump Trips - One or Both			X	X
14.	Inadvertent MSIV Closure - One or Four Valves			X	X
15.	Inadvertent Operation of One Safety/Relief Valve			X	X
16.	Continuous Control Rod Withdrawal Error - During Startup		X		
	- During Refueling	X			
17.	Continuous Control Rod Withdrawal Error - At Power			X	X
18.	RHR - Shutdown Cooling Failure - Loss of Cooling	X	X	X	X
19.	RHR - Shutdown Cooling Failure - Increased Cooling	X	X	X	X

TABLE 15A.6-2: PLANT EVENTS APPLICABLE IN EACH BWR OPERATING STATE ANTICIPATED (EXPECTED) OPERATIONAL TRANSIENTS (Continued)

Types of Operation and Events	BWR Operating States A	B	C	D
20. Loss of All Feedwater Flow			X	X
21. Loss of One Feedwater Heater				X
22. Feedwater Controller Failure - Maximum Demand	X	X	X	X
23. Pressure Control Failure - Open			X	X
24. Pressure Control Failure - Closed			X	X
25. Main Turbine Trips- With Bypass				X
26. Loss of Main Condenser Vacuum			X	X
27. Main Generator Trip (Load Rejection) - With Bypass				X
28. Loss of Plant Normal Onsite ac Power - Auxiliary Transformer Loss	X	X	X	X
29. Loss of Plant Normal Offsite ac Power - Grid Connection Loss	X	X	X	X

TABLE 15A.6-3: PLANT EVENTS APPLICABLE IN EACH BWR OPERATING STATE PLANNED (NORMAL) OPERATION

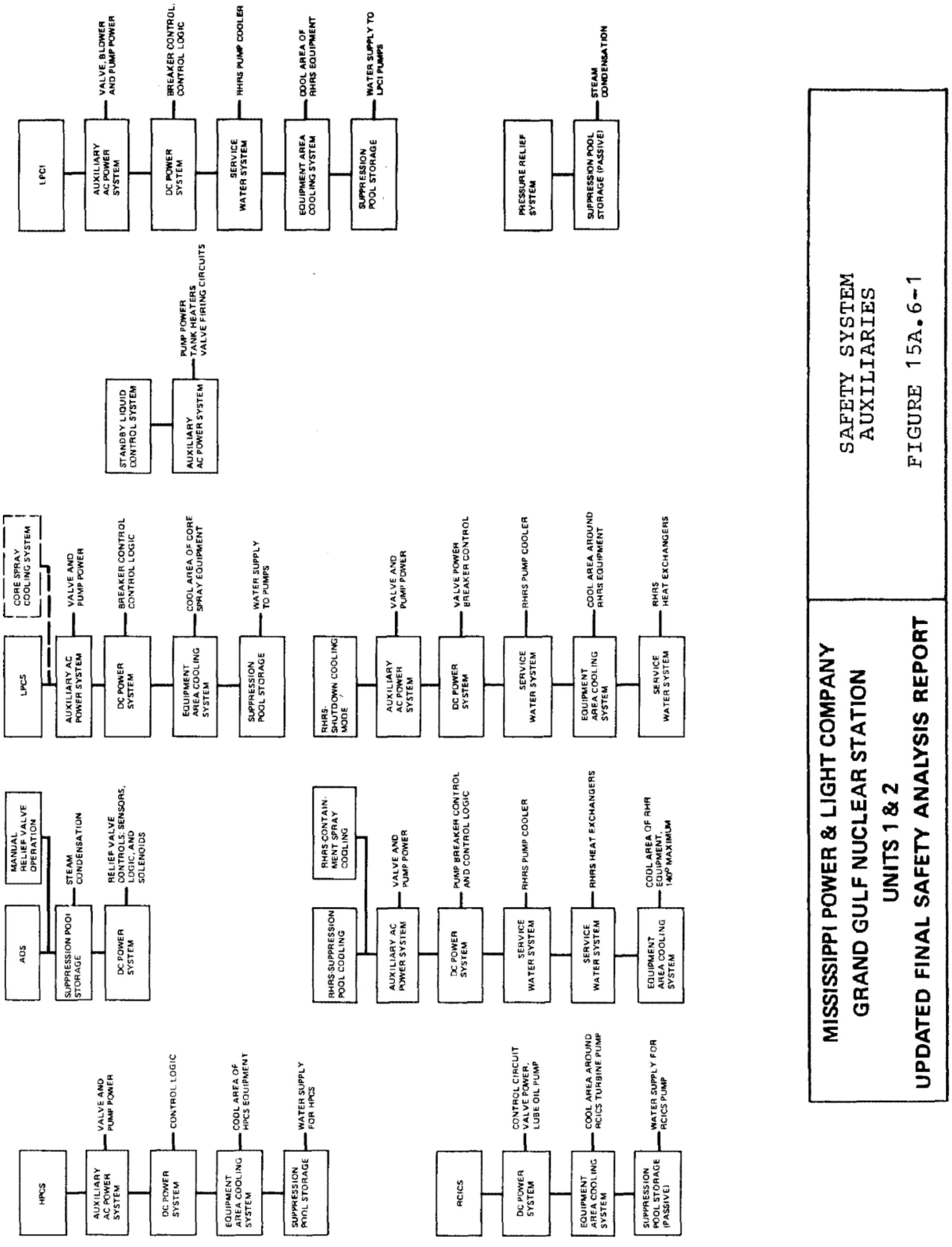
Types of Operation and Events	BWR Operating States A	B	C	D
30. Main Generator Trip (Load Rejection) - Without Bypass				X
31. Main Turbine Trip - Without Bypass				X
32. Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	X	X	X	X
33. Recirculation Loop Pump Seizure				X
34. Recirculation Loop Pump Shaft Break				X

TABLE 15A.6-4: PLANT EVENTS APPLICABLE IN EACH BWR OPERATING STATE DESIGN BASIS (POSTULATED) ACCIDENTS

		BWR Operating States			
Types of Operation and Events		**A**	**B**	**C**	**D**
35.	Control Rod Drop Accident			X	X
36.	Fueling Handling Accident	X	X	X	X
37.	Loss of Coolant Accident Resulting from Spectrum of Postulated Piping Breaks Within RPCB Inside Containment			X	X
38.	Steam System Piping Break Outside Containment			X	X
39.	Instrument Line Break Outside Containment			X	X
40.	Feedwater Line Break Outside Containment			X	X
41.	Gaseous Radwaste System Leak or Failure	X	X	X	X
42.	Augmented Offgas Treatment System Failure	X	X	X	X
43.	Liquid Radwaste System Leak or Failure	X	X	X	X
44.	Liquid Radwaste System Storage Tank Failure	X	X	X	X

TABLE 15A.6-5: PLANT EVENTS APPLICABLE IN EACH BWR OPERATING STATE SPECIAL (PLANT CAPABILITY) EVENTS

Types of Operation and Events	BWR Operating States A	B	C	D
45. Spent Fuel Cask Drop				
46. Reactor Shutdown from Anticipated Transient - Without Scram (ATWS)	X	X	X	X
47. Reactor Shutdown - From Outside Control Room	X	X	X	X
48. Reactor Shutdown - Without Control Rods	X	X	X	X


MISSISSIPPI POWER & LIGHT COMPANY
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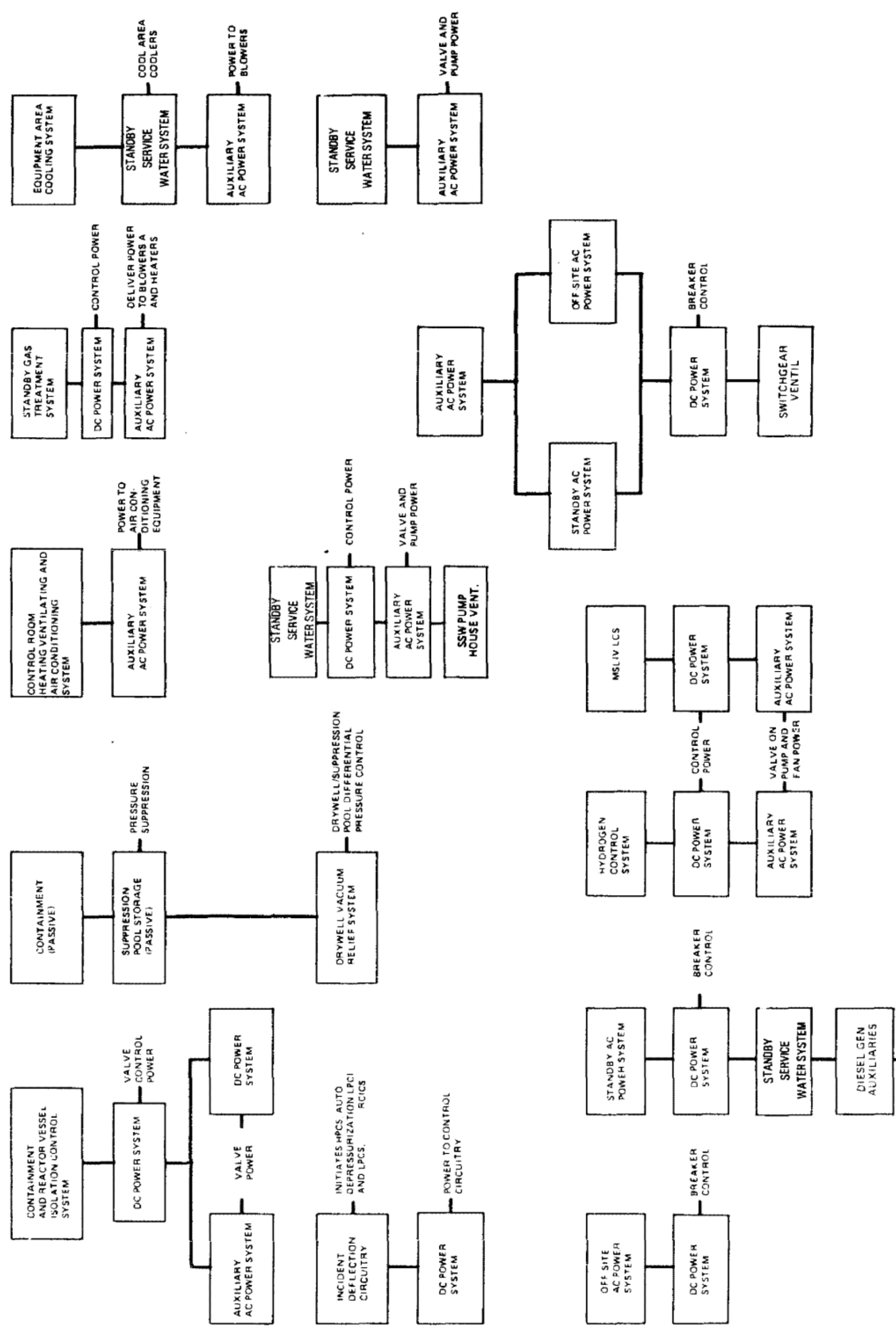
SAFETY SYSTEM
AUXILIARIES
FIGURE 15A.6-1



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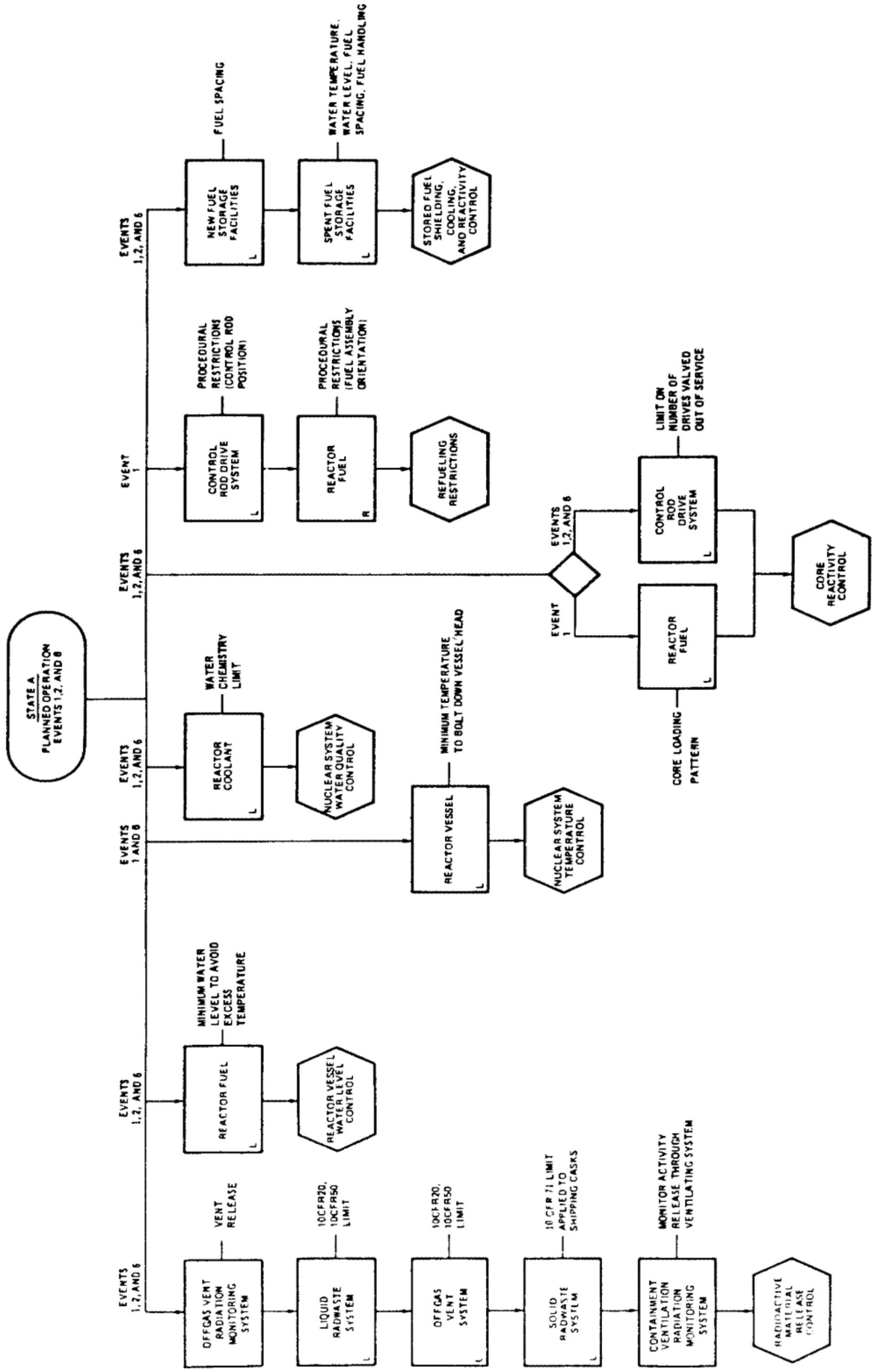
SAFETY SYSTEM AUXILIARIES

FIGURE 15A.6-2



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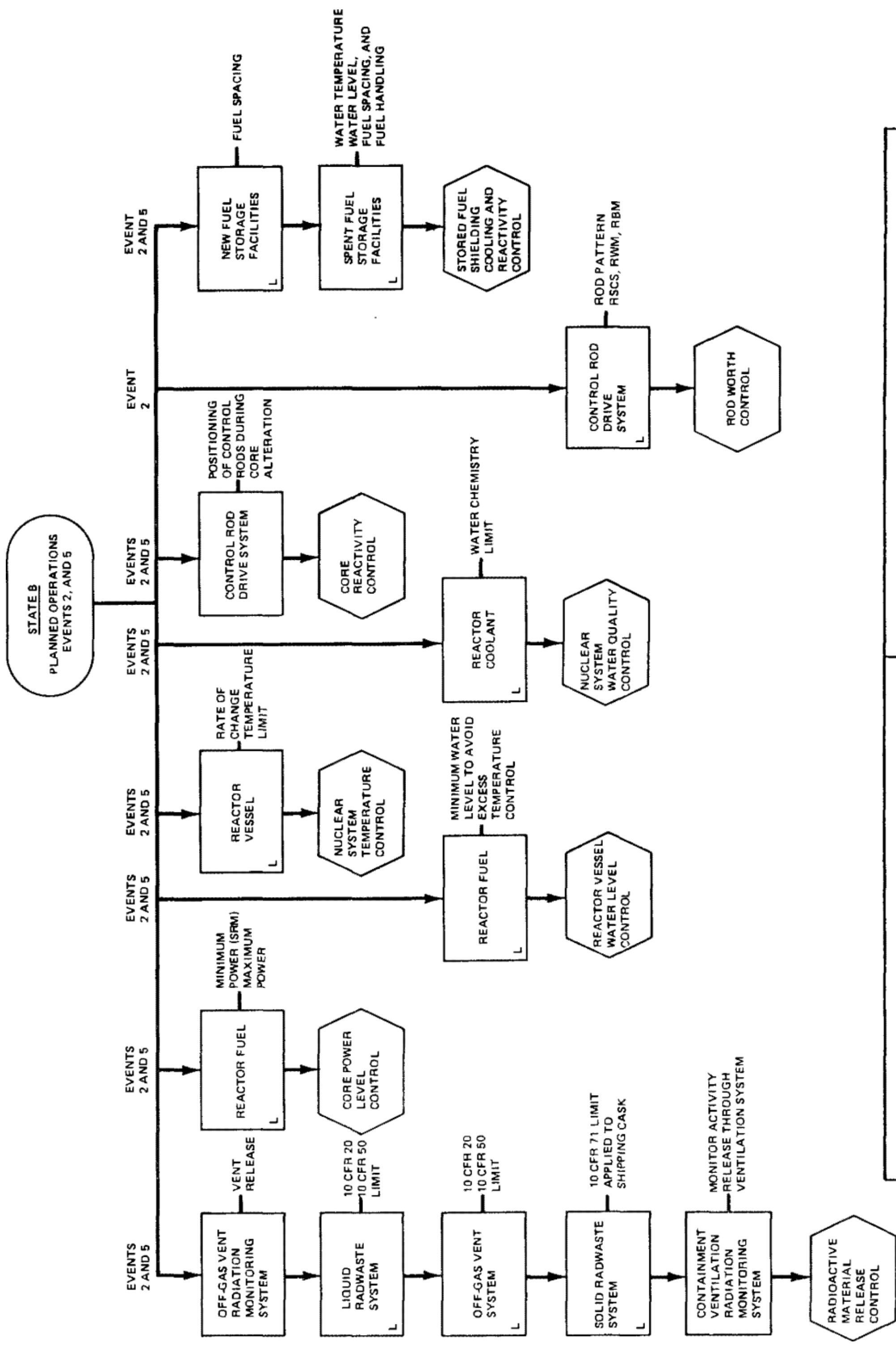
SAFETY ACTION SEQUENCES FOR PLANNED OPERATIONS IN STATE A
FIGURE 15A.6-3



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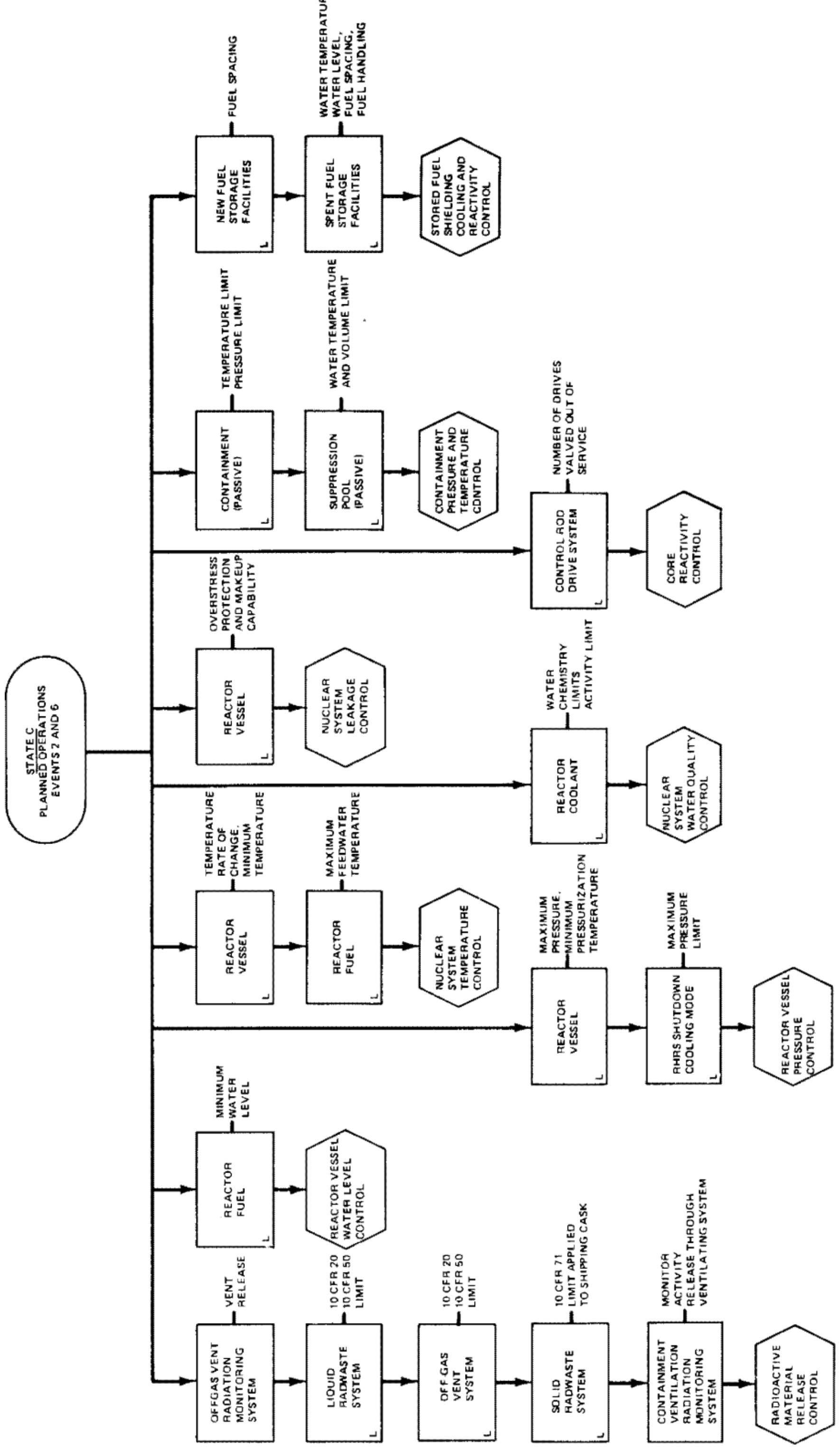
SAFETY ACTION SEQUENCES
FOR PLANNED OPERATIONS
IN STATE B
FIGURE 15A.6-4



SAFETY ACTION SEQUENCES
FOR PLANNED OPERATIONS
IN STATE C
FIGURE 15A.6-5

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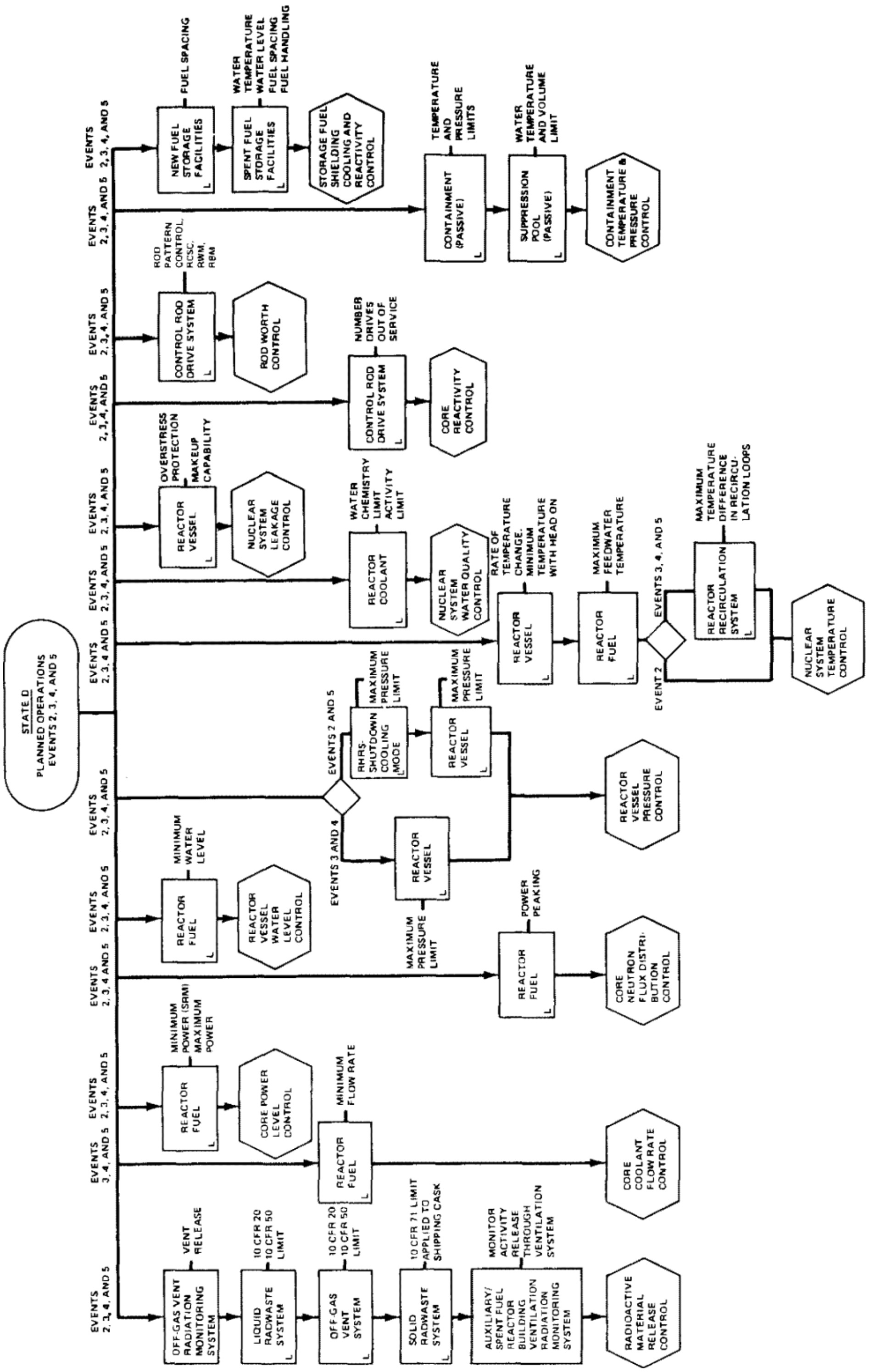

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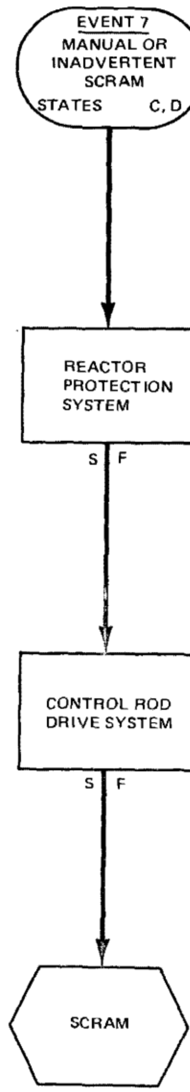
SAFETY ACTION SEQUENCES
FOR PLANNED OPERATIONS
IN STATE D
FIGURE 15A.6-6



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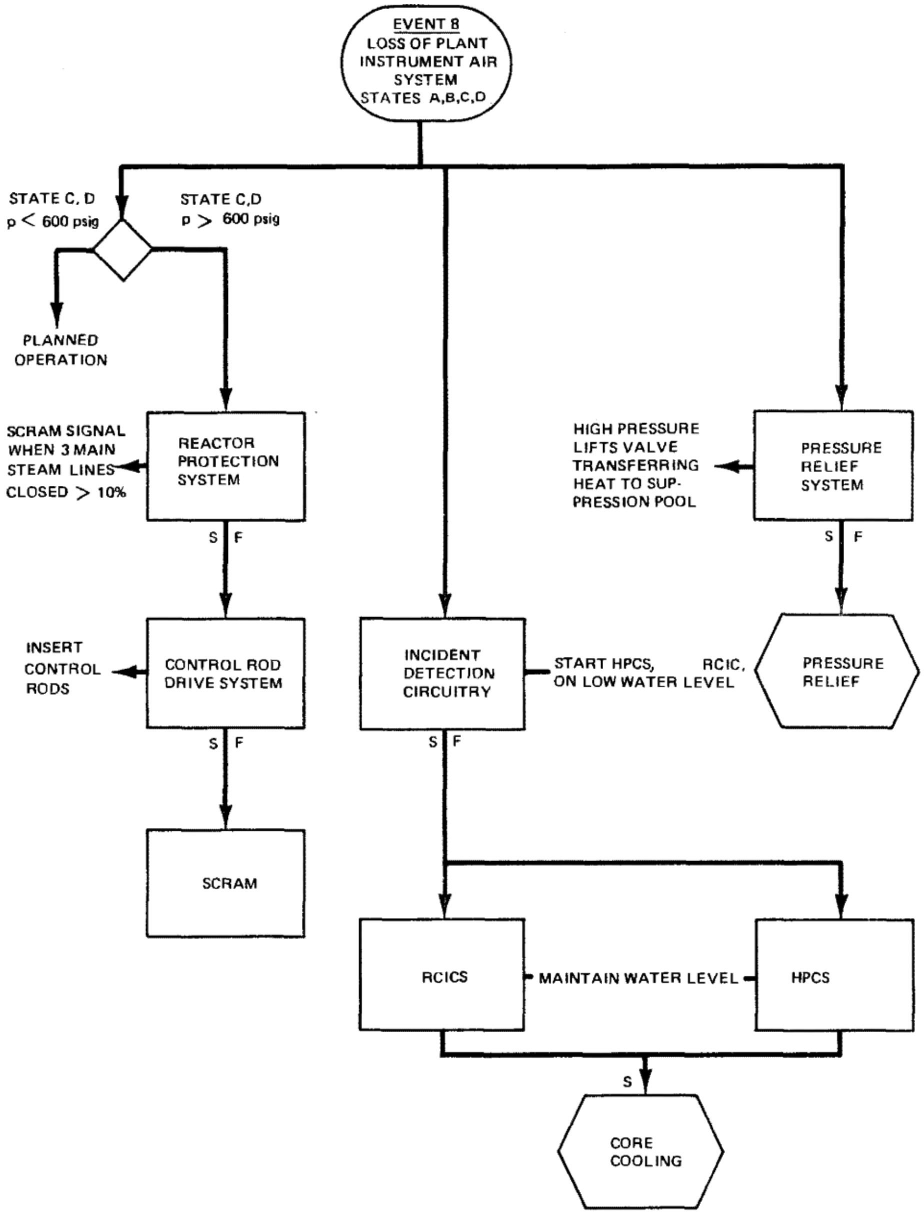
PROTECTION SEQUENCE FOR
MANUAL OR INADVERTENT SCRAM

FIGURE 15A.6-7



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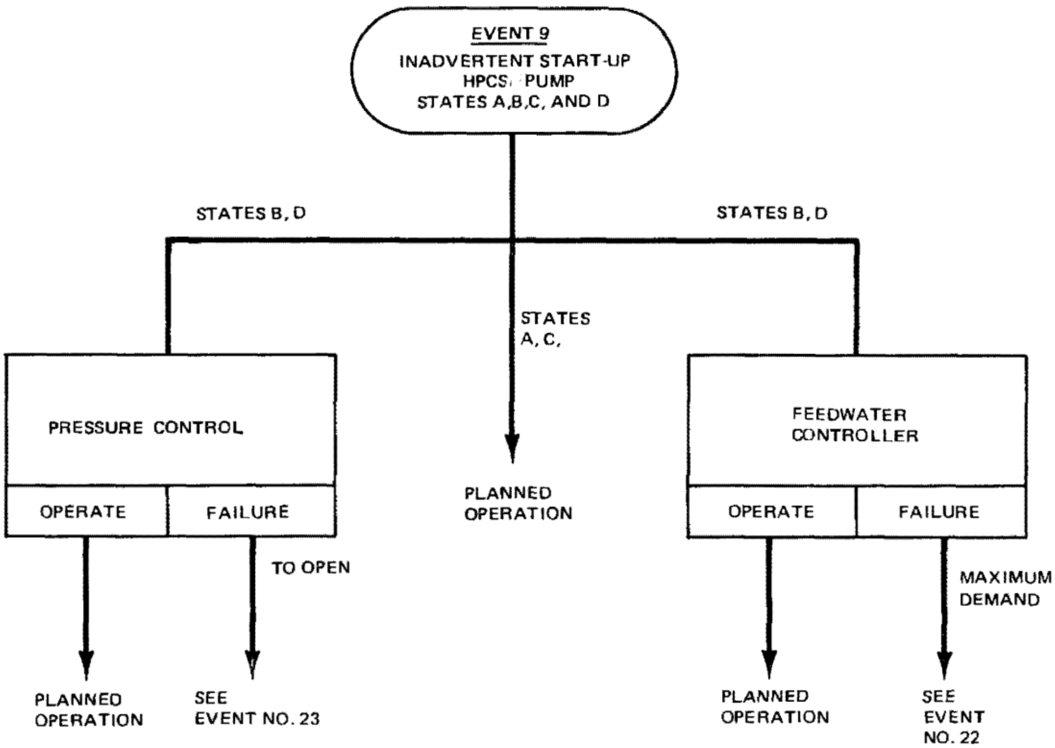
PROTECTION SEQUENCE FOR
LOSS OF PLANT INSTRUMENT/
SERVICE AIR SYSTEM
FIGURE 15A.6-8



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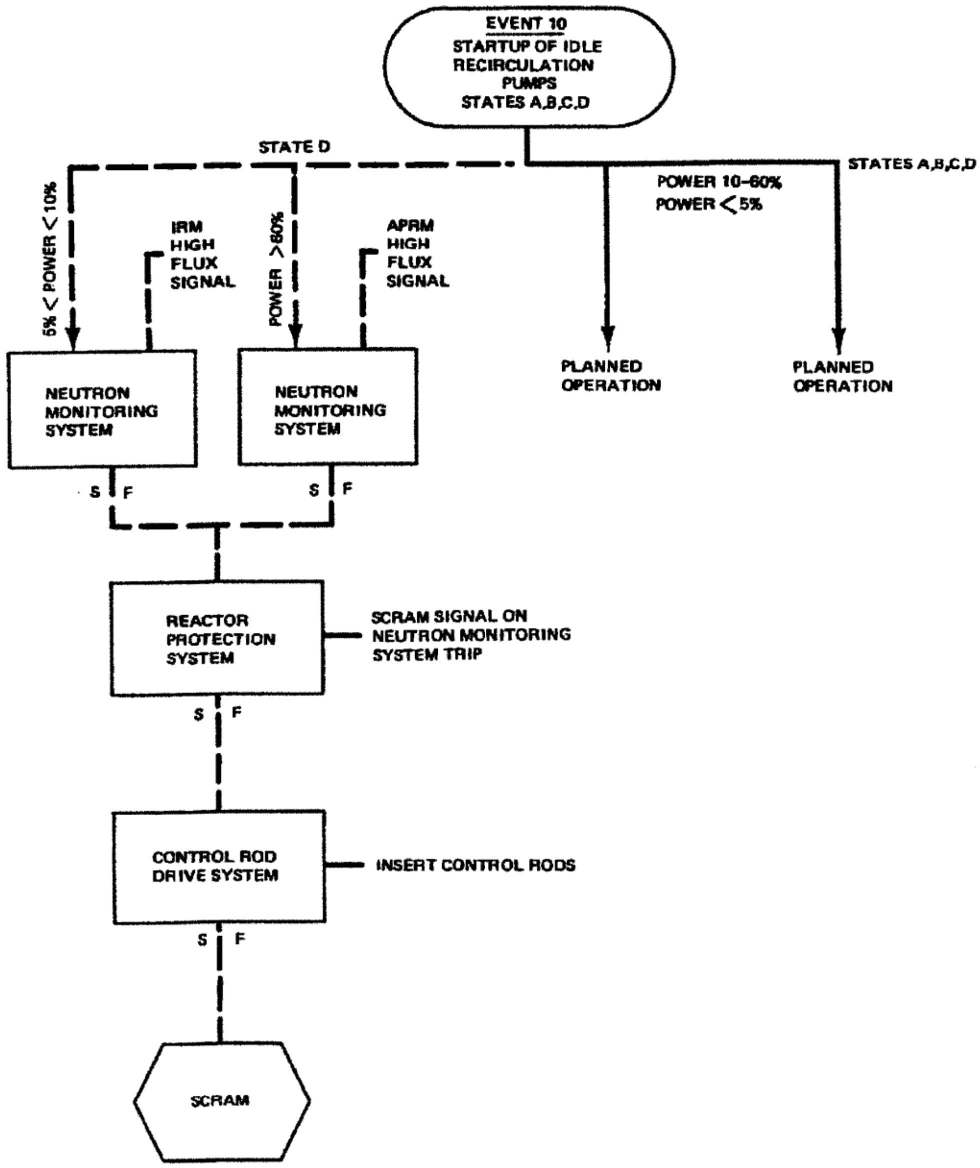
PROTECTION SEQUENCE FOR
INADVERTENT START-UP OF
HPCS PUMPS
FIGURE 15A.6-9



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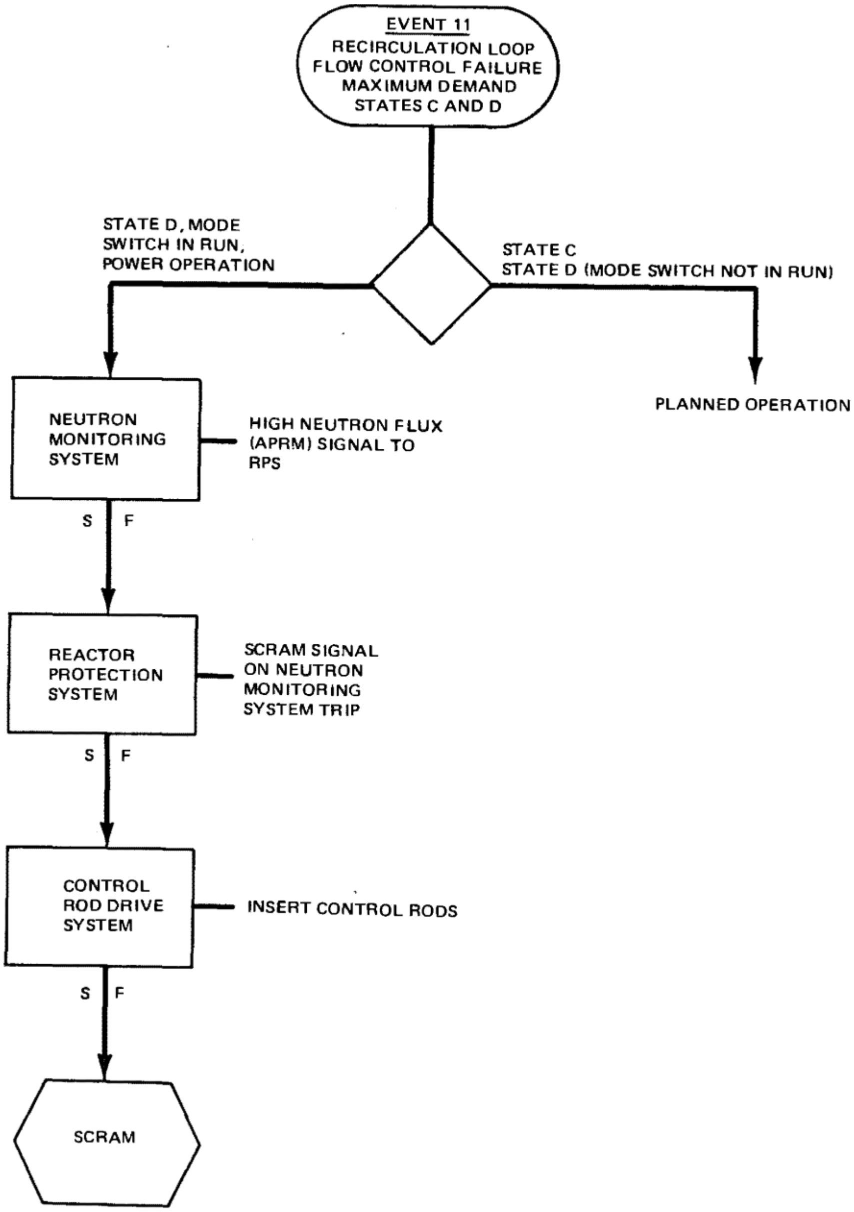
PROTECTION SEQUENCES FOR START-UP OF IDLE RECIRCULATION LOOP PUMP FIGURE 15A.6-10



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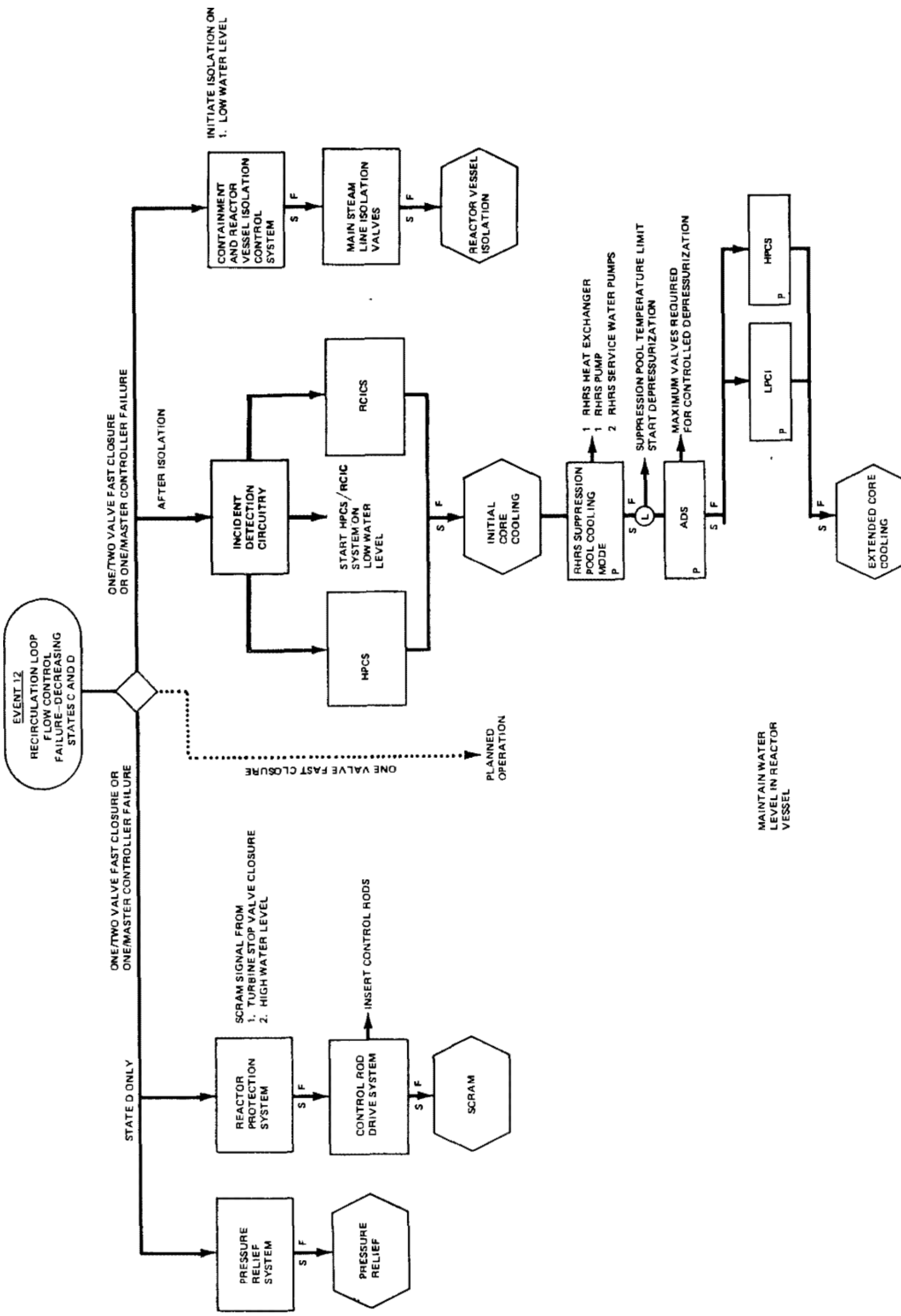
PROTECTION SEQUENCE FOR
RECIRCULATION LOOP FLOW
CONTROL FAILURE-MAXIMUM DEMAND
FIGURE 15A.6-11



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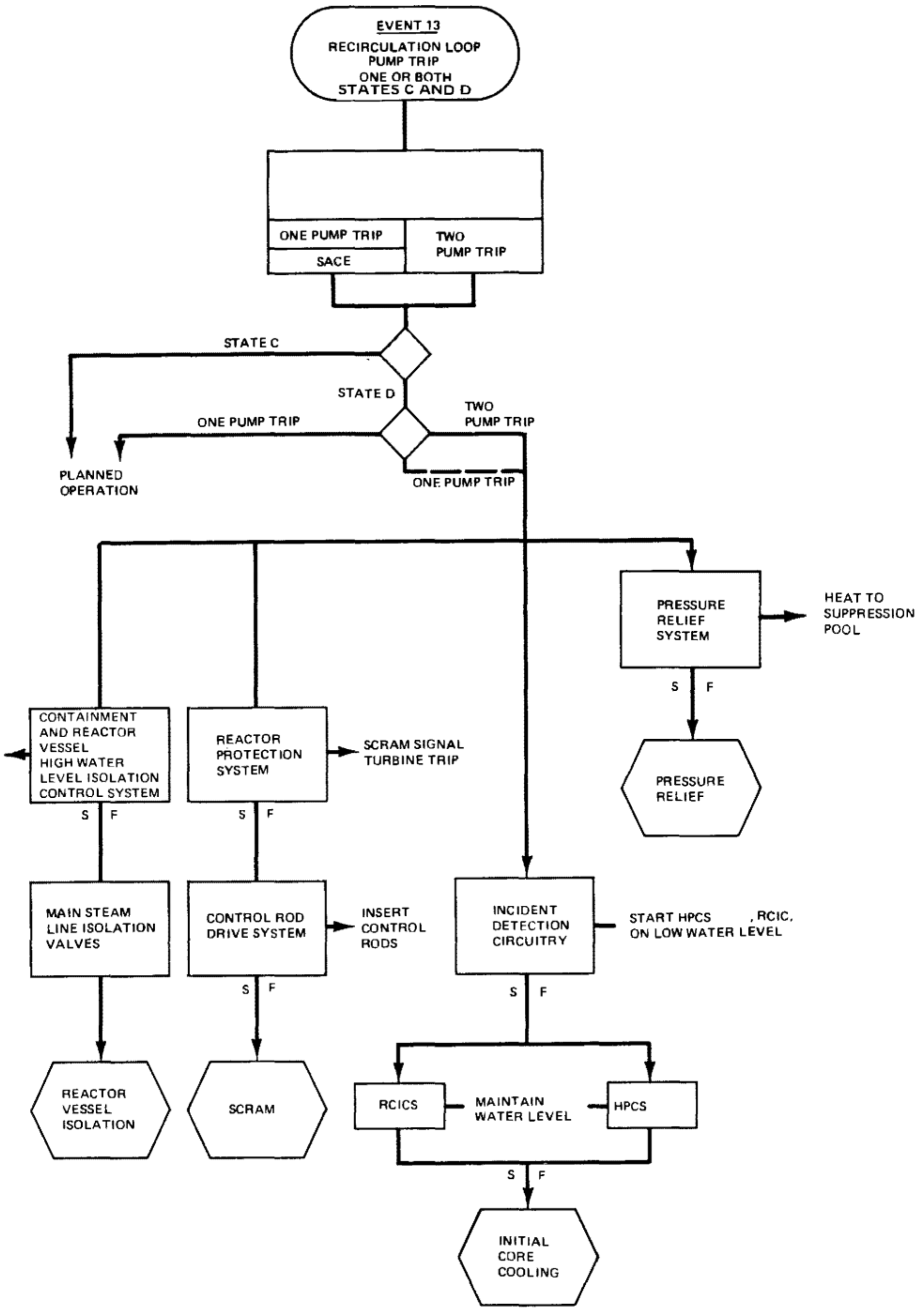
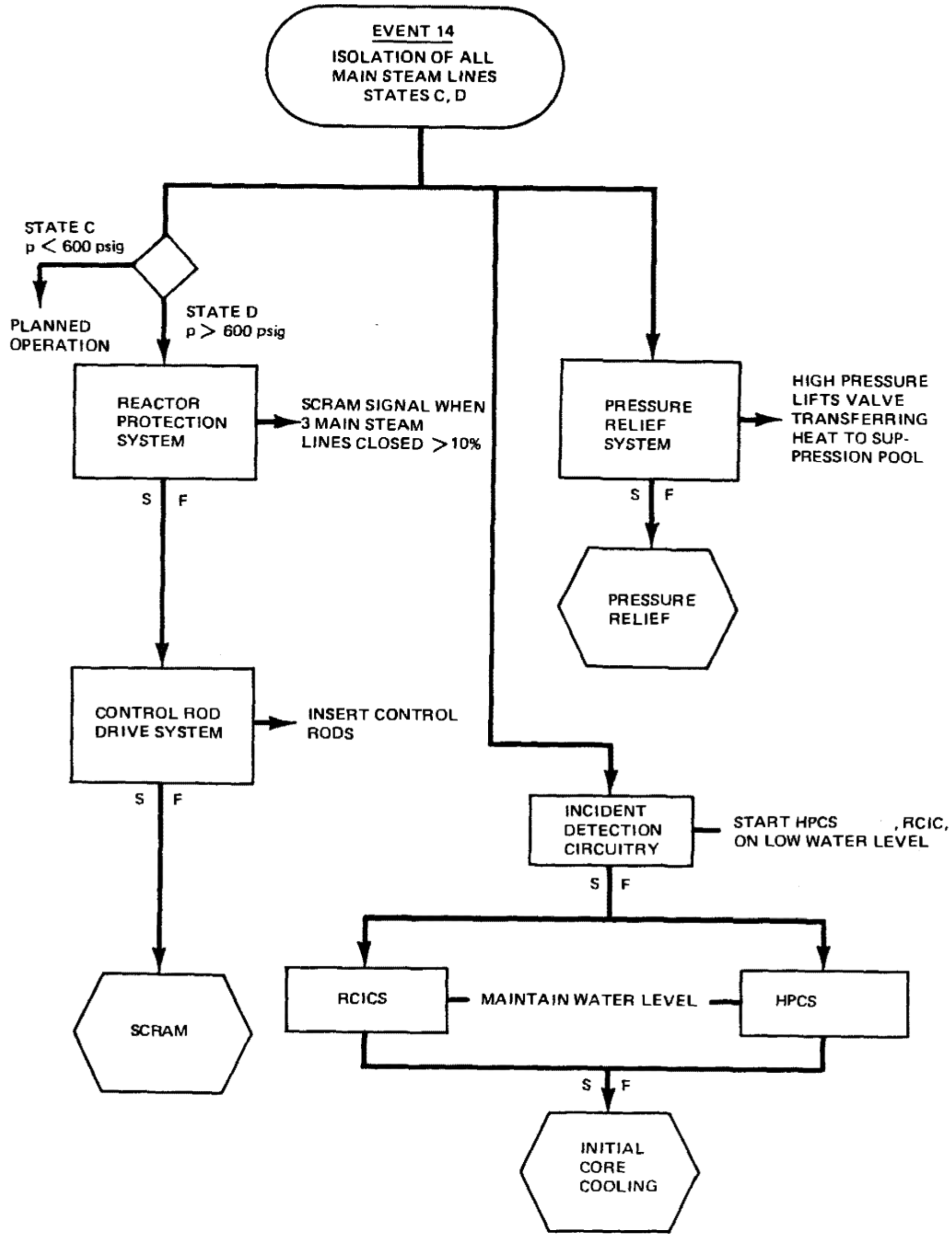
PROTECTION SEQUENCE FOR RECIRCULATION LOOP FLOW CONTROL FAILURE-DECREASING

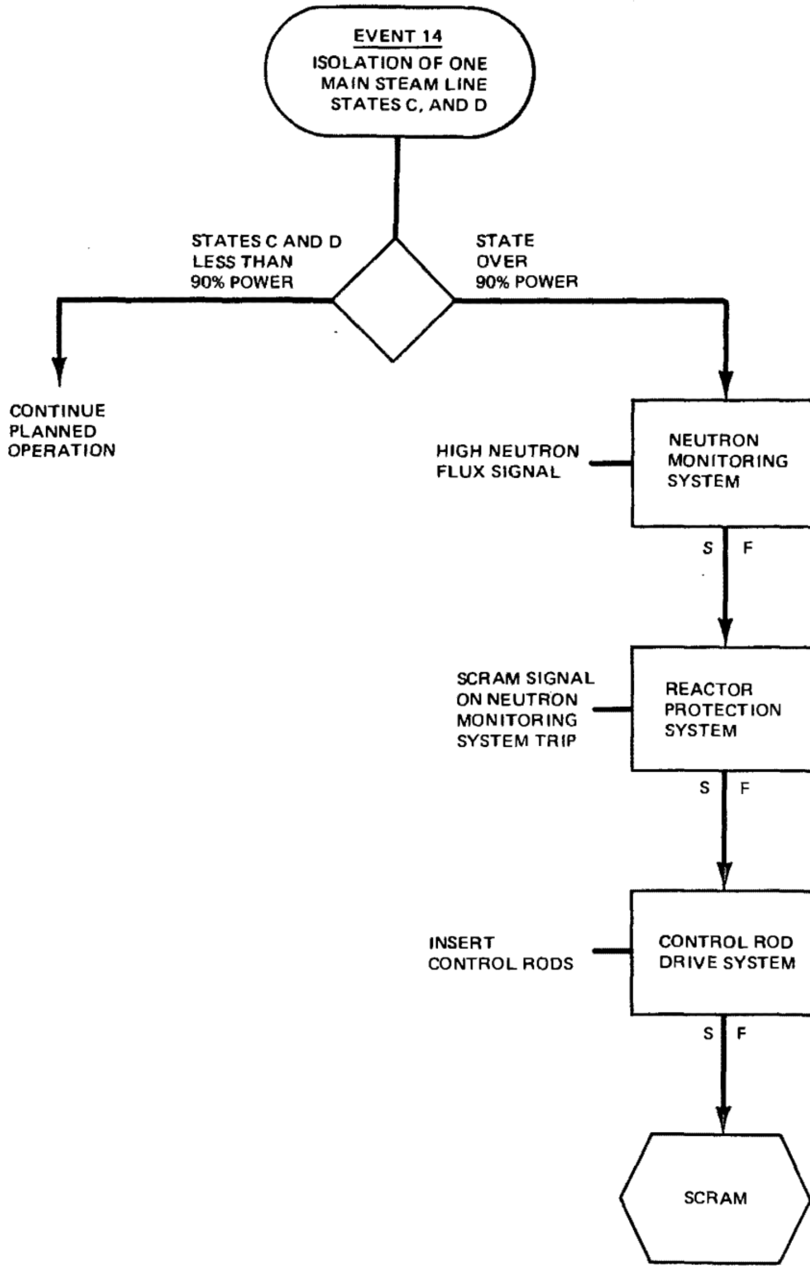
FIGURE 15A.6 - 12



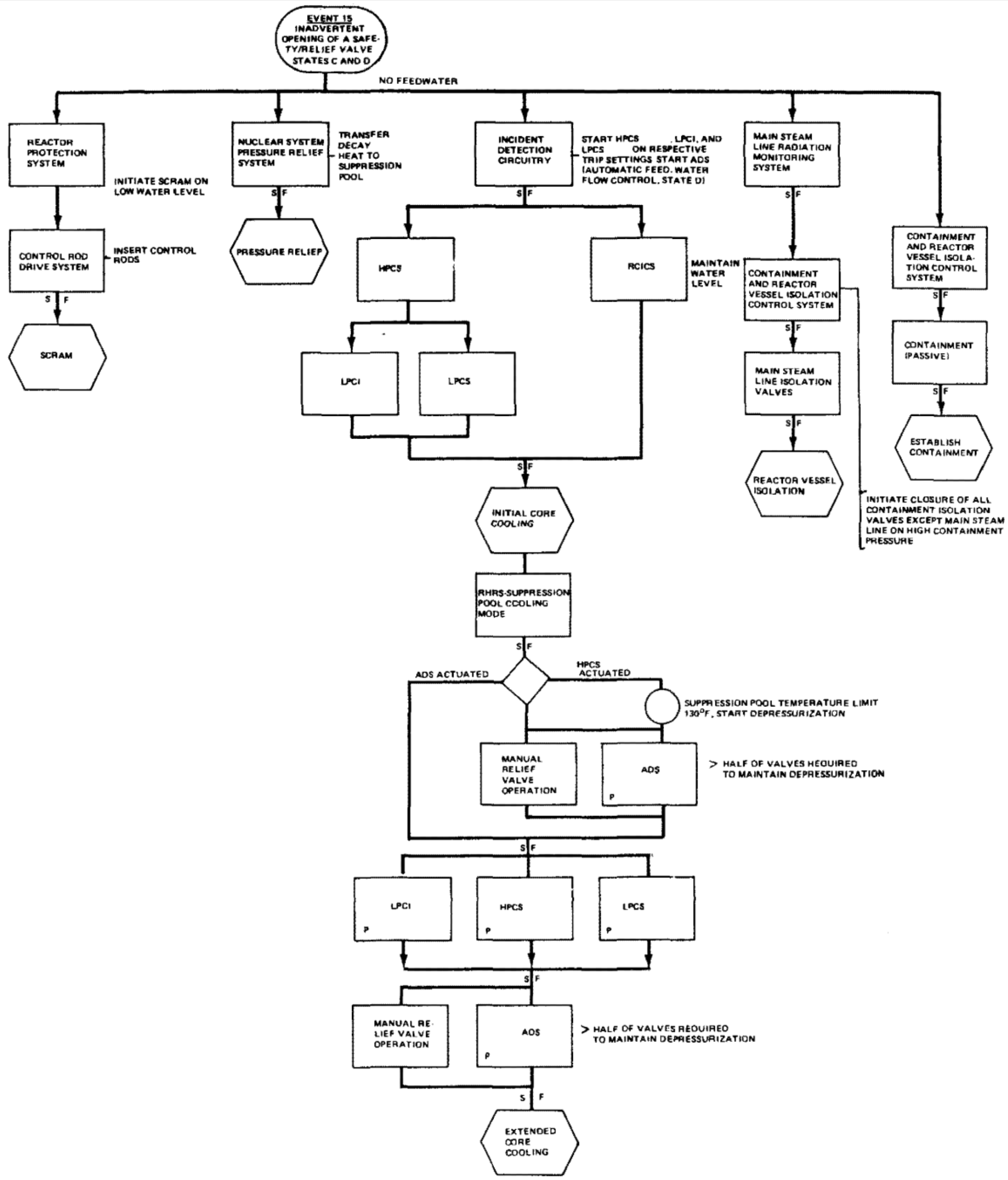
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	RECIRCULATION LOOP PUMP TRIP-ONE OR BOTH FIGURE 15A.6-13



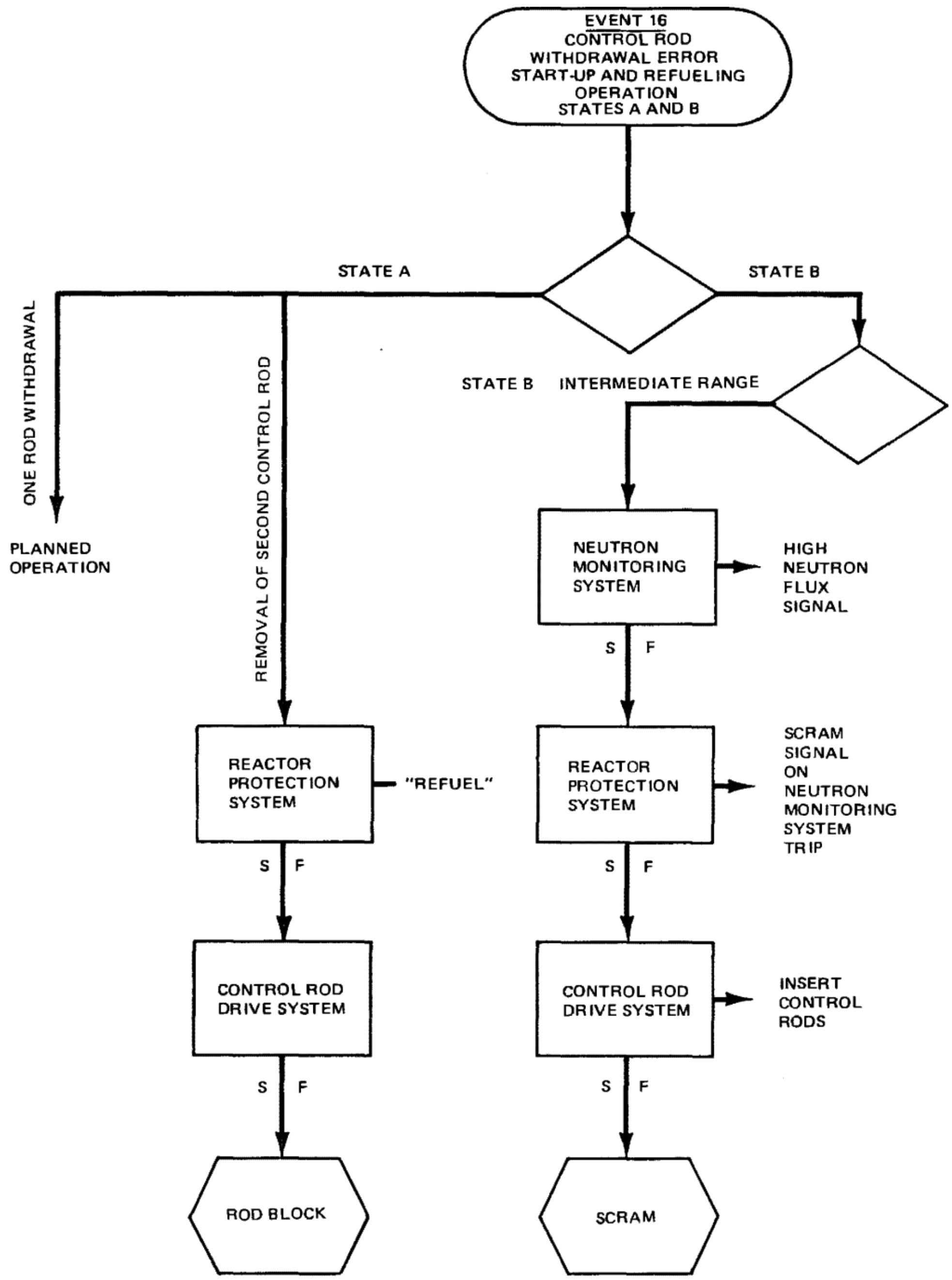
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PROTECTION SEQUENCES FOR ISOLATION OF ALL MAIN STEAM LINES FIGURE 15A.6-14a



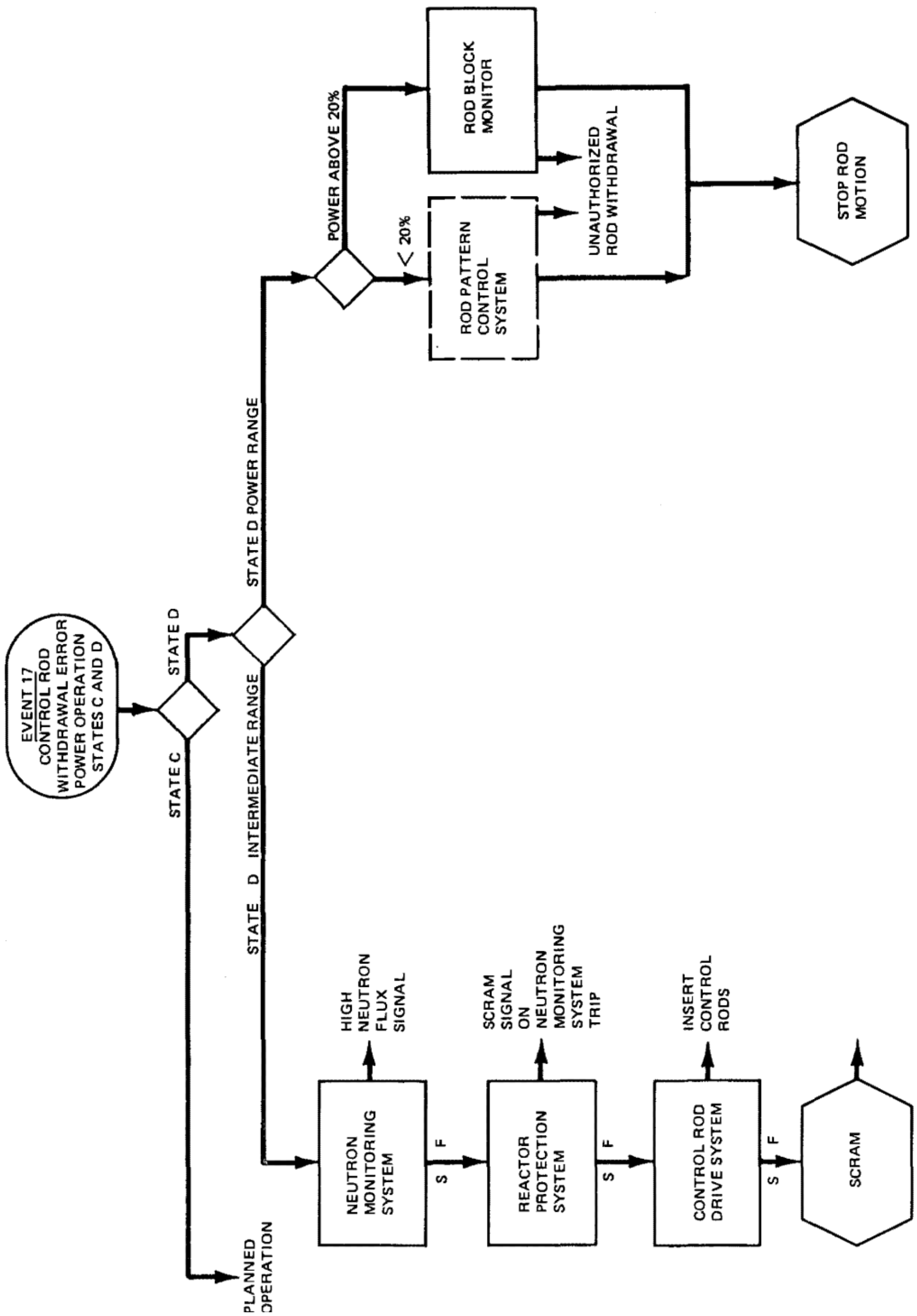
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PROTECTION SEQUENCE FOR ISOLATION OF ONE MAIN STEAM LINE FIGURE 15A.6-14b



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PROTECTION SEQUENCES FOR INADVERTENT OPENING OF A RELIEF OR SAFETY VALVE FIGURE 15A.6-15



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PROTECTION SEQUENCE FOR CONTROL ROD WITHDRAWAL ERROR START-UP AND REFUELING OPERATION FIGURE 15A.6-16



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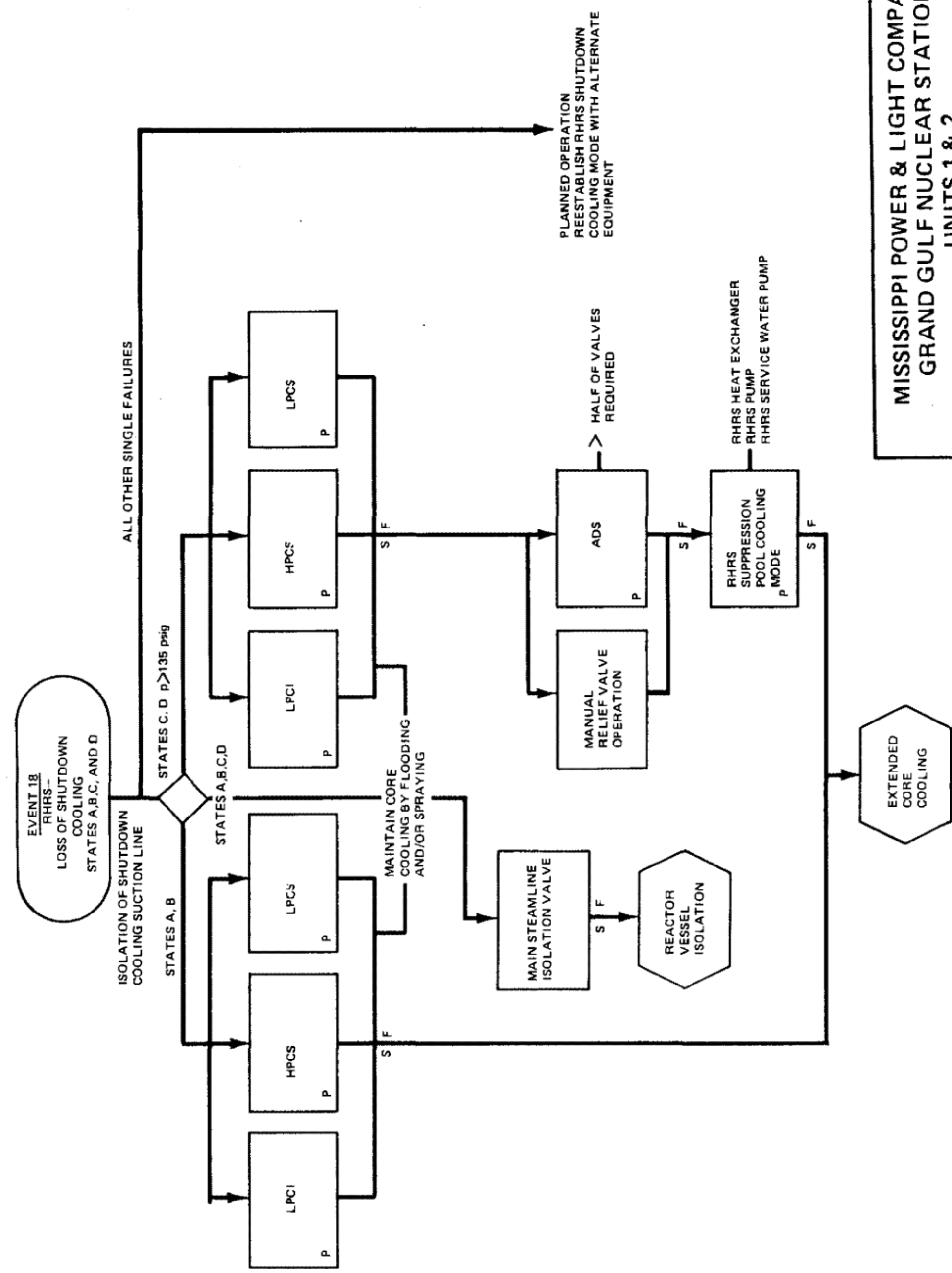
PROTECTION SEQUENCE FOR
CONTROL ROD WITHDRAWAL ERROR -
POWER OPERATION
FIGURE 15A.6-17



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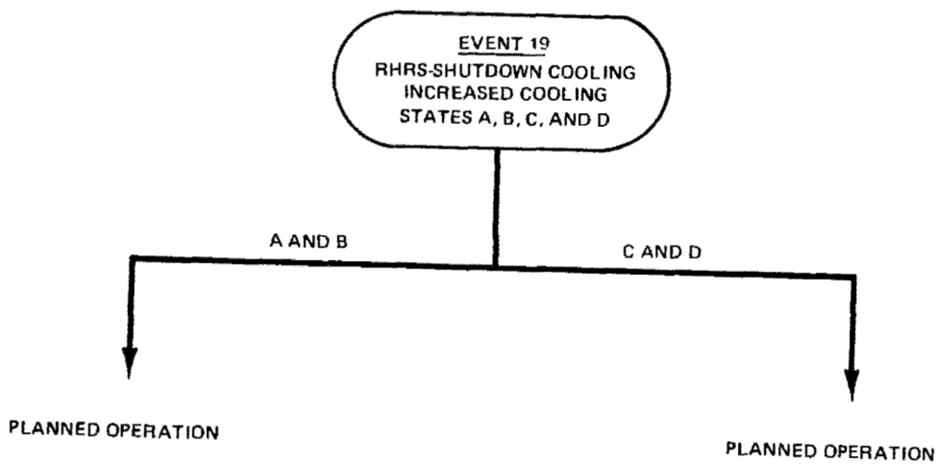
PROTECTION SEQUENCES FOR RHRS – LOSS OF SHUTDOWN COOLING FAILURE
FIGURE 15A.6-18



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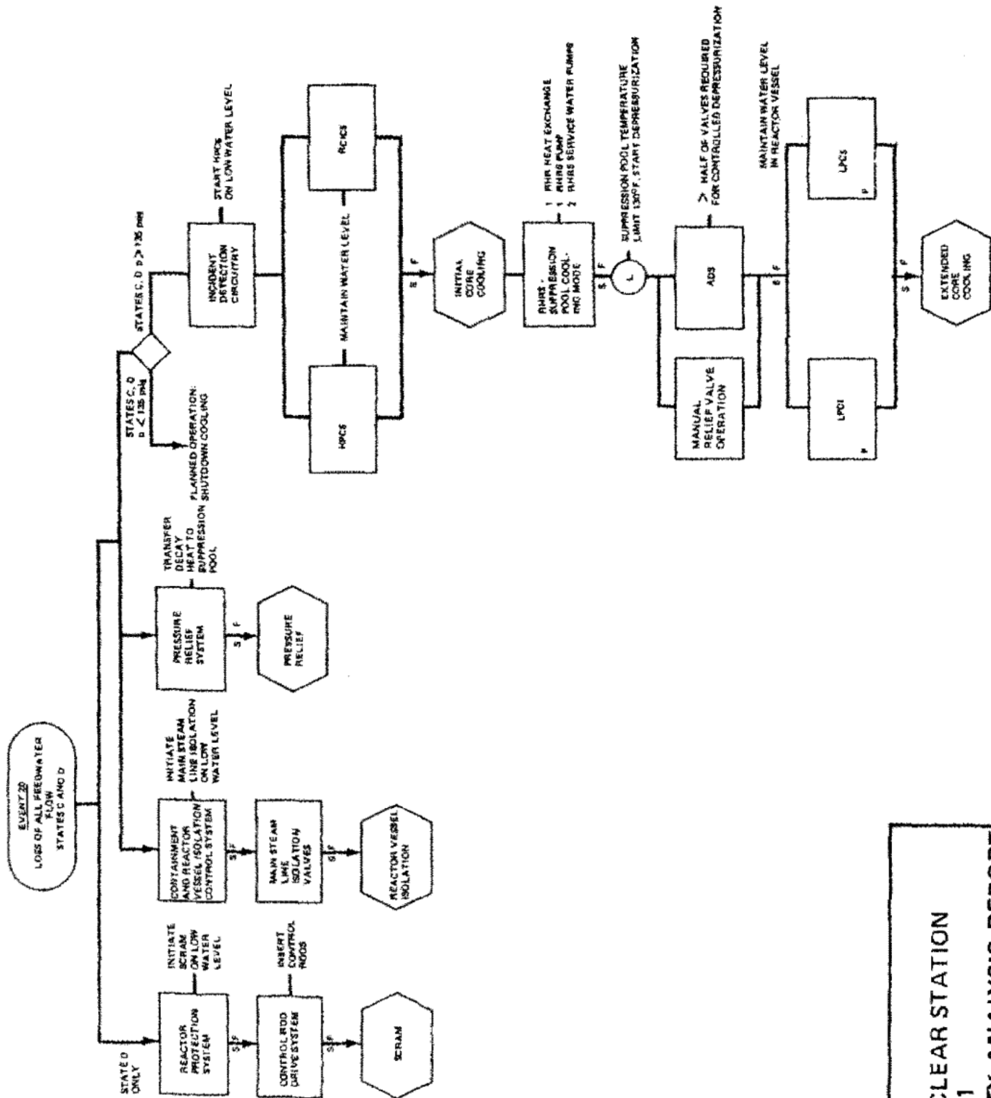
RHRS - SHUTDOWN COOLING FAILURE - INCREASED COOLING
FIGURE 15A.6-19



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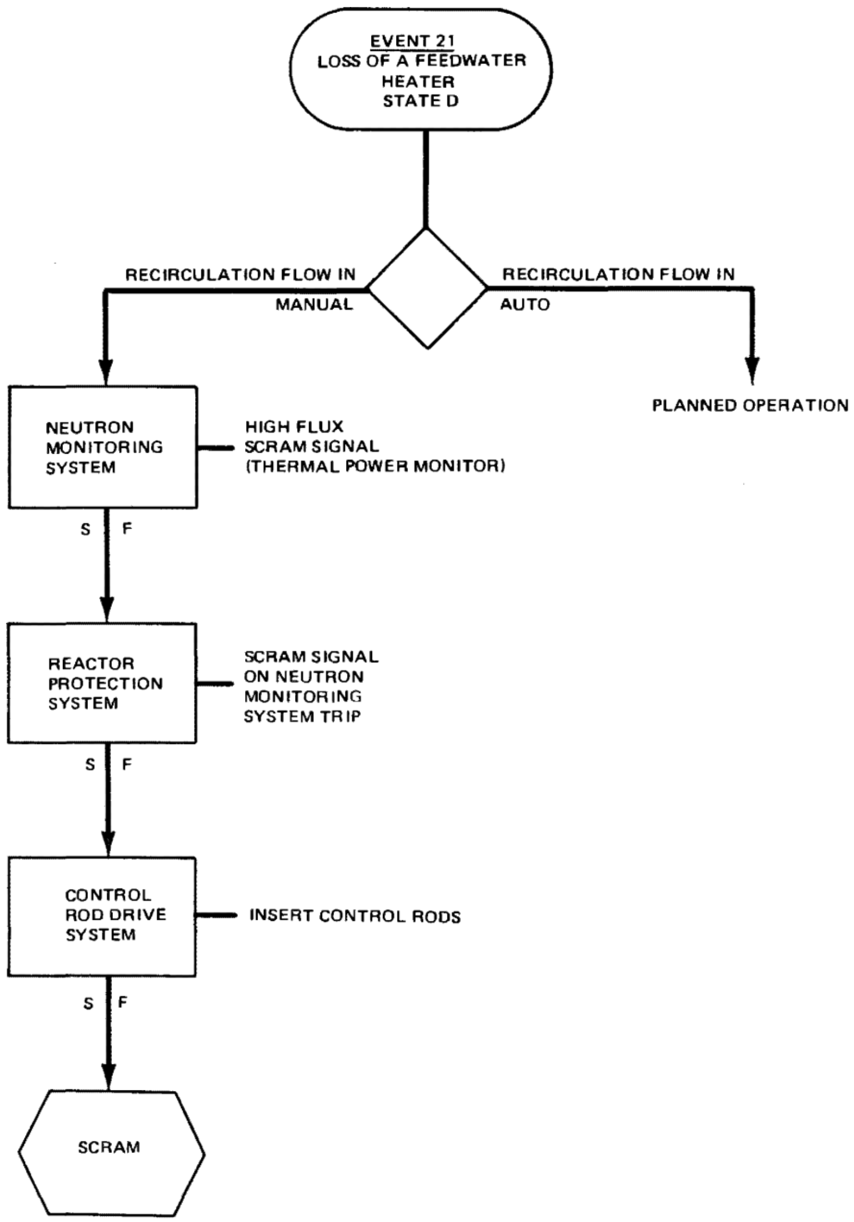
PROTECTION SEQUENCES FOR
LOSS OF FEEDWATER FLOW
FIGURE 15A.6-20



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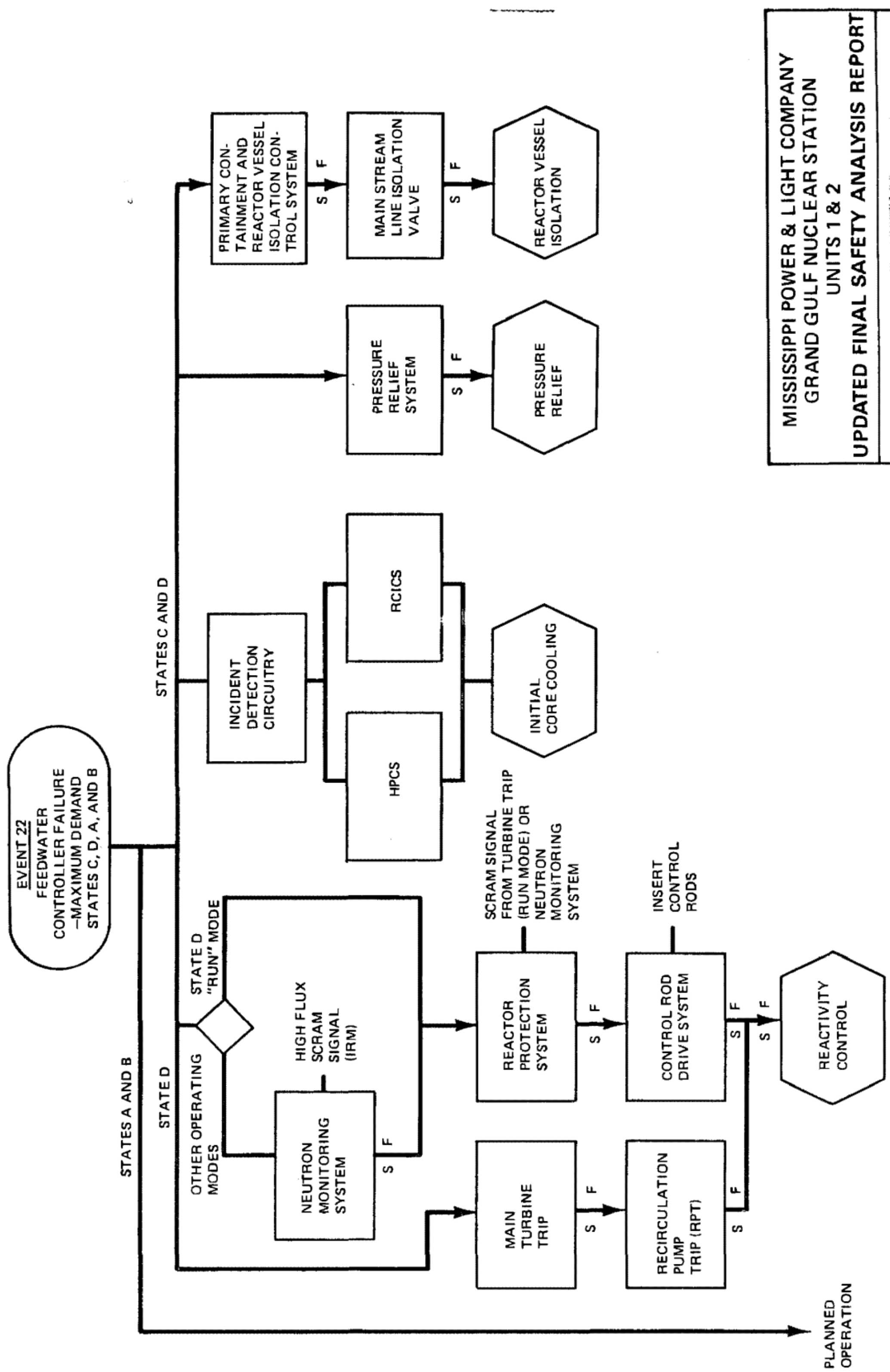
PROTECTION SEQUENCE FOR
LOSS OF A FEEDWATER HEATER
FIGURE 15A.6-21



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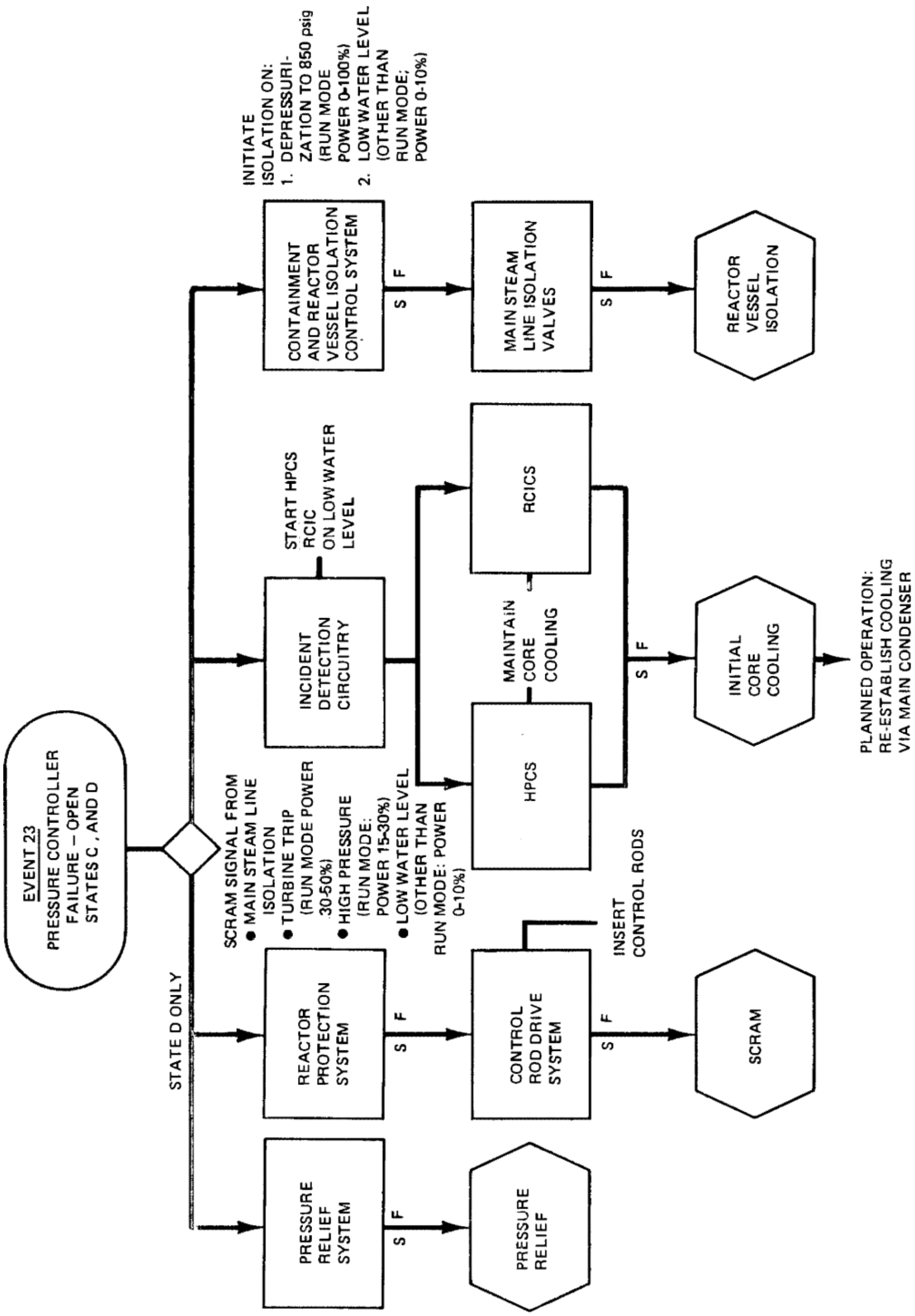
PROTECTION SEQUENCES FOR
FEEDWATER CONTROLLER FAILURE -
MAXIMUM DEMAND
FIGURE 15A.6-22



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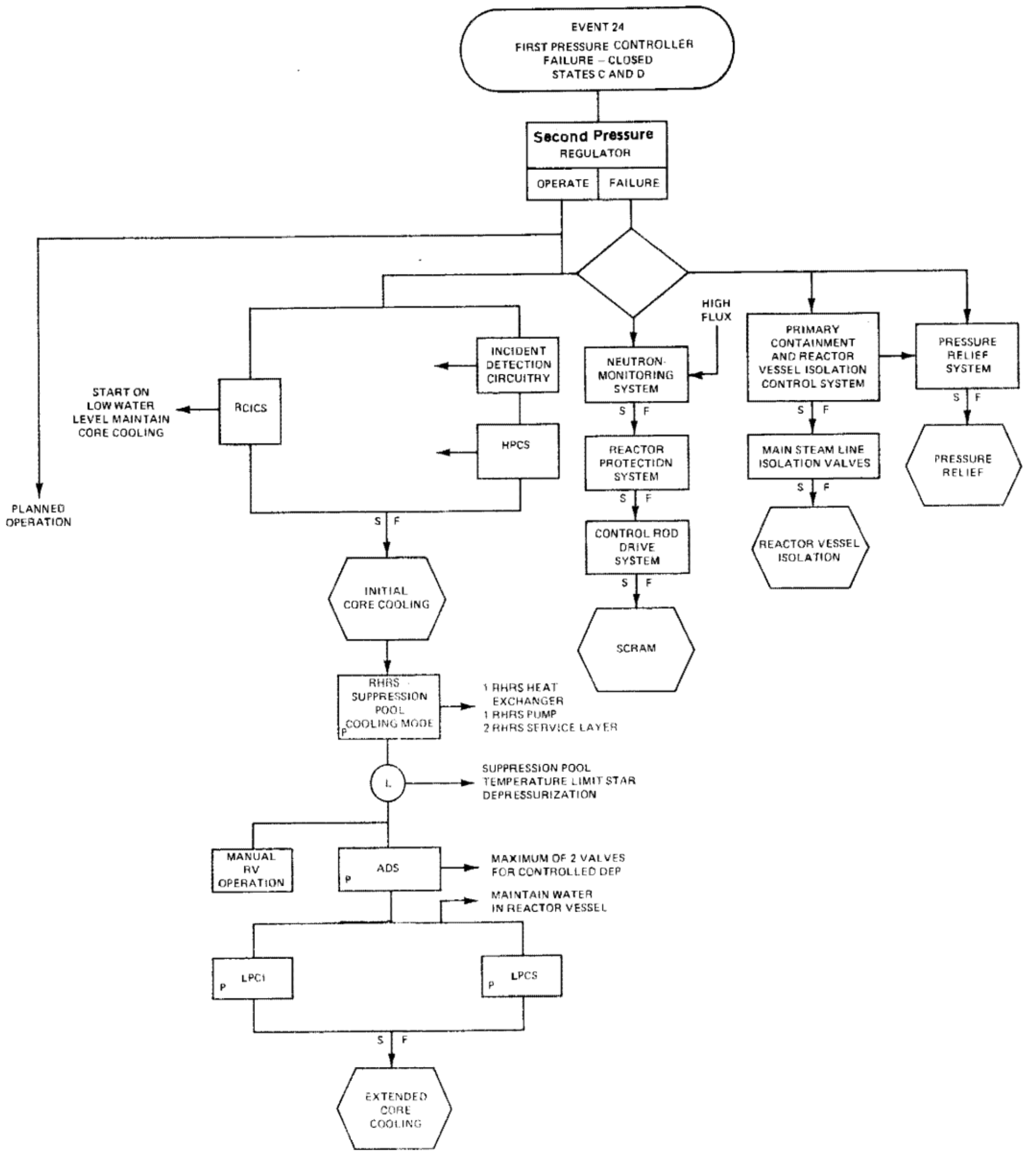
PROTECTION SEQUENCES FOR PRESSURE CONTROLLER FAILURE - OPEN

FIGURE 15A.6-23



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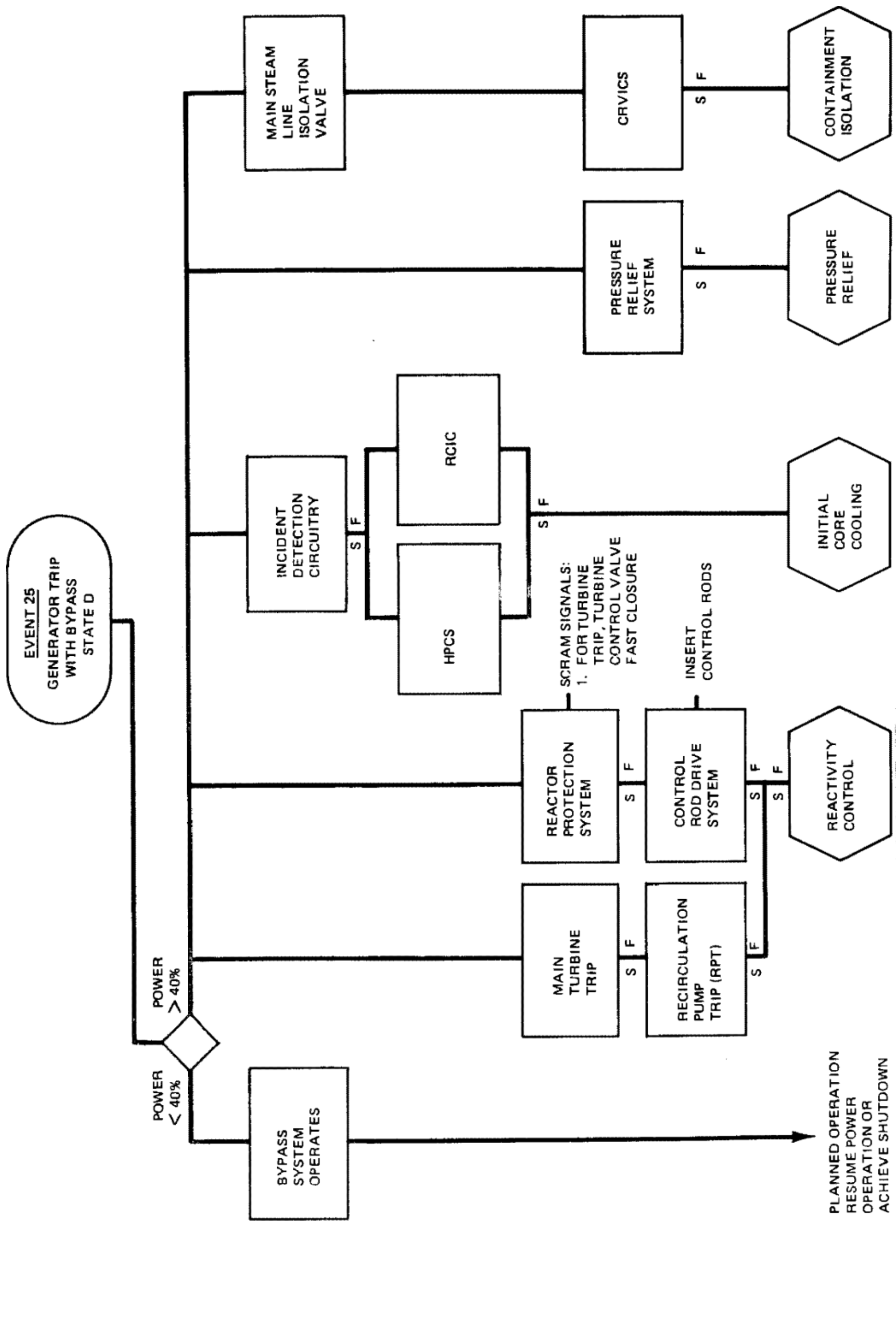
PROTECTION SEQUENCE FOR PRESSURE CONTROLLER FAILURE - CLOSED
FIGURE 15A.6-24



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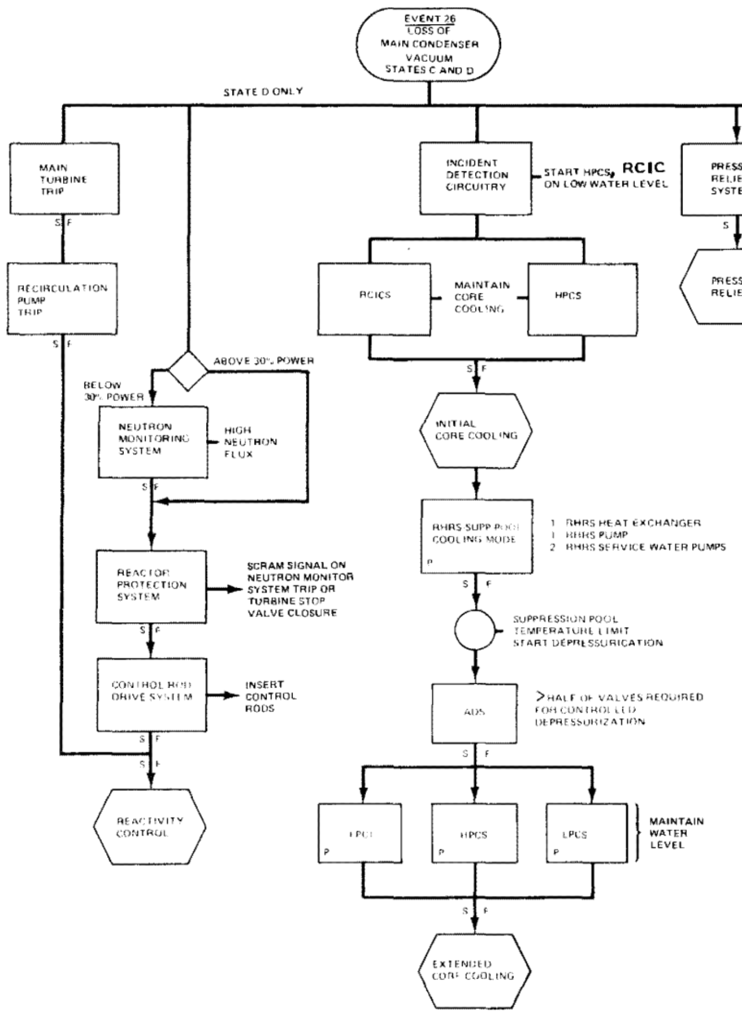
PROTECTION SEQUENCES FOR
MAIN TURBINE TRIP
WITH BYPASS

FIGURE 15A.6–25



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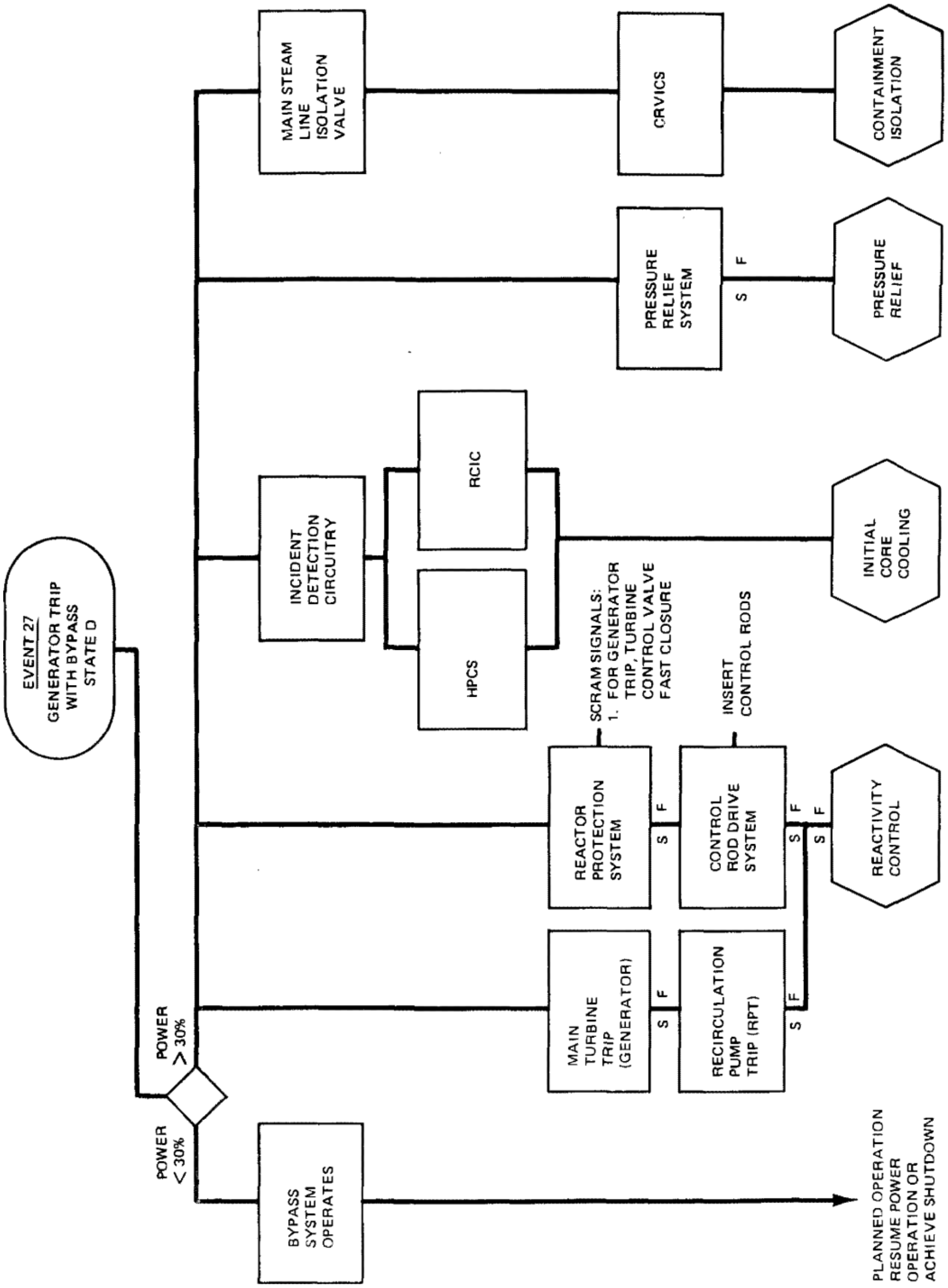
PROTECTION SEQUENCES FOR
LOSS OF MAIN
CONDENSER VACUUM
FIGURE 15A.6-26



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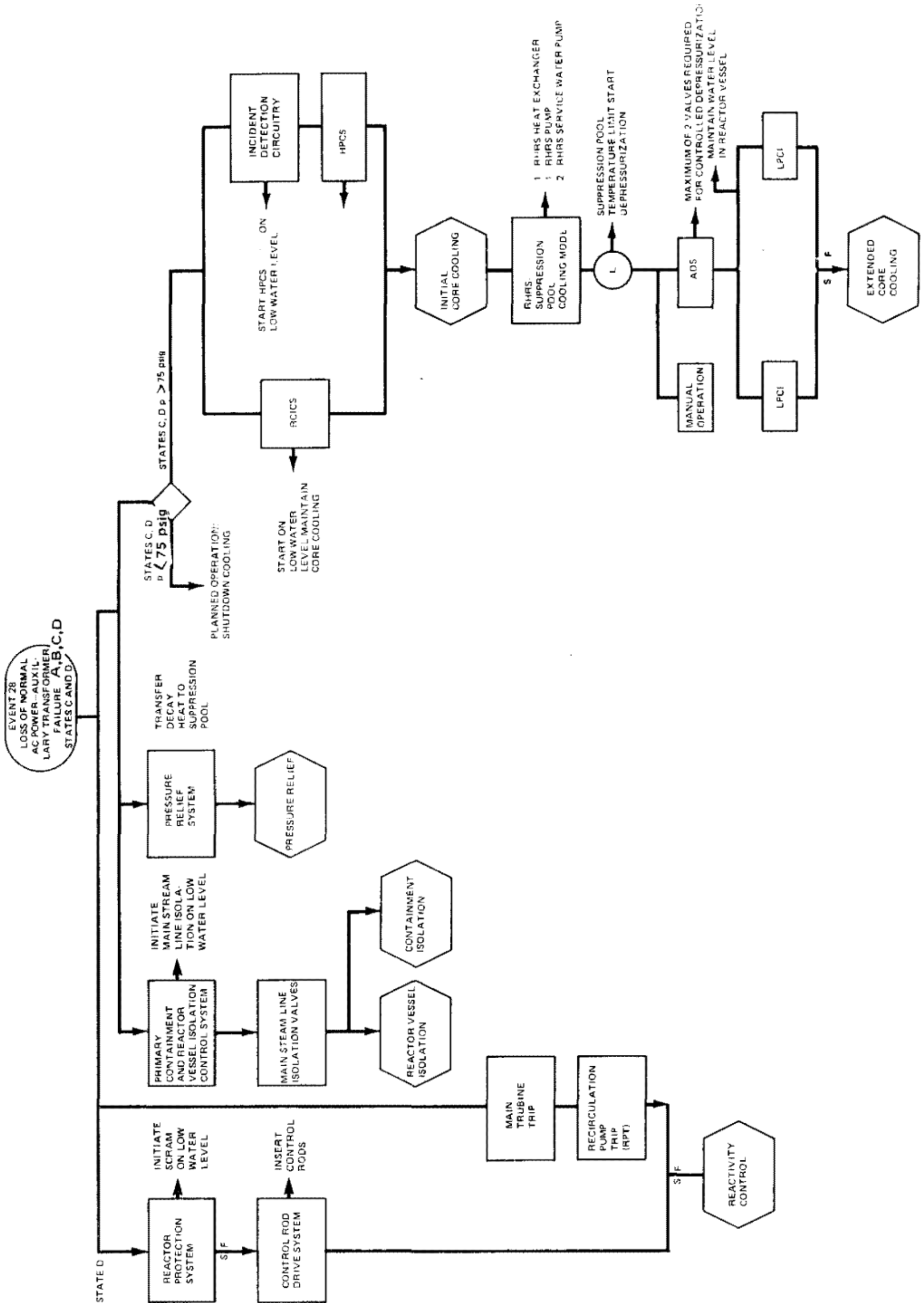
PROTECTION SEQUENCES FOR
MAIN GENERATOR TRIP
WITH BYPASS
FIGURE 15A.6-27



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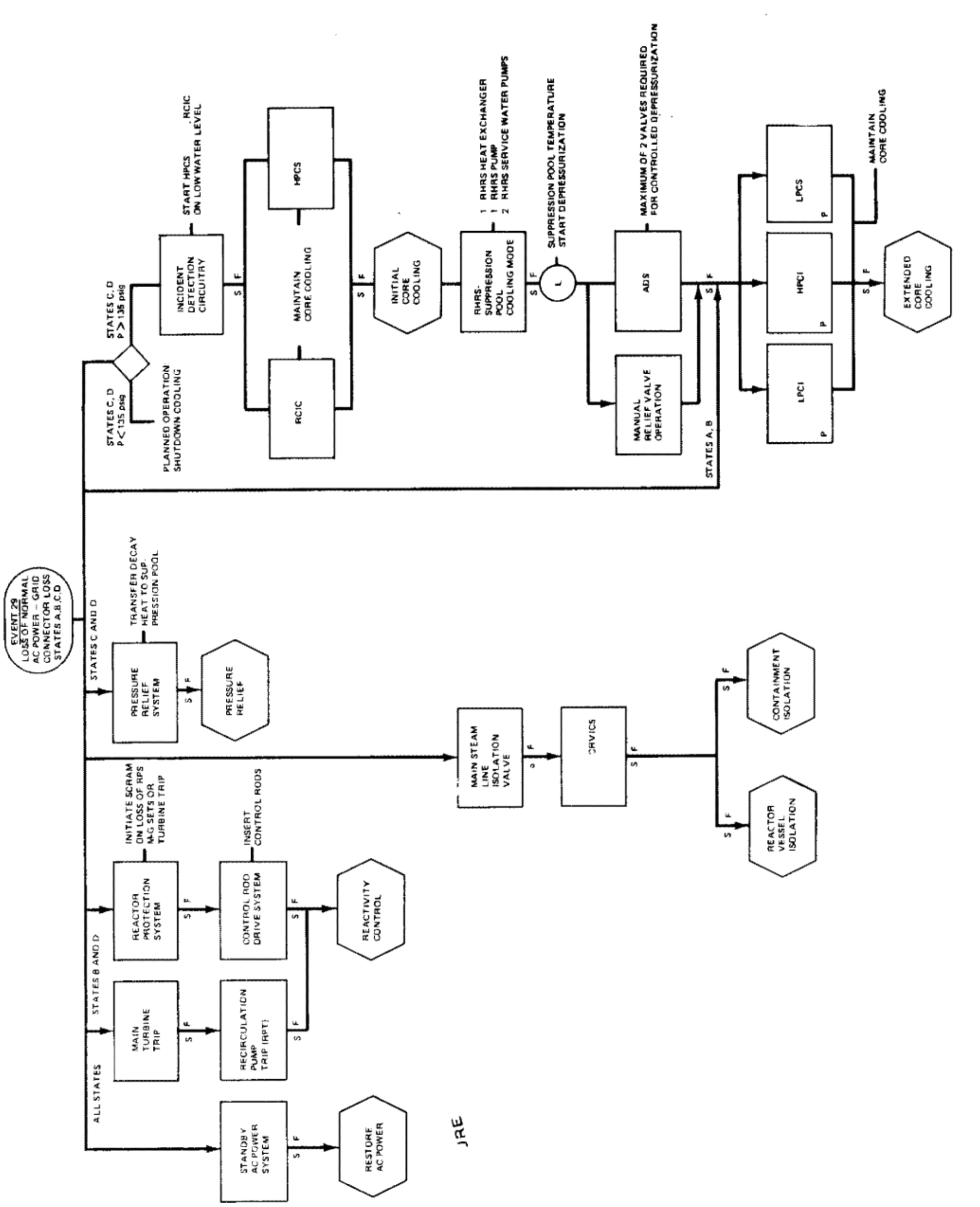
PROTECTION SEQUENCE FOR LOSS OF NORMAL AC POWER – AUXILIARY TRANSFORMER FAILURE
FIGURE 15A.6-28



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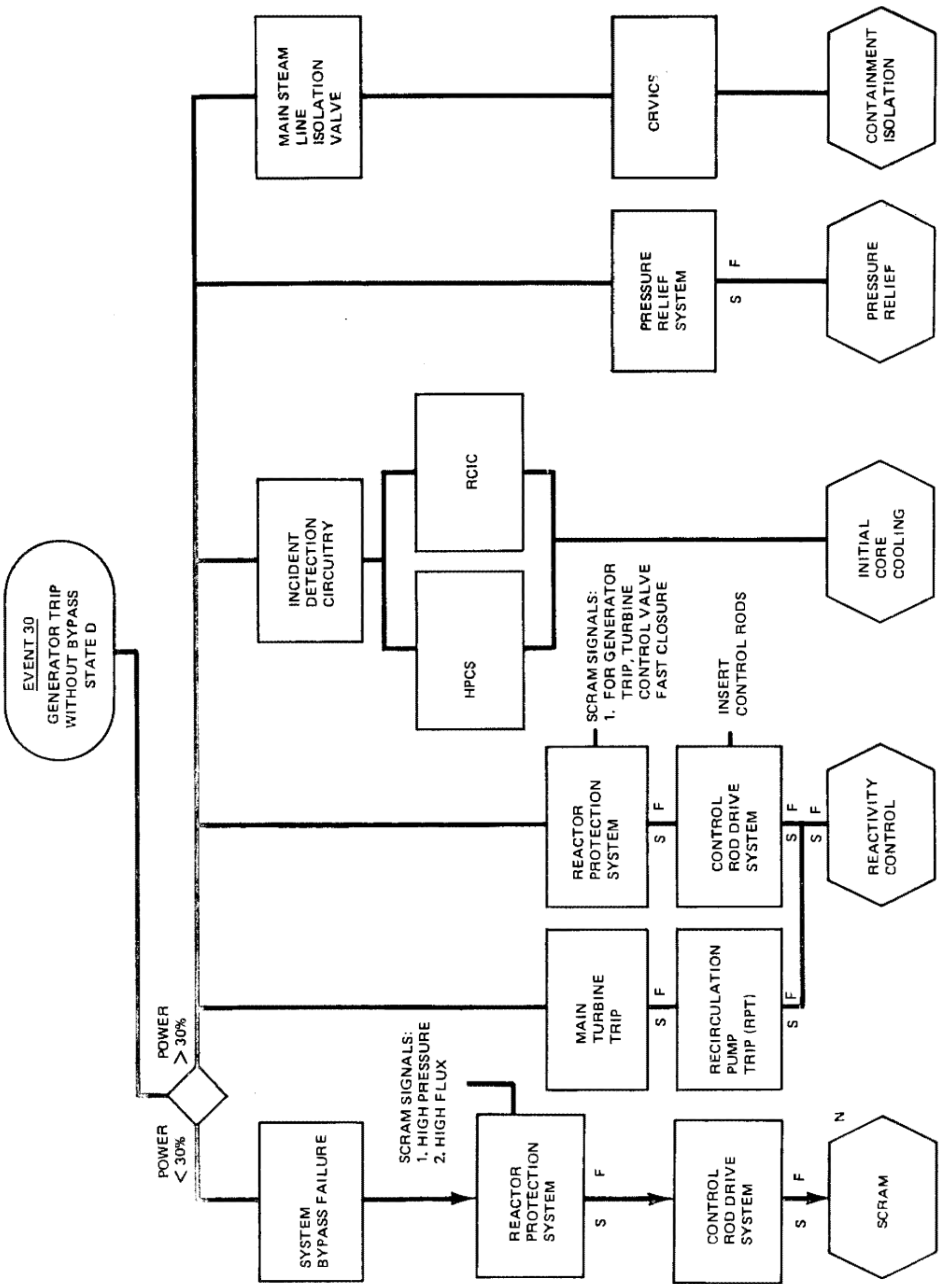
PROTECTION SEQUENCES FOR LOSS OF NORMAL AC POWER – GRID CONNECTION LOSS
FIGURE 15A.6-29



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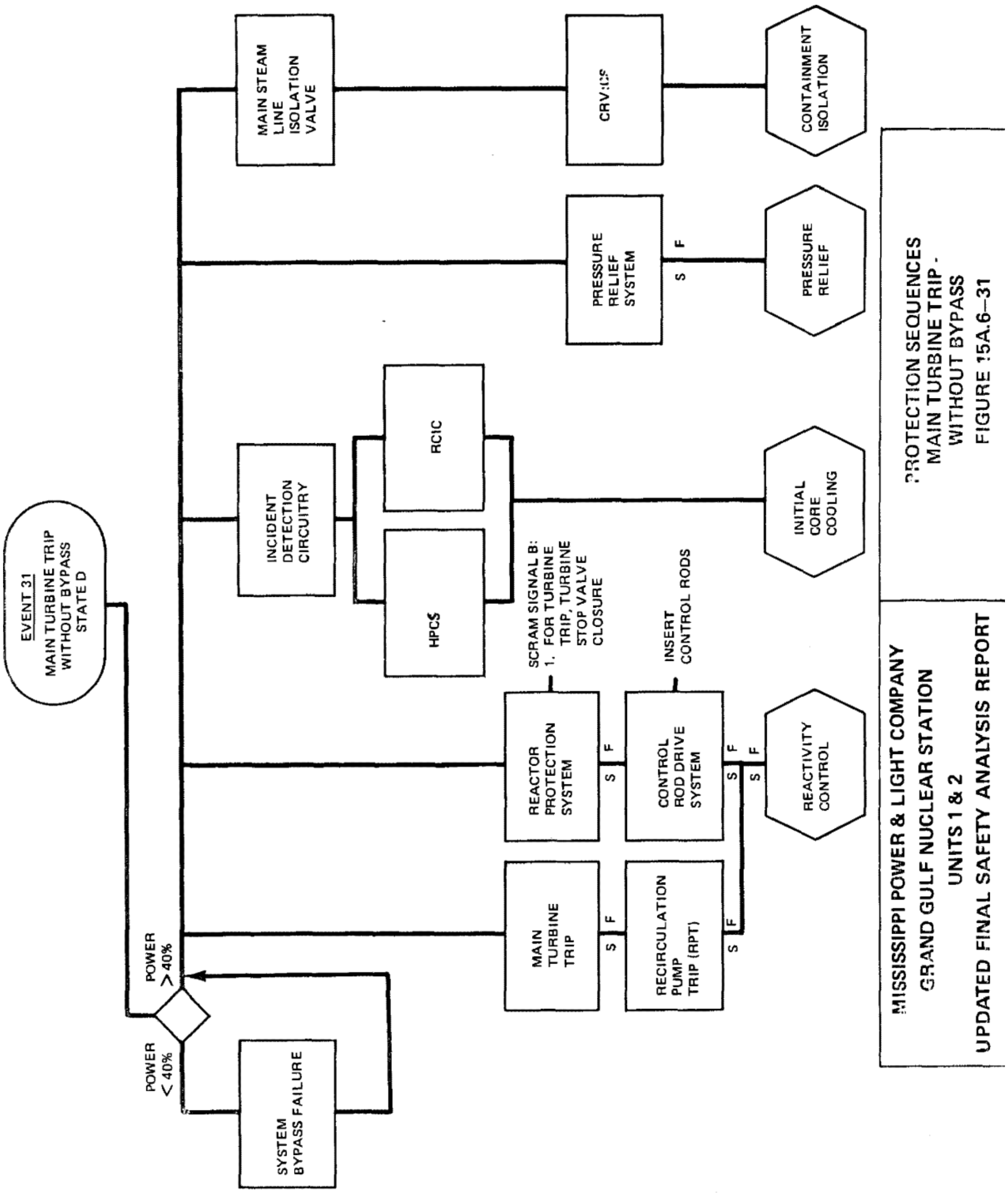
PROTECTION SEQUENCES
MAIN GENERATOR TRIP -
WITHOUT BYPASS
FIGURE 15A.6-30


EVENT 31
MAIN TURBINE TRIP WITHOUT BYPASS STATE D
POWER > 40%
POWER < 40%
SYSTEM BYPASS FAILURE
MAIN TURBINE TRIP
RECIRCULATION PUMP TRIP (RPT)
REACTOR PROTECTION SYSTEM
CONTROL ROD DRIVE SYSTEM
SCRAM SIGNAL B: 1. FOR TURBINE TRIP, TURBINE STOP VALVE CLOSURE
INSERT CONTROL RODS
REACTIVITY CONTROL
INCIDENT DETECTION CIRCUITRY
HPCS
RCIC
INITIAL CORE COOLING
PRESSURE RELIEF SYSTEM
PRESSURE RELIEF
MAIN STEAM LINE ISOLATION VALVE
CRVICS
CONTAINMENT ISOLATION
S
F
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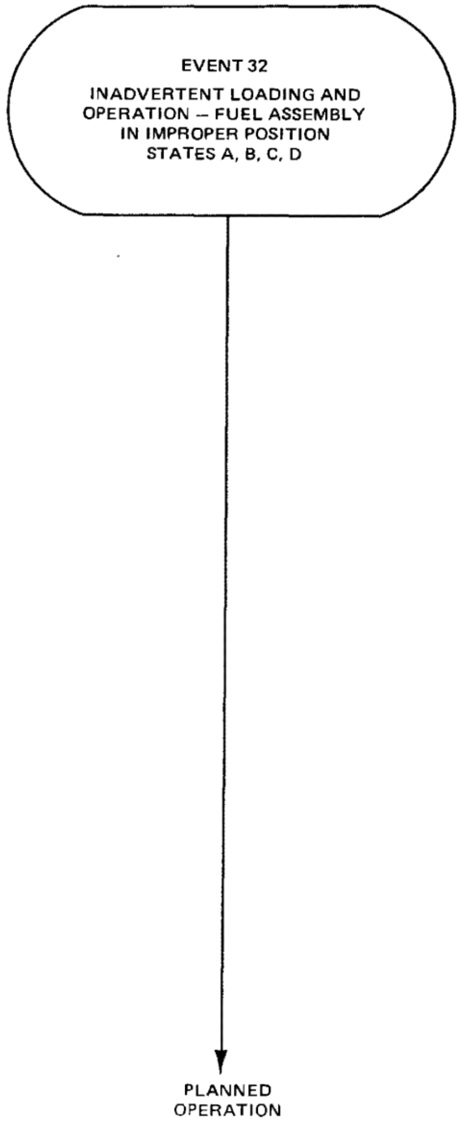
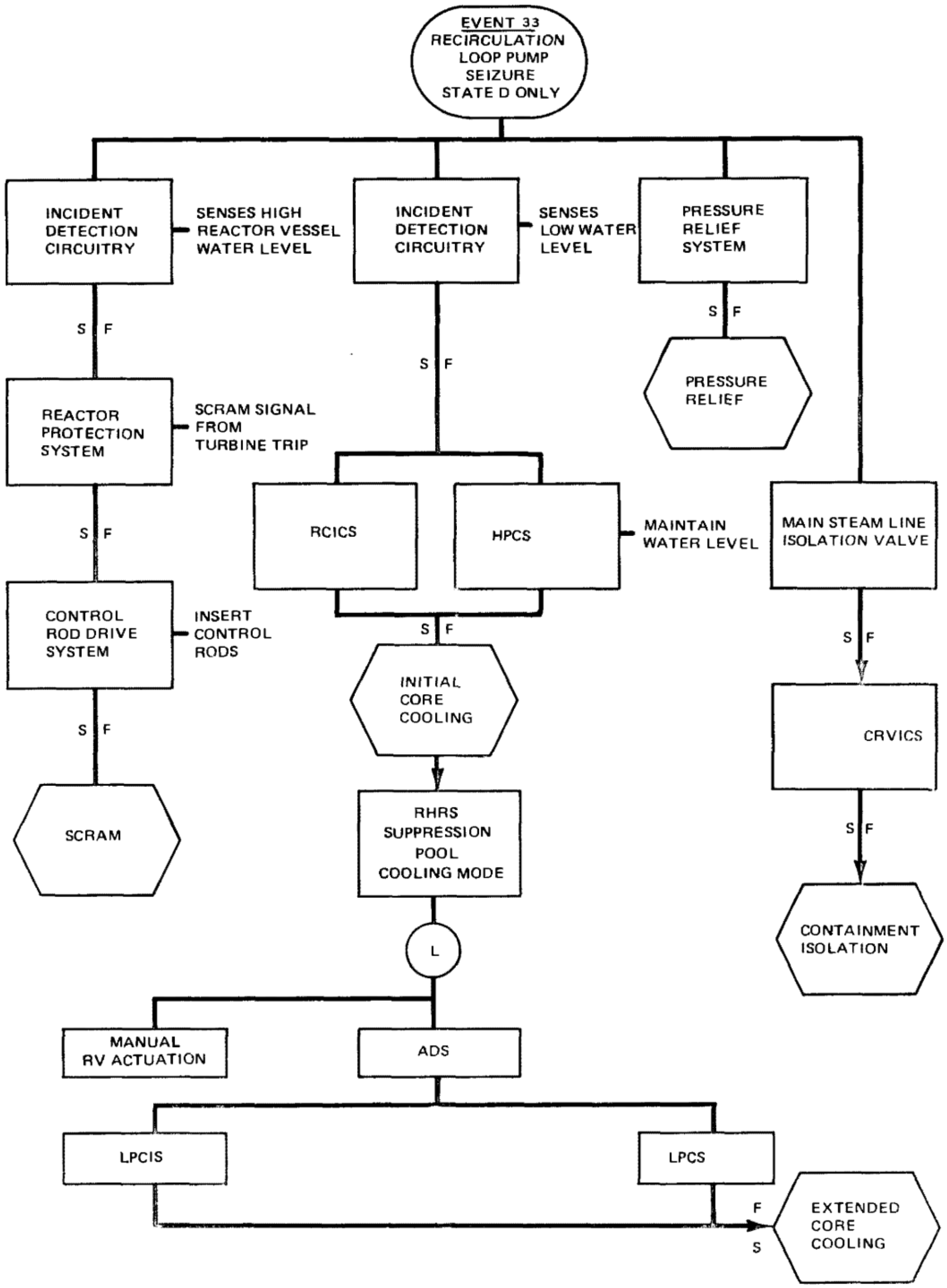
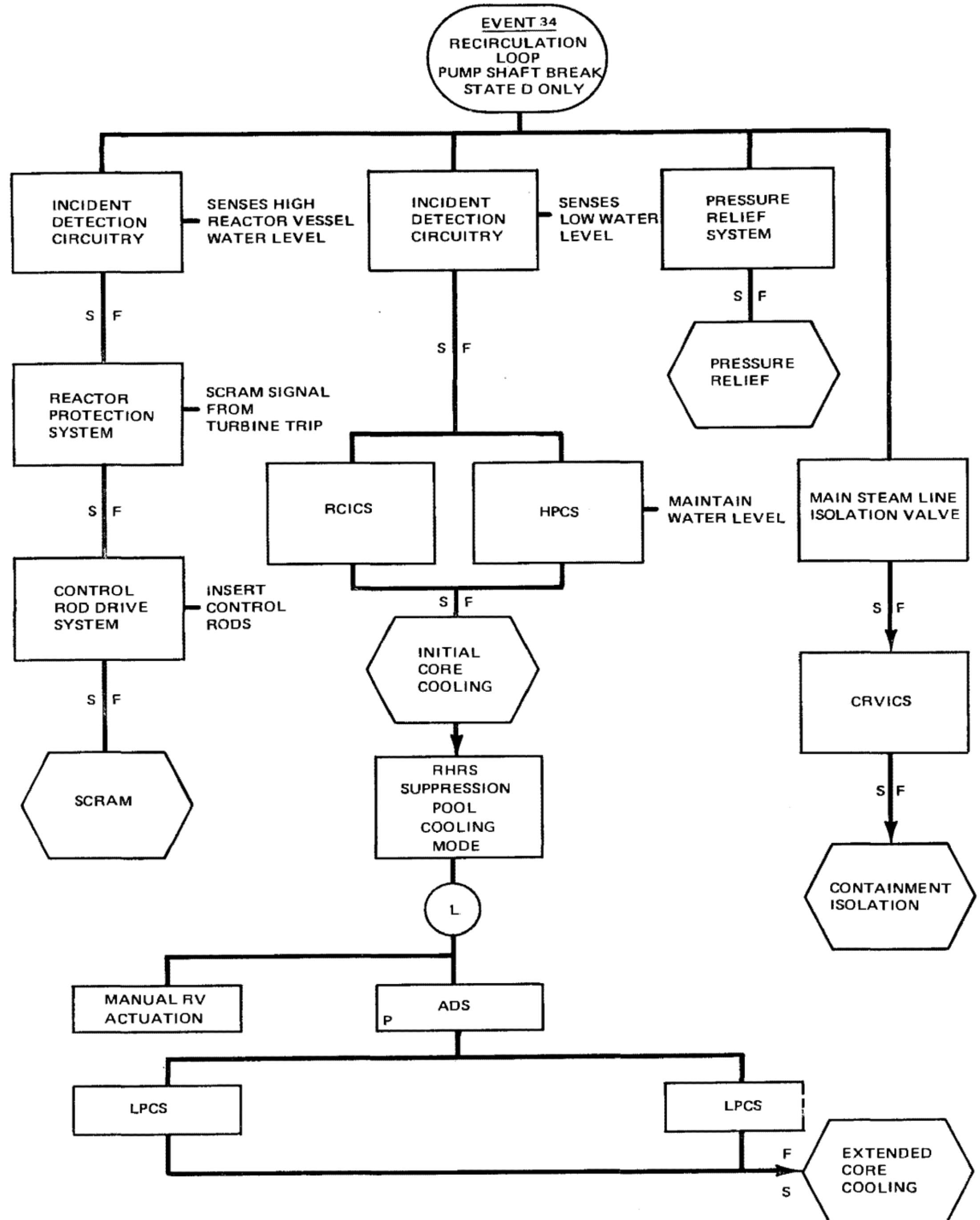
PROTECTION SEQUENCES MAIN TURBINE TRIP - WITHOUT BYPASS
FIGURE 15A.6–31



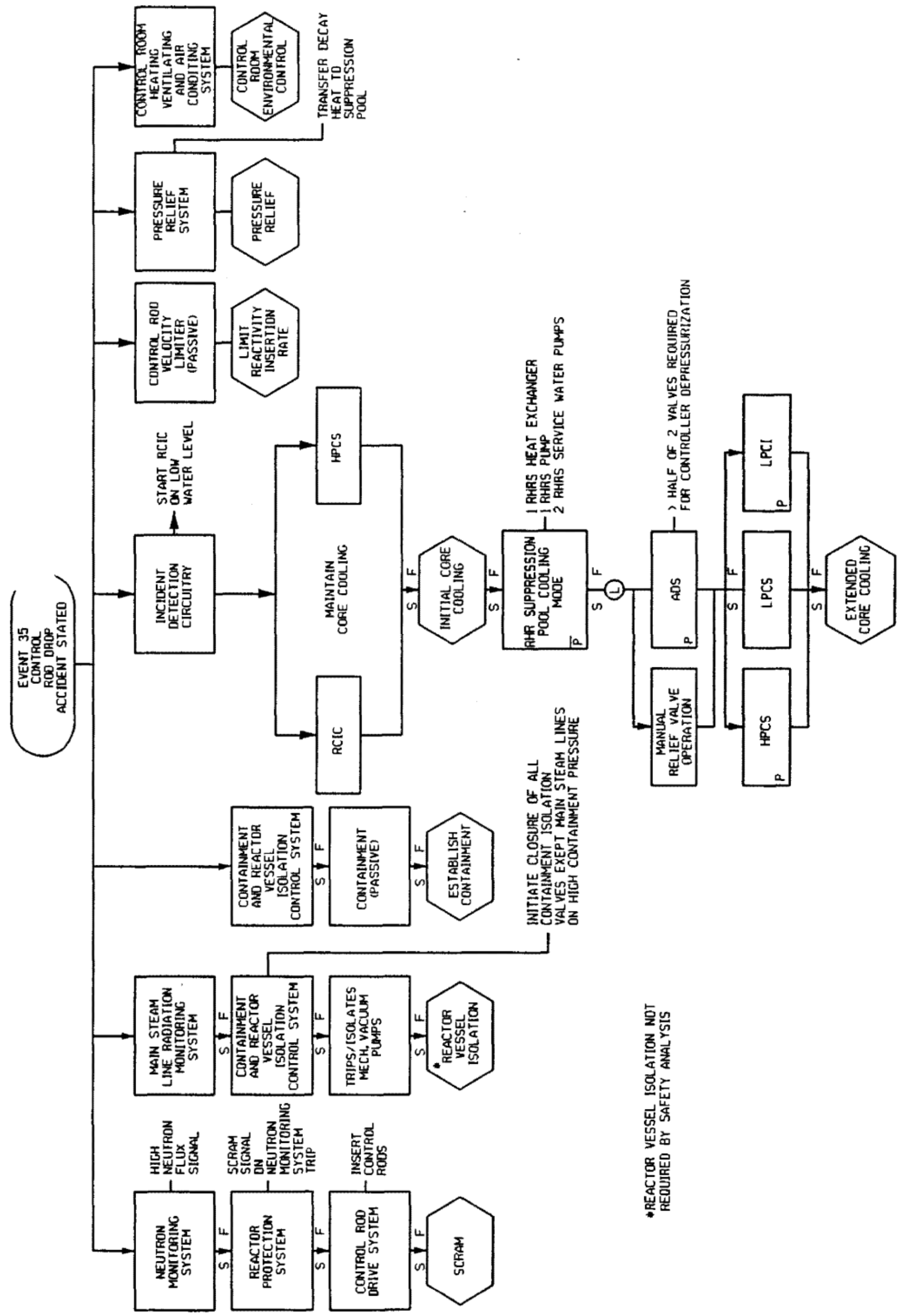
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PROTECTION SEQUENCE FOR INADVERTENT LOADING AND OPERATION OF FUEL ASSEMBLY IN IMPROPER POSITION FIGURE 15A.6-32



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PROTECTION SEQUENCE FOR RECIRCULATION LOOP PUMP SEIZURE FIGURE 15A.6-33



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PROTECTION SEQUENCE FOR RECIRCULATION LOOP PUMP SHAFT BREAK FIGURE 15A.6-34



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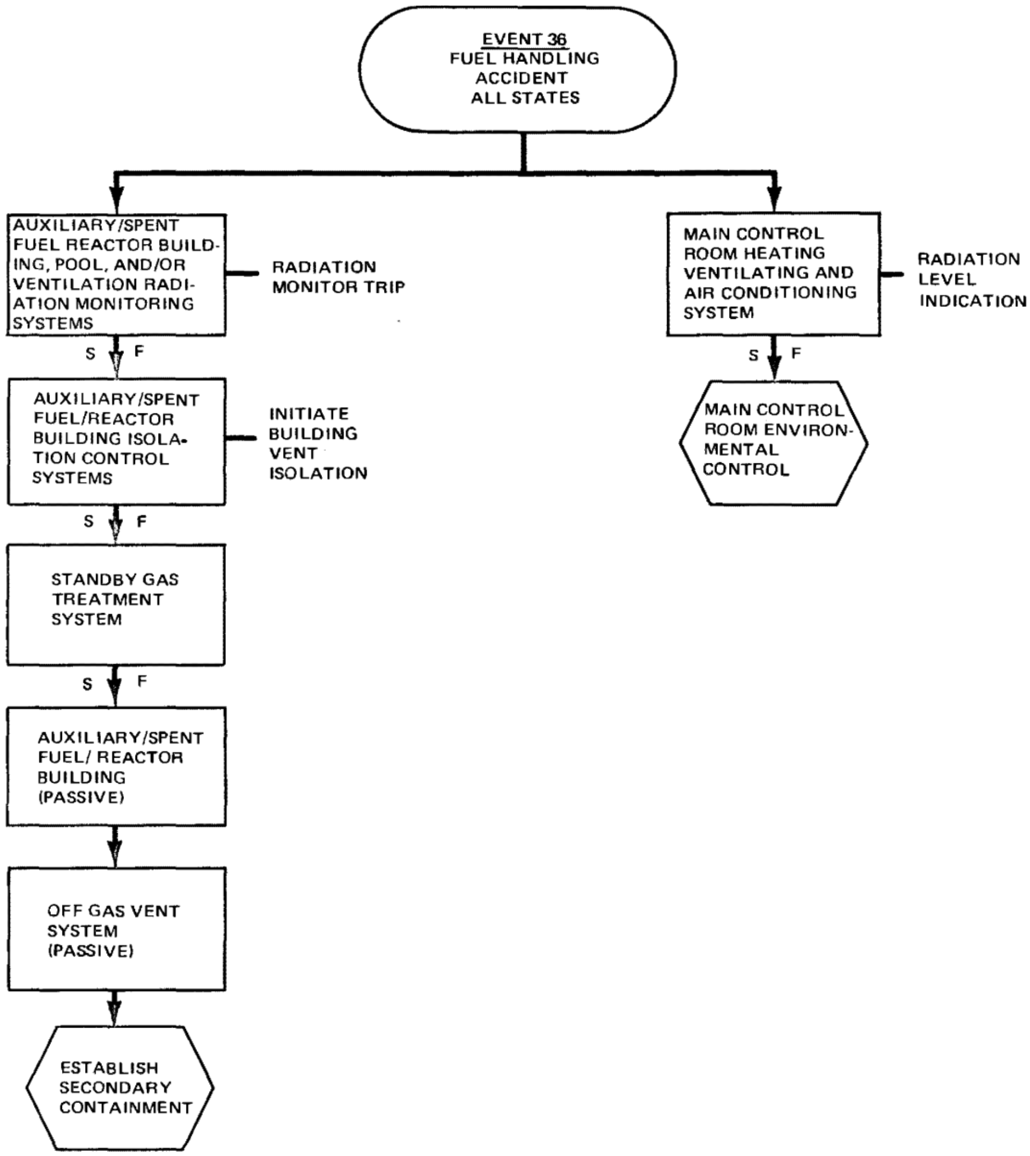
PROTECTION SEQUENCES FOR CONTROL ROD DROP ACCIDENT
FIGURE 15A.6-35



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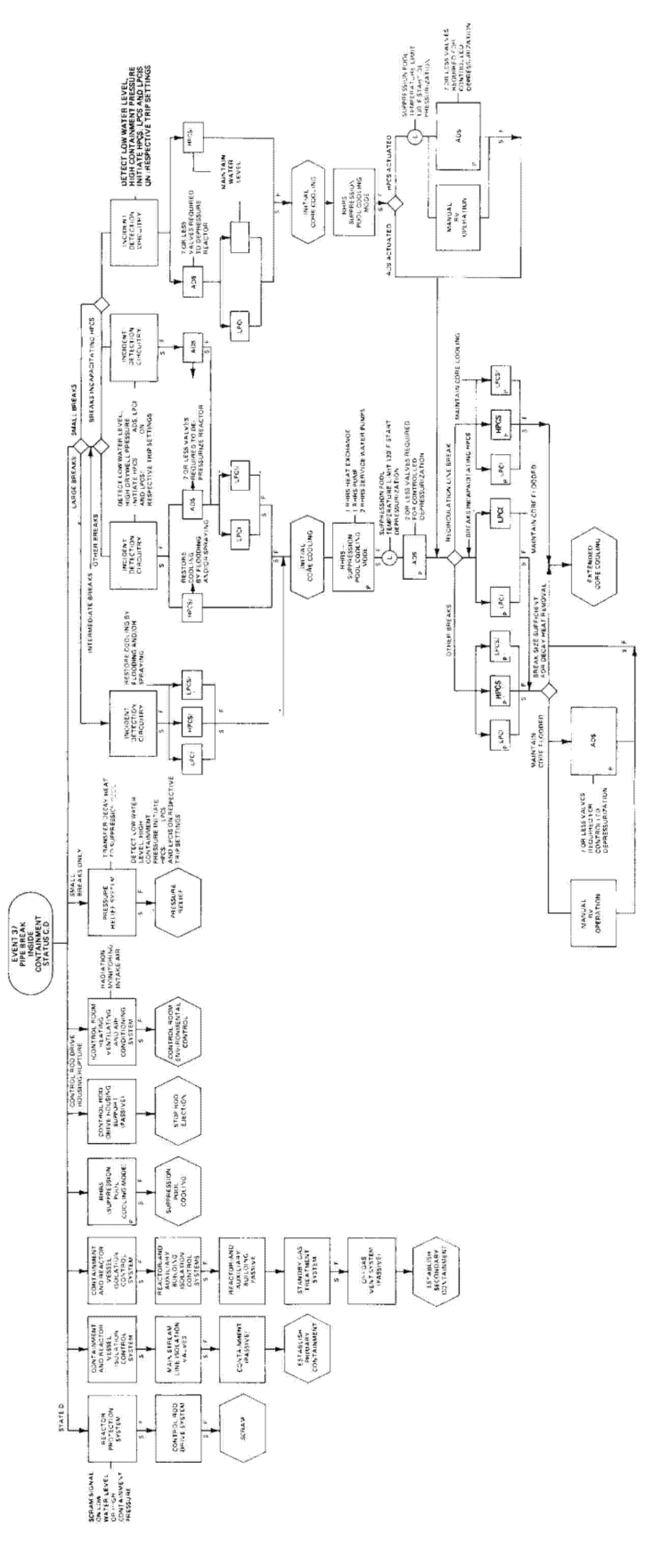
PROTECTION SEQUENCES FOR
FUEL HANDLING ACCIDENT
FIGURE 15A.6-36



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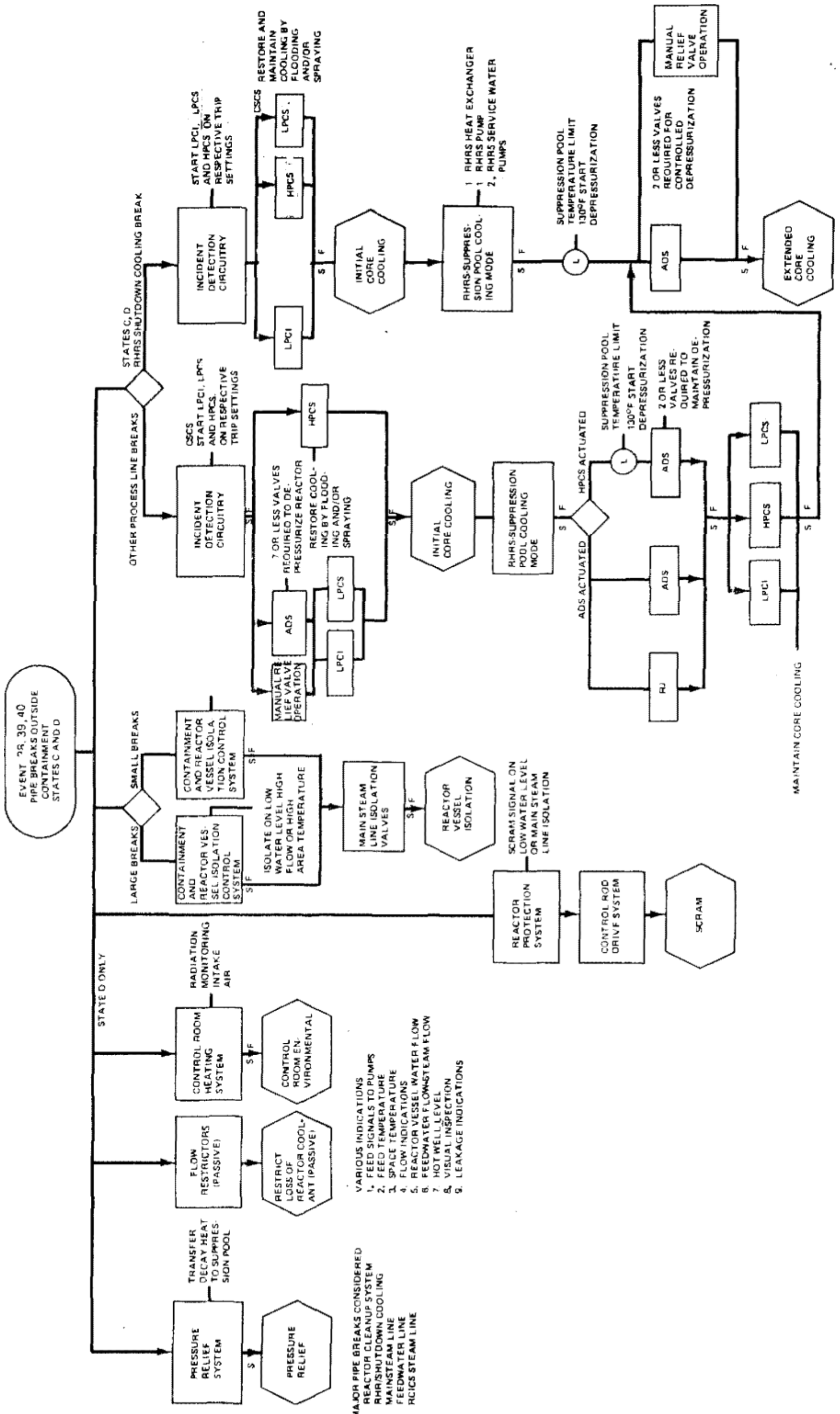
PROTECTION SEQUENCES FOR
LOSS-OF-COOLANT PIPING BREAKS
IN RCPB-INSIDE CONTAINMENT
FIGURE 15A.6-37



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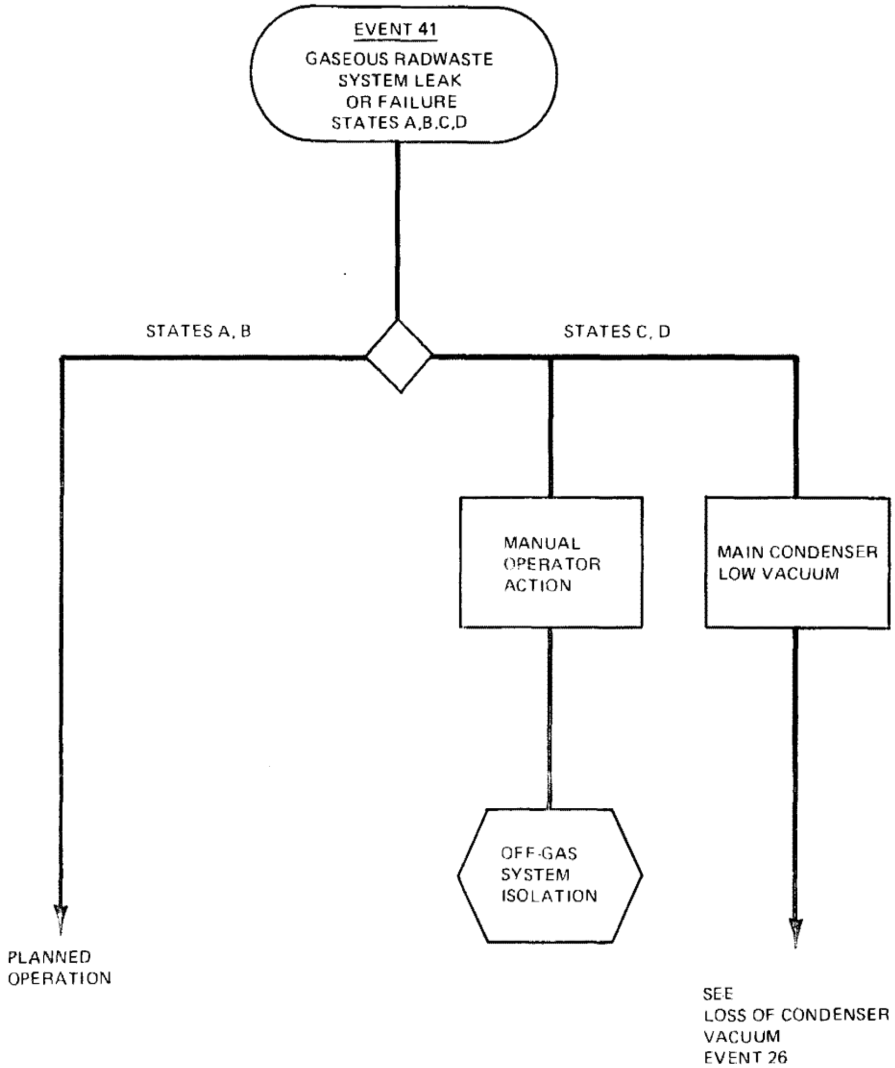
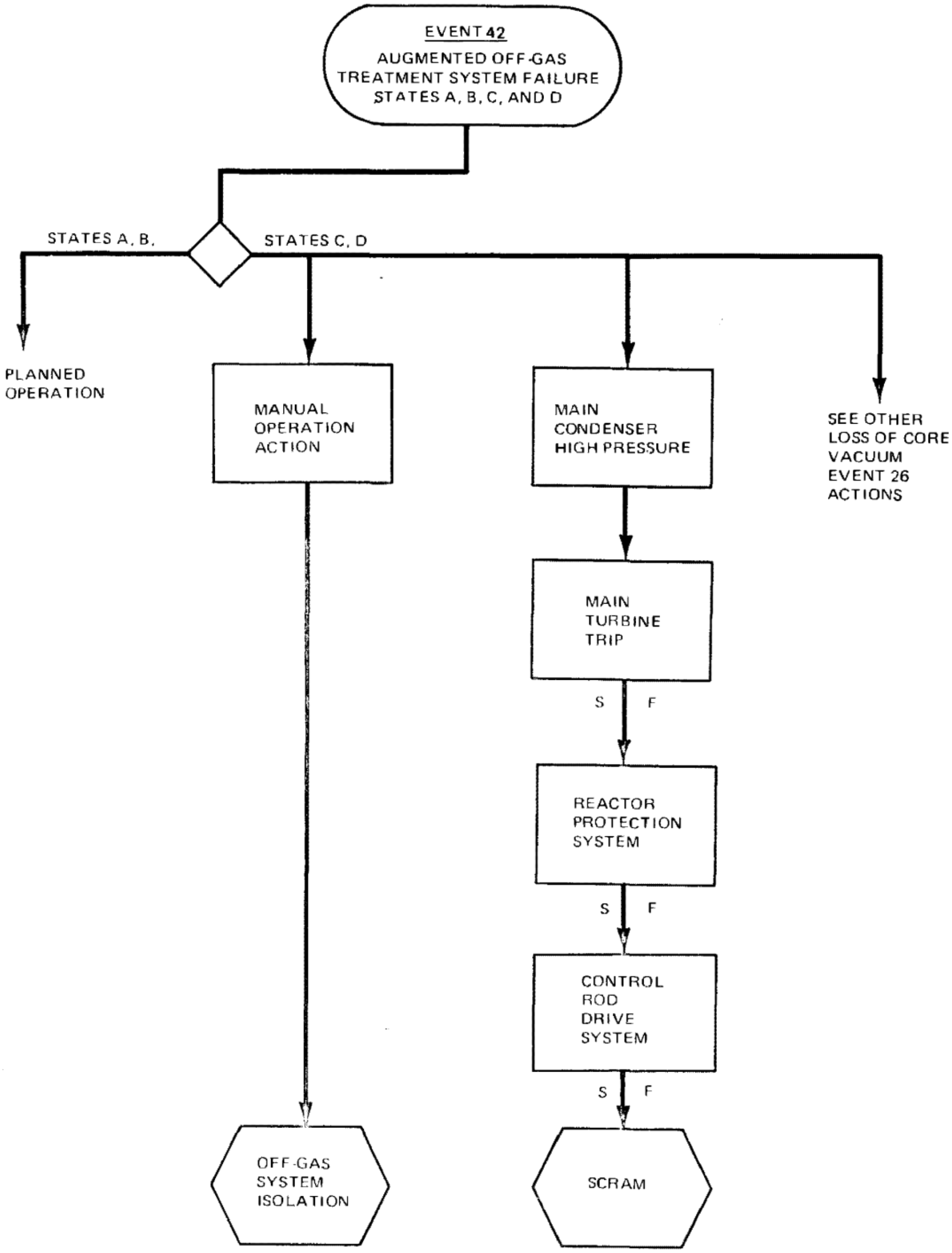
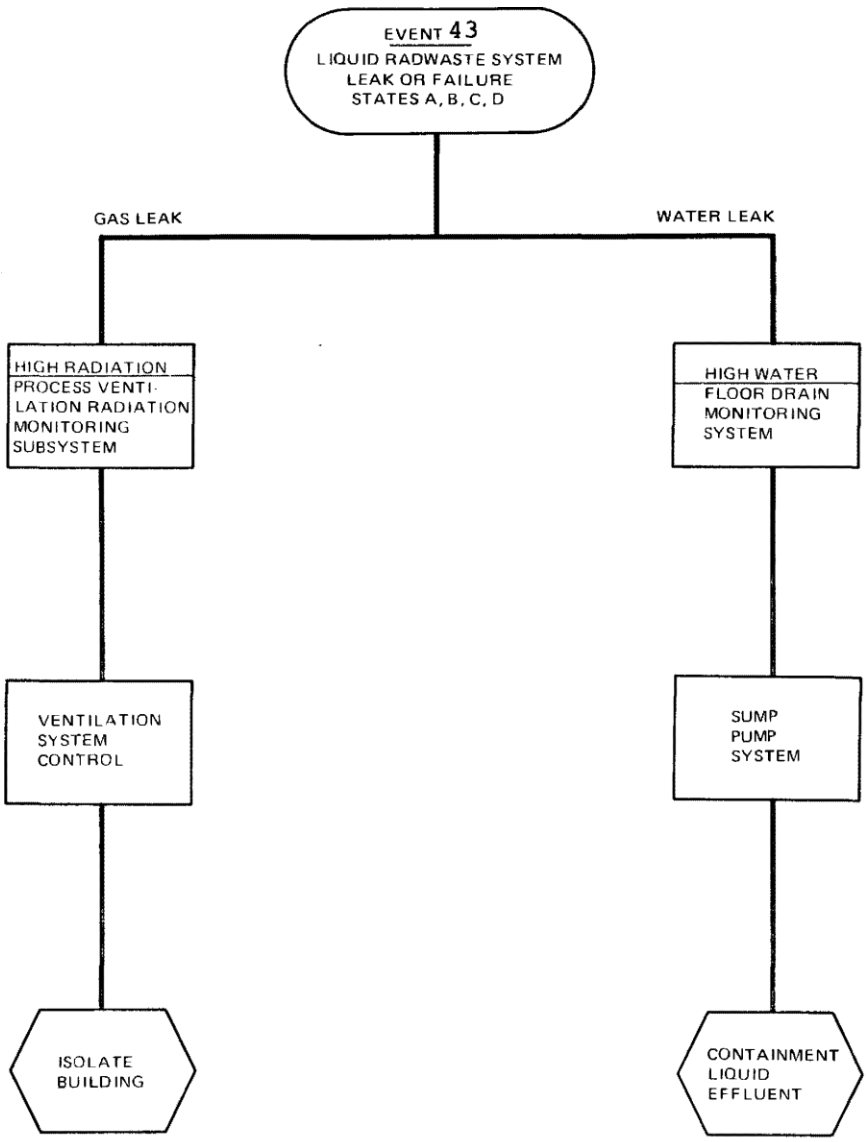
PROTECTION SEQUENCES FOR LIQUID, STEAM, LARGE, SMALL PIPING BREAKS OUTSIDE CONTAINMENT
FIGURE 15A.6-38



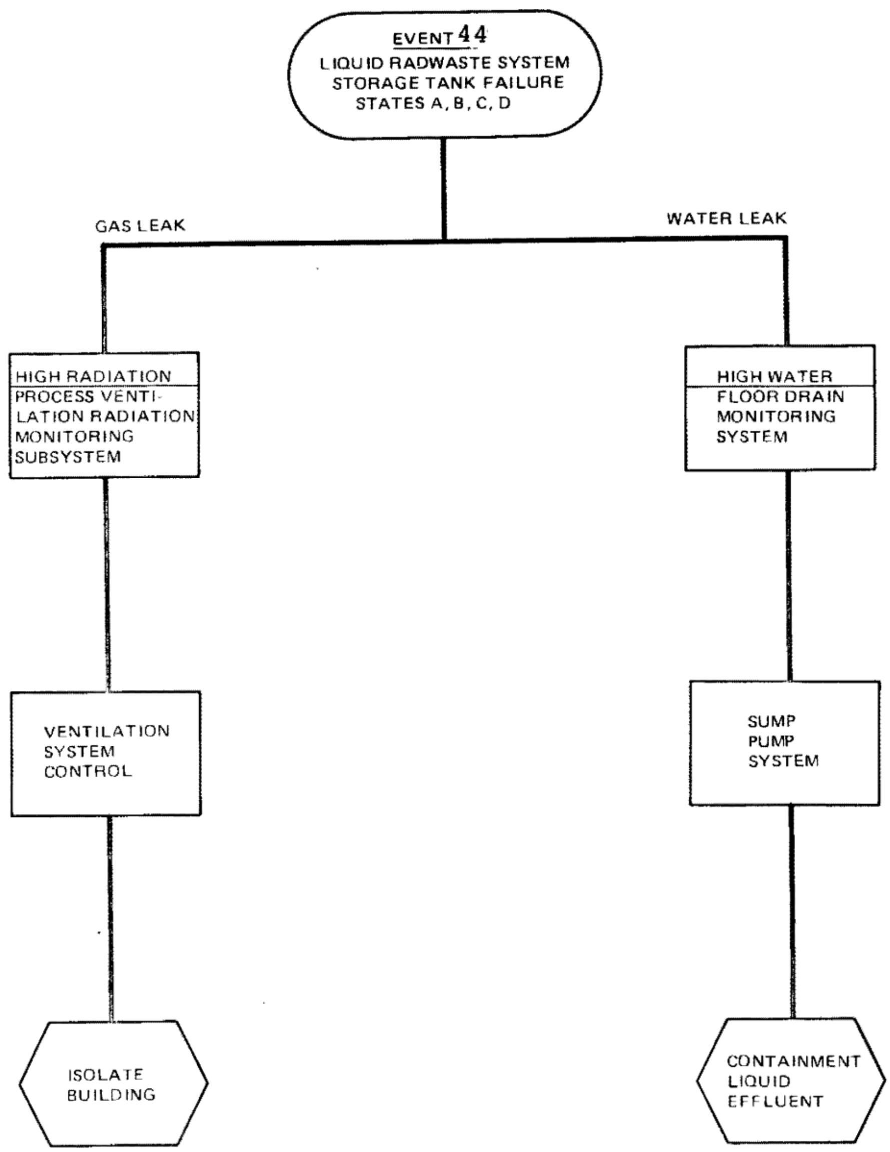
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PROTECTION SEQUENCES FOR GASEOUS RADWASTE SYSTEM LEAK OR FAILURE FIGURE 15A.6-39



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PROTECTION SEQUENCE FOR AUGMENTED OFF-GAS TREATMENT SYSTEM FAILURE FIGURE 15A.6-40

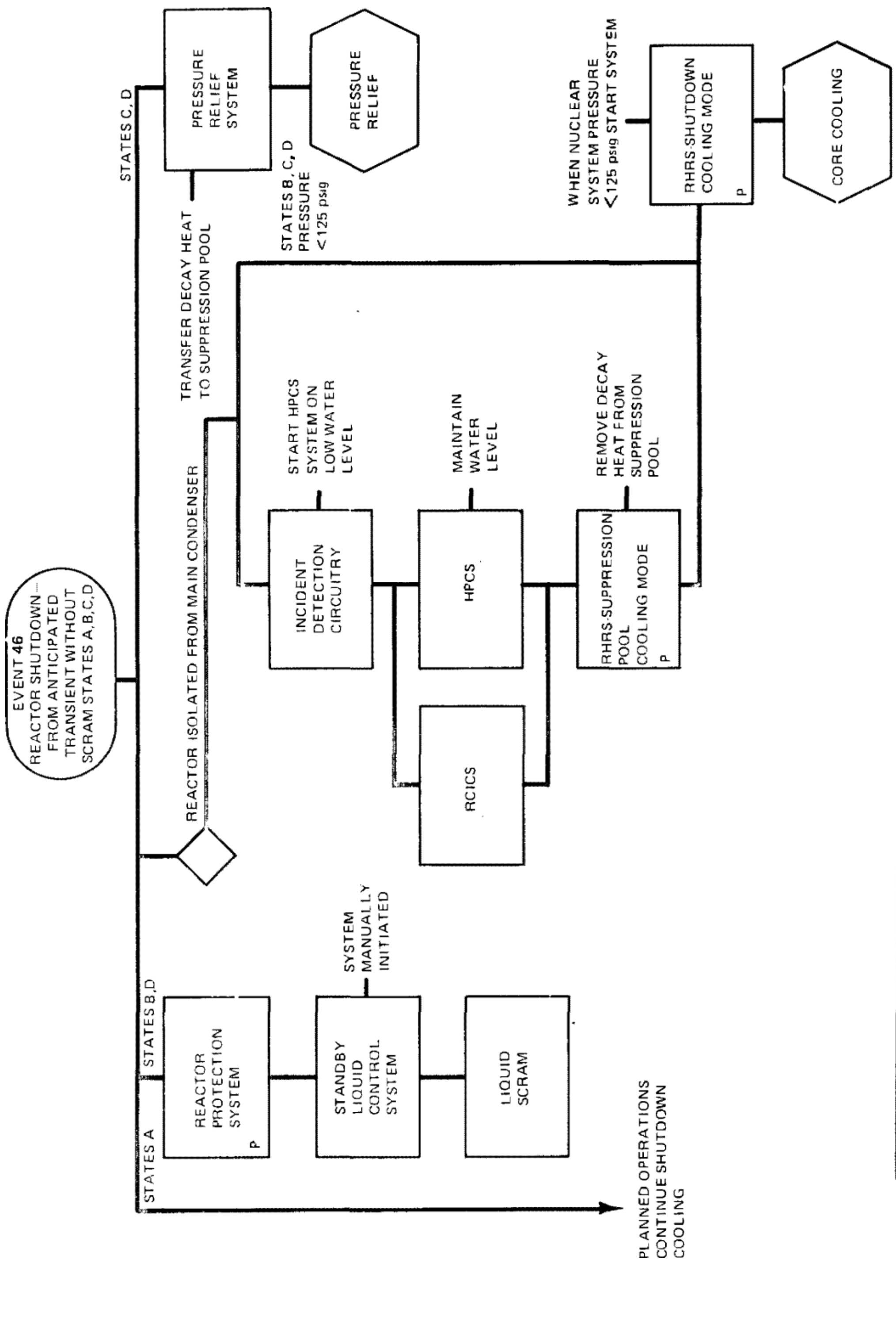


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PROTECTION SEQUENCE FOR LIQUID RADWASTE SYSTEM LEAK OR FAILURE FIGURE 15A.6-41



[HISTORICAL INFORMATION]

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PROTECTION SEQUENCE FOR LIQUID RADWASTE SYSTEM STORAGE TANK (ABANDONED IN PLACE) FAILURE FIGURE 15A.6-42



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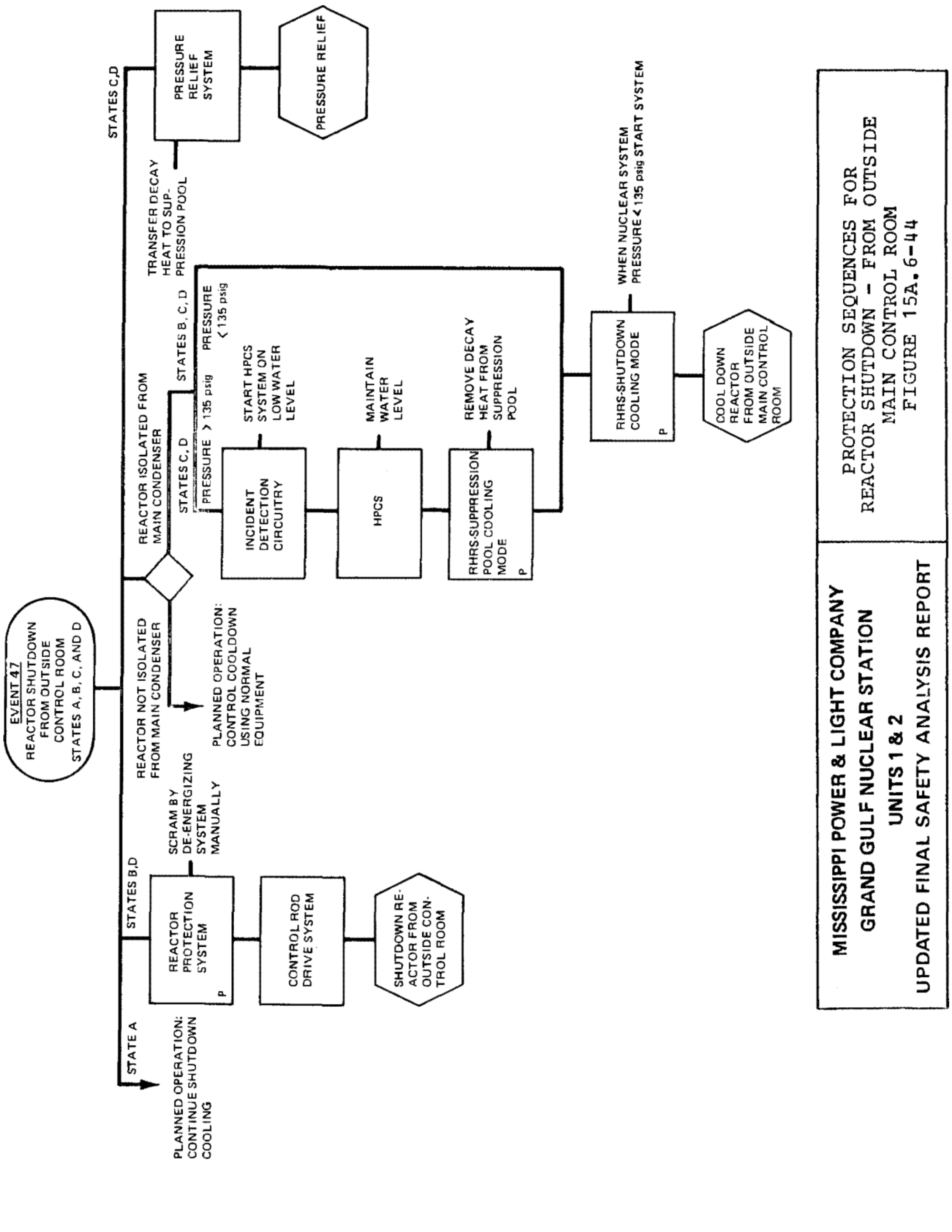
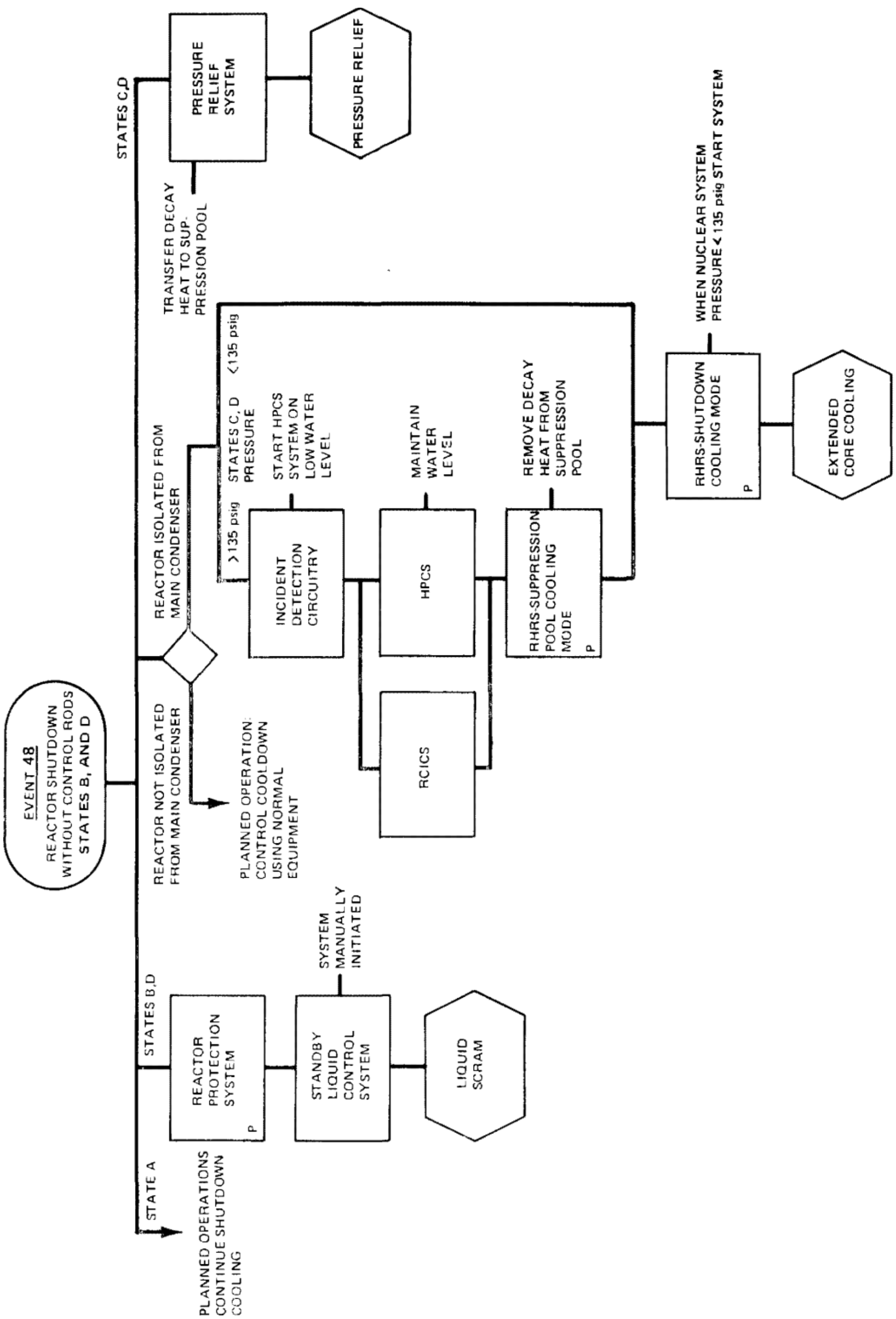
PROTECTION SEQUENCE FOR REACTOR SHUTDOWN - FROM ANTICIPATED TRANSIENT WITHOUT SCRAM
FIGURE 15A.6-43



PROTECTION SEQUENCES FOR REACTOR SHUTDOWN - FROM OUTSIDE MAIN CONTROL ROOM
FIGURE 15A.6-44



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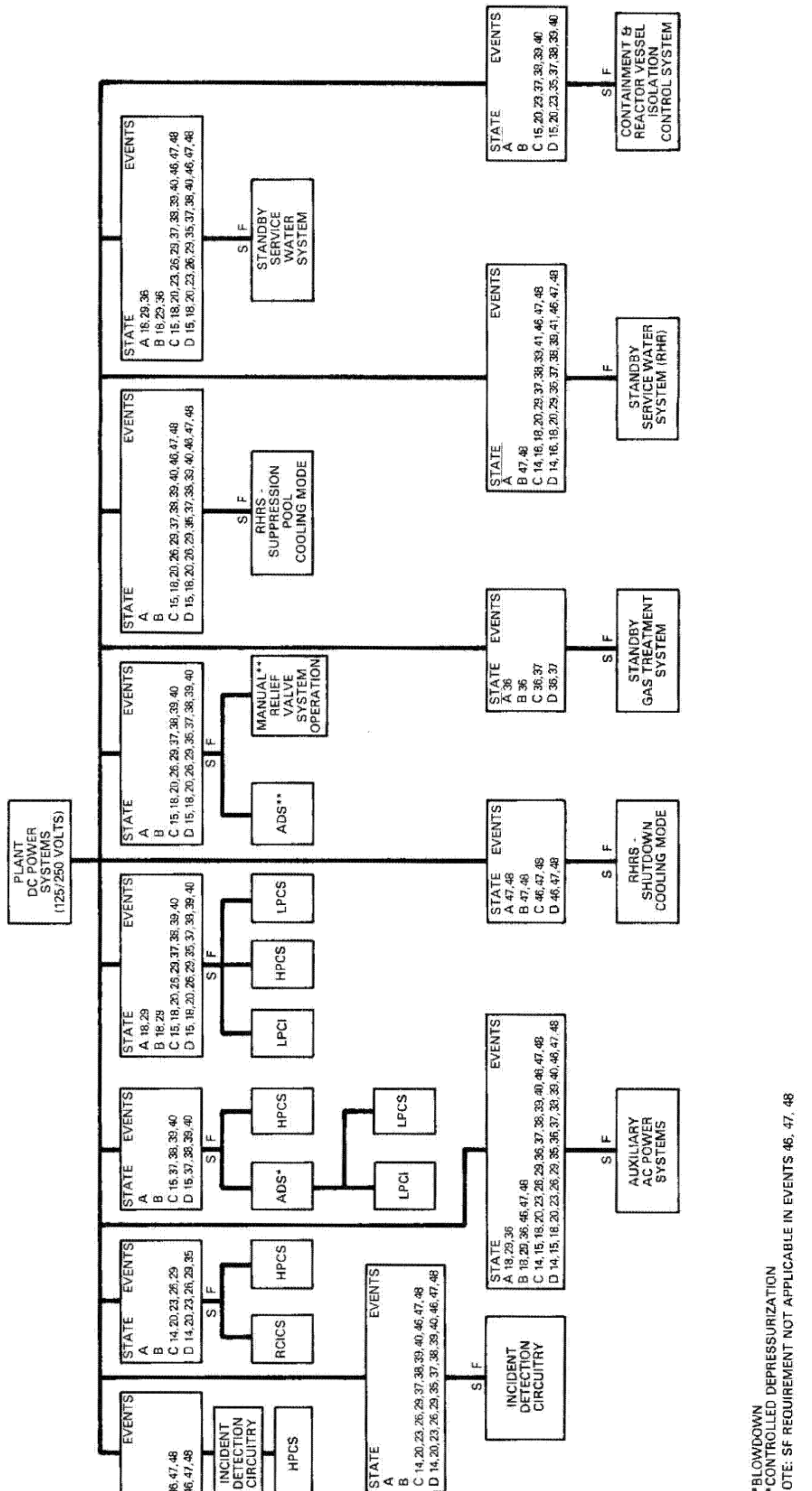
PROTECTION SEQUENCE FOR REACTOR SHUTDOWN - WITHOUT CONTROL RODS
FIGURE 15A.6-45



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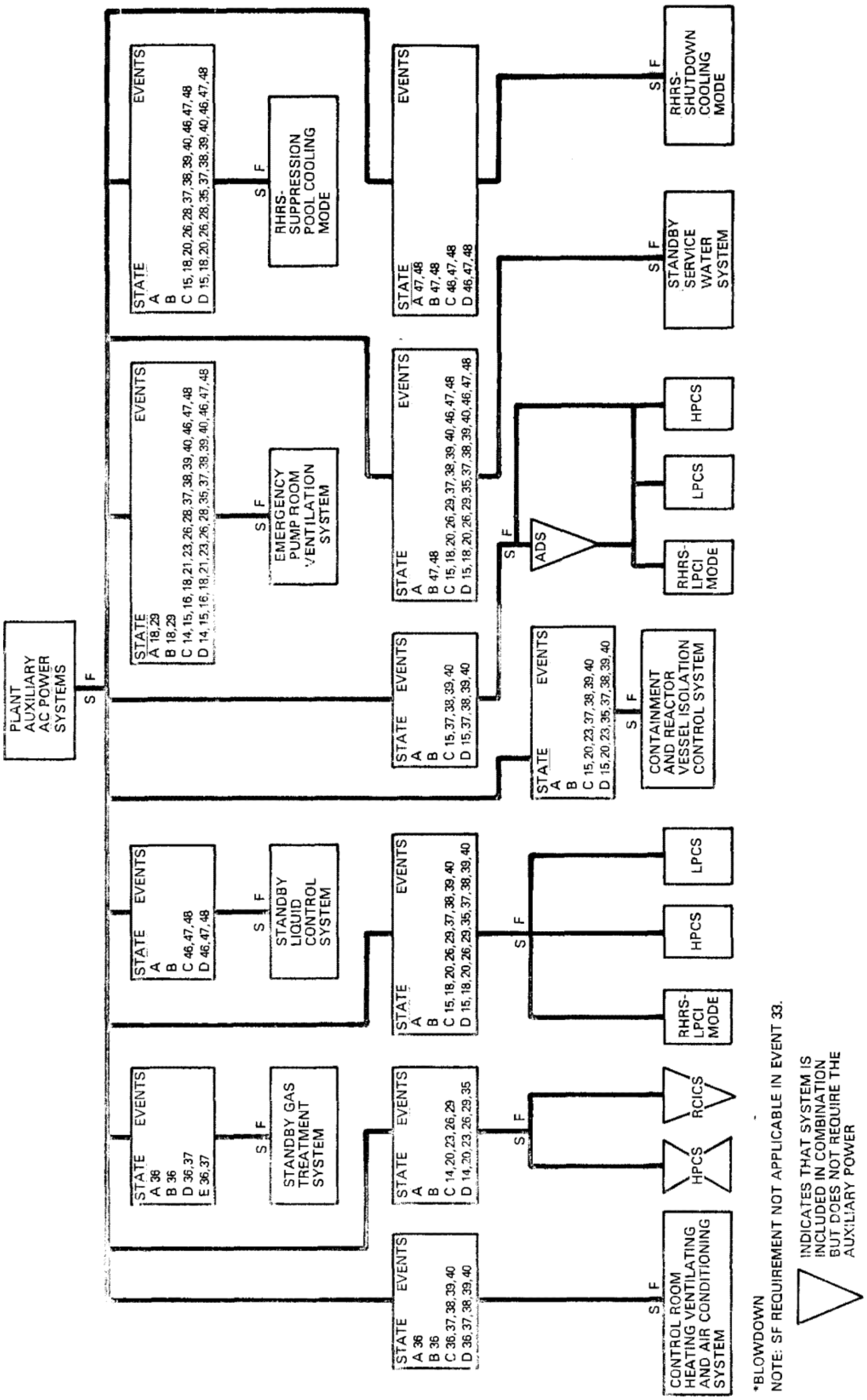
COMMONALITY OF AUXILIARY SYSTEMS – DC POWER SYSTEMS (125/250 VOLTS)
FIGURE 15A.6-46



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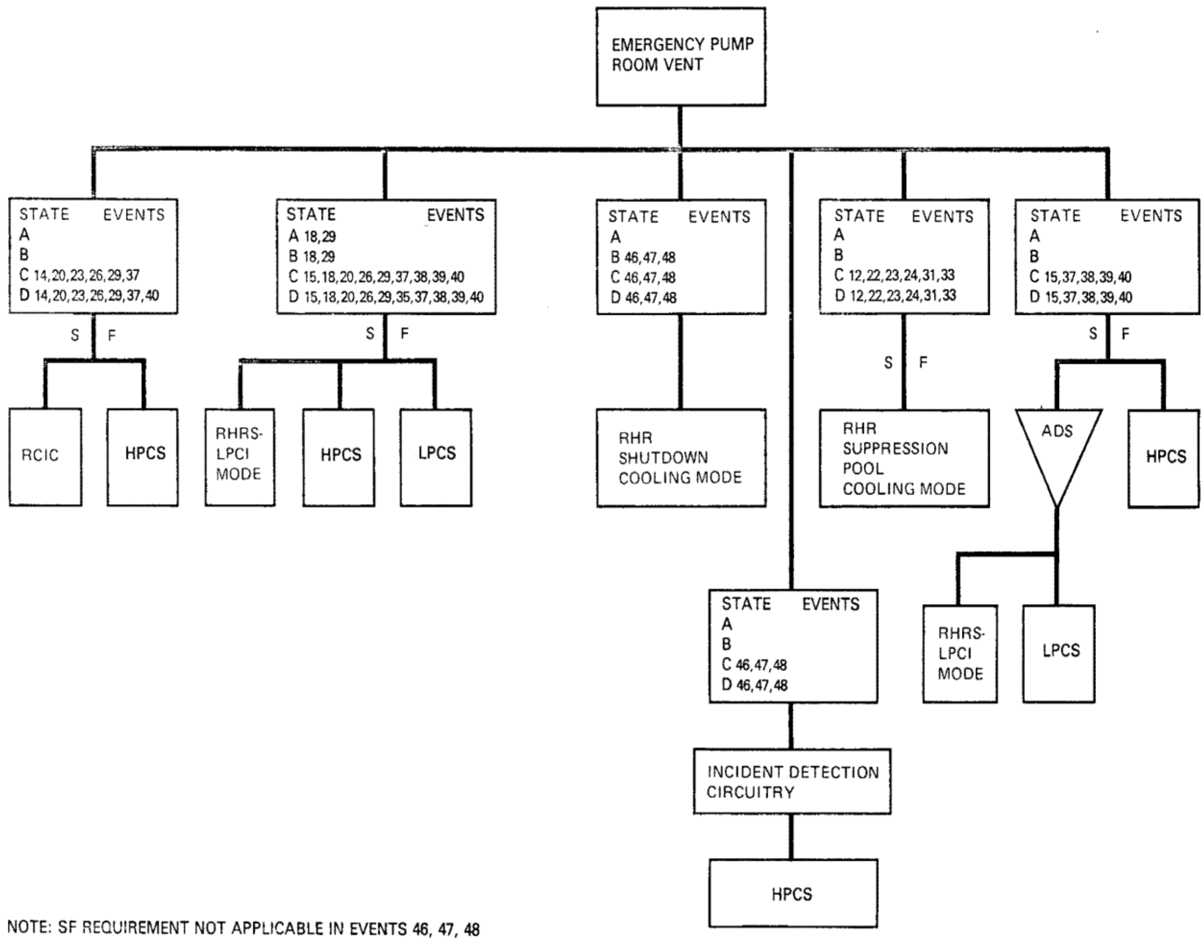
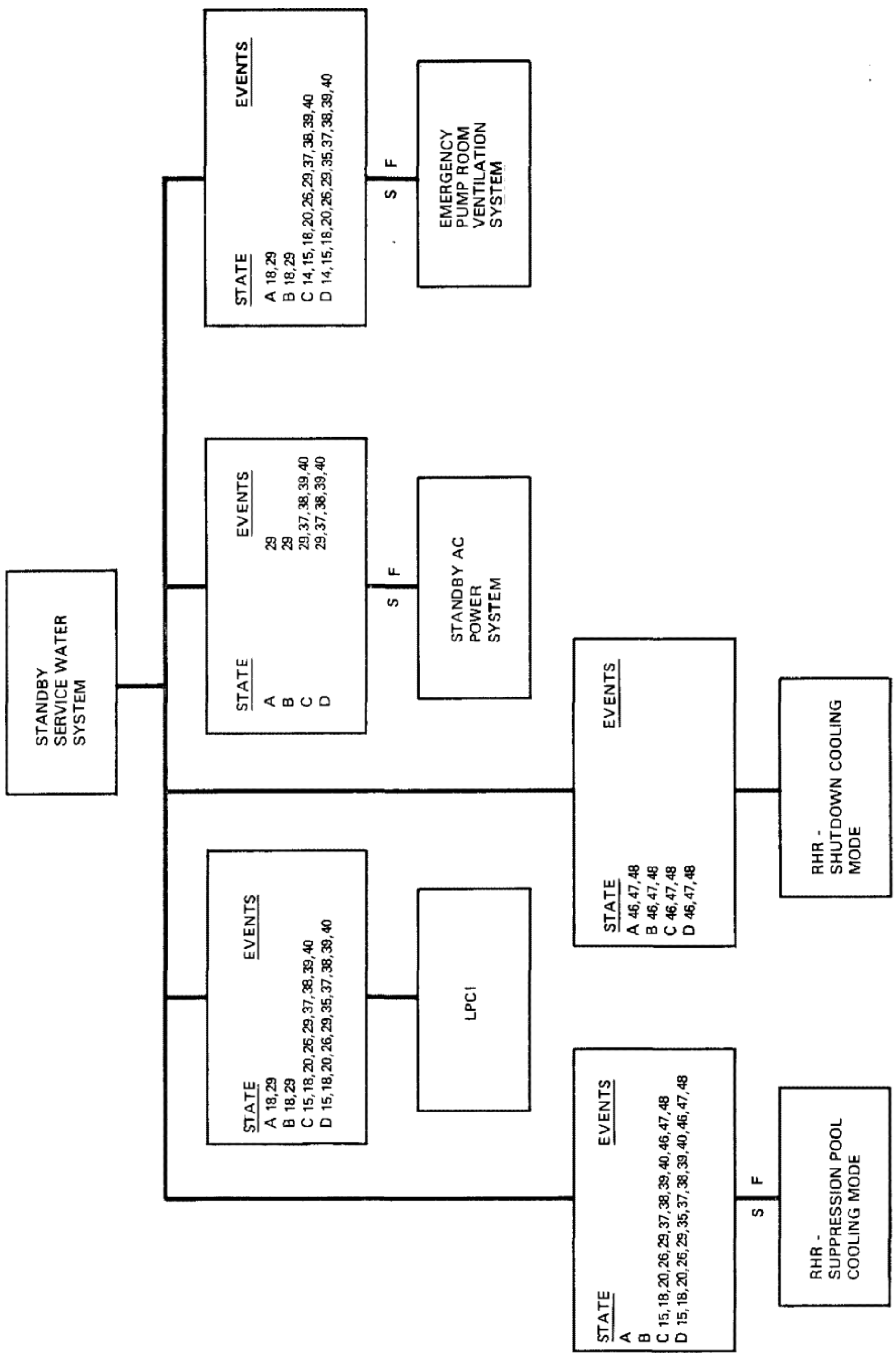
COMMONALITY OF AUXILIARY SYSTEMS – AC POWER SYSTEMS
(120/480/4100 VOLTS)
FIGURE 15A.6-47



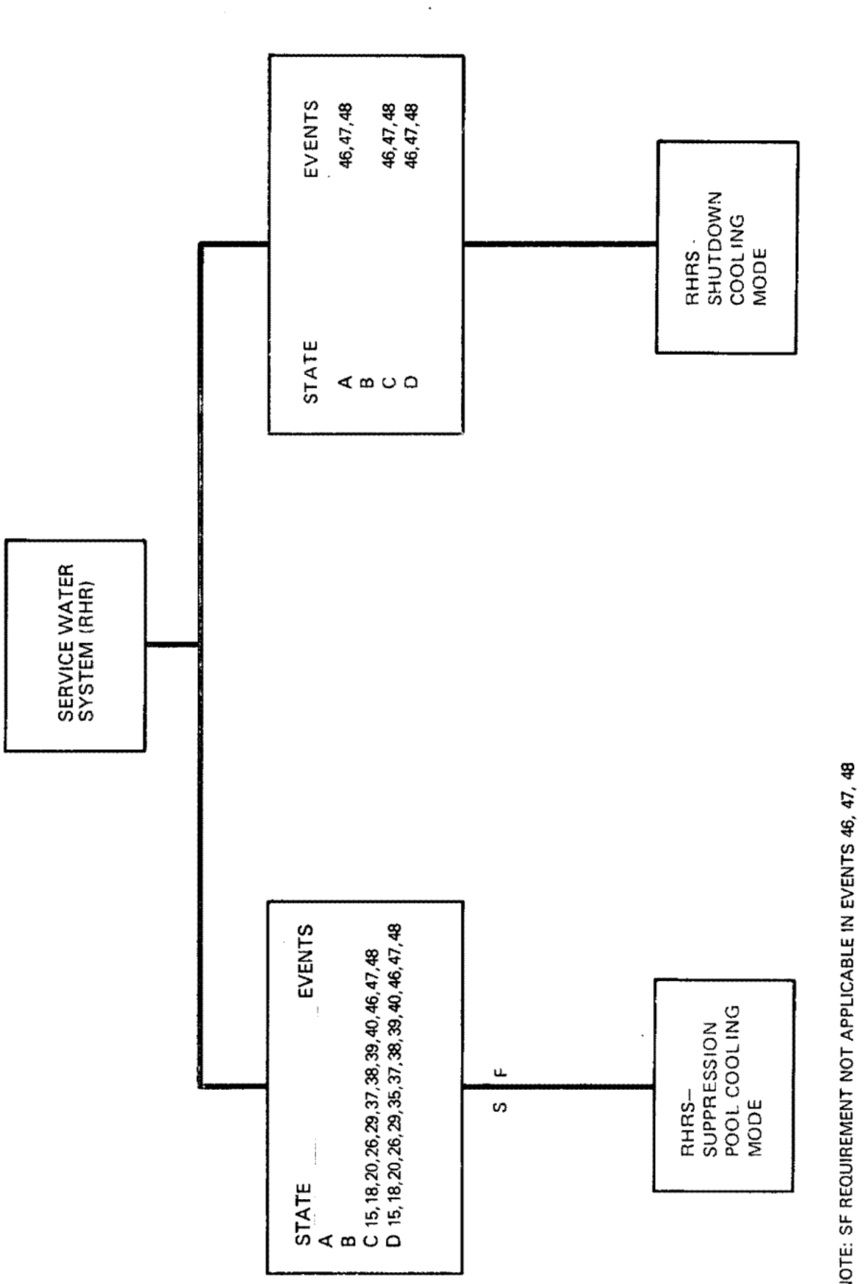
NOTE: SF REQUIREMENT NOT APPLICABLE IN EVENTS 46, 47, 48

MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	COMMONALITY OF AUXILIARY SYSTEMS - EMERGENCY PUMP ROOM VENTILATION SYSTEM FIGURE 15A.6-48



NOTE: SF REQUIREMENT NOT APPLICABLE IN EVENTS 46, 47, 48

MISSISSIPPI POWER & LIGHT COMPANY **GRAND GULF NUCLEAR STATION** **UNITS 1 & 2** **UPDATED FINAL SAFETY ANALYSIS REPORT**	COMMONALITY OF AUXILIARY SYSTEMS - STANDBY SERVICE WATER SYSTEMS FIGURE 15A.6-49



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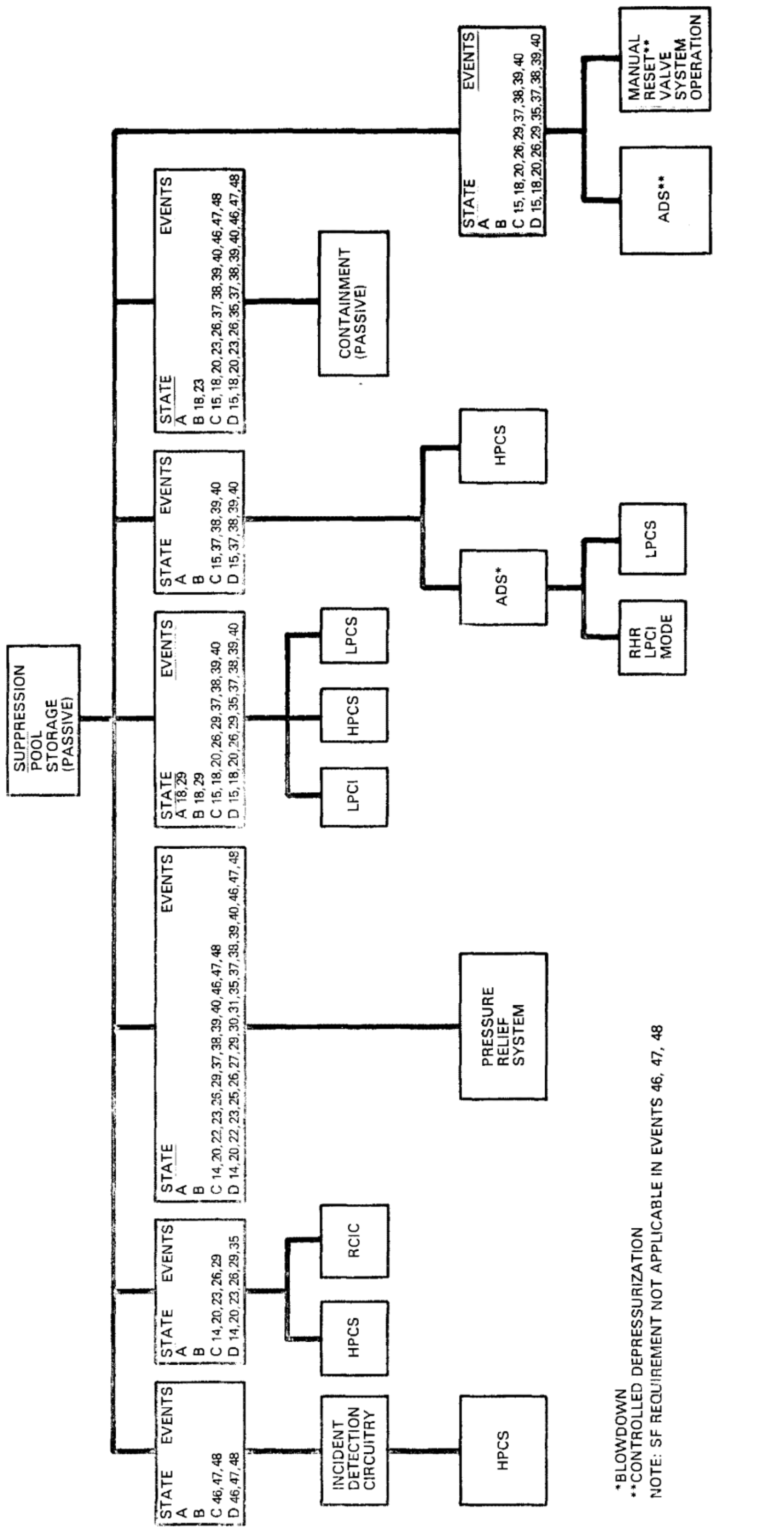
COMMONALITY OF AUXILIARY SYSTEMS - SERVICE WATER SYSTEM (RHR)
FIGURE 15A.6-50



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COMMONALITY OF AUXILIARY SYSTEMS - SUPPRESSION POOL STORAGE

FIGURE 15A.6-51

15A.7 REMAINDER OF NSOA

With the information presented in the protection sequence block diagrams, the auxiliary diagrams, and the commonality of auxiliary diagrams, it is possible to determine the exact functional and hardware requirements for each system. This is done by considering each event in which the system is employed and deriving a limiting set of operational requirements. This limiting set of operational requirements establishes the lowest acceptable level of performance for a system or component, or the minimum number of components or portions of a system that must be operable in order that plant operation may continue.

The operational requirements derived using the above process may be complicated functions of operating states, parameter ranges, and hardware conditions. The final step is to simplify these complex requirements into technical specifications that encompass the operational requirements but are easily used by plant operations and management personnel.

15A.8 CONCLUSIONS

It is concluded that the nuclear safety operational and plant design basis criteria are satisfied when the plant is operated in accordance with the nuclear safety operational requirements determined by the method presented in this appendix.

15A.9 LIST OF REFERENCES

1. Hirsch, M. M., "Methods for Calculating Safe Test Intervals and Allowable Repair Times for Engineered Safeguard Systems," January 1973 (NEDO-10739).

APPENDIX 15B FEEDWATER HEATER(S) OUT OF SERVICE

15B.1 INTRODUCTION AND SUMMARY

This appendix addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Feedwater Heater Out-of-Service (FWHOS) is an operational flexibility option and the input parameters related to FWHOS, when limiting, have been considered in the safety analyses for EPU and in the core reload analyses (COLR), as applicable. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [This appendix presents results from a safety and impact evaluation for the operation of the Grand Gulf Nuclear Stations (GGNS) during the initial operating cycle with feedwater heater(s) out of service (Reference 7). The evaluation supports operation within the power flow region as illustrated in Figure 15D.3-2 of Appendix 15D. This evaluation is performed for the GE-6 fueled GGNS at initial cycle with a target Haling end of cycle exposure distribution. The condition of operation considered in the evaluation is that of continued 100% thermal power operation during the normal operation cycle with a maximum rated feedwater temperature reduction of up to 100°F (licensed for 50°F) due to feedwater heater(s) out of service. This evaluation is used to justify GGNS continued operation during the initial operating cycle between 420°F and 370°F feedwater temperature at rated power.]

GGNS is currently licensed to EPU and MELLA+ conditions and to the FWHOS as an operation flexibility option. The FWHOS evaluation is performed every fuel cycle and was re-evaluated when GGNS changed to the GNF2 fuel due to impact to the Minimum Critical Power Ratio (MCPR). The FWHOS is licensed for 50°F feedwater temperature reduction, or a feedwater heater temperature range between 420°F and 370°F. In addition, operation with feedwater heaters out of service is prohibited while in the MELLLA+ operating domain.

Operation at a reduced feedwater temperature occurs in the event that certain stage(s) or string(s) or individual heaters become inoperable. Loss of feedwater heating from the highest pressure heaters would result in the highest temperature reduction. Loss of heating from the low pressure heaters would result in only a slight reduction of feedwater temperature. Chapter 15 has already evaluated the transient response for the worst feedwater temperature loss up to 100°F during an operating cycle due to inoperable, out of service or unavailable heater stages. Therefore, this appendix will justify the operation for GGNS with the rated feedwater temperatures between 420°F and 370°F due to inoperable feedwater heater(s).

This evaluation is termed Feedwater Heater(s) out of service (FWHOS) in the remaining content of this section. The only adjustment for the FWHOS operating condition is to change the RPS scram function on the turbine stop valve closure and the turbine control valve closure to assure that the scram bypass is consistent with the 35.4% of rated power in FWHOS conditions. No operating limit MCPR change needs to be made for operation between 420°F and 370°F rated feedwater temperature.

The following evaluations and conclusions resulted:

a. The abnormal operating transients in Chapter 15 were reevaluated at rated feedwater temperature of 370°F to determine the required operating MCPR limits for FWHOS operation. The results show that no operating limit MCPR change is required for operation between 420°F and 370°F rated feedwater temperature.

b. It is determined that the fuel mechanical limits are met during FWHOS operation under steady state and anticipated operational occurrences.

c. The Loss of Coolant Accident (LOCA) and containment response as described in Chapter 6 were reevaluated for FWHOS operation. It is found that the normal feedwater temperature analysis adequately bound those events with FWHOS conditions.

d. Fuel integrity thermal-hydraulic stability was evaluated with respect to General Design Criterion 12 (10CFR50, Appendix A). It is shown that the FWHOS operation satisfies the stability criterion and fuel integrity is not compromised.

e. The effect of acoustic and flow induced loads on the reactor shroud, shroud support and jet pumps were analyzed to show that the design limits are not exceeded. The effect of FWHOS on feedwater nozzle and sparger fatigue usage factor was determined. It was found that the increased fatigue usage on the feedwater nozzle adequately meets the acceptance criterion for unlimited operation up to 40 years. The increased fatigue usage on the feedwater sparger meets the acceptance criterion with some limitation on the maximum allowable number of days for FWHOS operation. These specific FWHOS operation time limits are of economic concern only.

f. The turbine stop valve and the turbine control valve scram bypass setpoints in the Reactor Protection System are contained in the Technical Specifications (TS).

There are also other impact evaluations performed such as feedwater system piping, annulus pressurization load analysis, and Anticipated Transients Without Scram (ATWS) to justify FWHOS operation. These evaluations concluded that the standard operation design is adequate for FWHOS operation.

15B.2 MCPR OPERATING LIMIT

15B.2.1 Abnormal Operating Transients

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [All abnormal operating transients described in Chapter 15 were examined for FWHOS operation. Three limiting abnormal operating transients were reanalyzed in detail with a bounding BWR 6 standard plant at 104.2% power at both 75% and 110% core flow and reevaluated for GGNS at about 110% core flow. These analyzed transients are:

a. Generator Load Rejection with Bypass Failure (LRNBP)

b. Feedwater Flow Controller Failure (FWCF)

c. Loss of 100°F Feedwater Heating (LFWH)

The GGNS specific LRNBP and FWCF transients were evaluated at 104.2% of the initially licensed power and about 110% core flow with rated feedwater temperature at 370°F at end of cycle 1 and

2000 MWD/T exposure before end of cycle. The 75% core flow analysis was not performed for GGNS because the bounding BWR 6 analysis performed at 370°F rated feedwater temperature has shown significant margin existed at this condition. Plant heat balance, core coolant hydraulic and nuclear transient data were developed and used in the analysis. The initial conditions for the analysis points are presented in Table 15B.2-2.

The end of cycle exposure point with all control rods fully withdrawn is a limiting point in the cycle with the worst scram worth reactivity characteristics. The 2000 MWD/T before end of cycle exposure point is chosen as an analyzed point because it is close enough to end of cycle such that the scram characteristics have not been significantly improved relative to earlier points in the cycle and the void reactivity characteristic is different than end of cycle. Scram characteristics are significantly improved at exposures lower than this point, and the transient responses will be bounded by the two points analyzed.

The computer model described in Reference 1 was used to simulate both of these events. The transient peak value results and critical power ratio results are summarized in Tables 15B.2-3 to 15B.2-6. The transient responses for these end of cycle and mid cycle cases are presented in Figures 15B.2-1, 15B.2-3, 15B.2-5, and 15B.2-7.

The bounding BWR 6 analysis and the GGNS specific analysis shown on Tables 15B.2-5 and 15B.2-6 indicate that the ΔCPR for the worst feedwater controller failure transient does not exceed the standard operating limit MCPR for operation at reduced rated feedwater temperature down to 370°F. Therefore, the OLMCPR value does not need to be changed between 420 and 370°F rated FWHOS operation.

Lower initial operating pressure and steam flow rate provide better overpressure protection for the most limiting MSIV closure (flux scram) event during FWHOS operation. Tables 15B.2-3 and 15B.2-4 also indicate that the peak pressures for the LRNBP and FWCF events analyzed are below the ASME code value of 1375 psig. Hence, it is concluded that the pressure barrier integrity is maintained under FWHOS operation conditions.

The 100°F loss of feedwater heating transient was evaluated for a bounding BWR 6 plant at initial feedwater temperatures of 250°F (to bound all FWHOS operation) and 420°F at 104.2% power, 100%

core flow at the end of the cycle using the computer model described in Reference 3 and methodology described in Reference 4. Results show that the 100°F LFWH has less effect on colder feedwater than on the normal feedwater temperature. The ΔCPR result for the 100°F loss initiated from 250°F is bounded by the 420°F initiation case. The generic LFWH analysis described in Reference 4 concluded that this event is insensitive to initial core flow. It is less likely that a 100°F loss would occur at an initial feedwater temperature of 370°F during FWHOS operation. A generic statistical LFWH analysis using the same model and methodology described above utilizes a large data base of LFWH cases to generate the 95% probability, 95% confidence bounding ΔCPR value for the normal 420°F condition. This data base includes transient responses at different exposure points throughout the operating cycle. Therefore, the loss of feedwater heating analysis for FWHOS is adequately bounded by the 420°F ΔCPR results.]

15B.2.2 Rod Withdrawal Error

The rod withdrawal error (RWE) transient documented in Chapter 15 is analyzed using a statistical evaluation of the minimum critical power ratio (MCPR) and Linear Heat Generation Rate (LHGR) response to the withdrawal of ganged control rods from both rated and off rated conditions over the entire operating region. Therefore, this analysis covers a wide variety of feedwater temperatures and core subcooling as different off rated conditions are included in the data base. The 95% probability 95% confidence values from this statistical data base are used to develop the Rod Withdrawal Limiter (RWL) system setpoints to protect against a rod withdrawal error.

The rod withdrawal error analysis does not need to be evaluated for FWHOS at end of cycle because all control rods will be fully withdrawn. A RWE analysis was performed at 2000 MWD/T before end of cycle to examine the effect of the initial feedwater temperature. An initial condition of 250°F was used to bound all FWHOS operation. Results show that ΔCPR resulting from the worst 2 feet of withdrawal for the 420°F and 250°F feedwater temperature are identical. Therefore, the ΔCPR values initiating from 250°F feedwater temperature condition fall within the statistical data base used to establish the RWL system setpoints. The change in linear heat generation rate is bounded by the fuel mechanical design bases. Therefore, it is concluded that operating limit MCPR does not need to be increased due to RWE for FWHOS operation.

15B.2.3 Operating Limit MCPR

For FWHOS operation between 420°F and 370°F rated feedwater temperature, the OLMCPR does not need to be changed. The off-rated power dependent $MCPR_p$ limits also do not need to be changed to cover FWHOS operation at this rated temperature range. The off-rated flow dependent $MCPR_f$ limits do not need to be changed because the current $MCPR_f$ limit curve was generated based on a steepest power flow rod line to protect against the recirculation flow runout transient. A power flow rod line was generated for a conservatively rated initial condition. It shows that the slope of this rod line is bounded by the current design basis rod line. Therefore, the current $MCPR_f$ limits are valid for FWHOS operation above 370°F. The current cycle operating limit MCPR is evaluated in Reference 8.

TABLE 15B.2-1:(SHEETS 1 OF 3 THRU 3 OF 3)

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TABLE 15B.2-2:[HISTORICAL INFORMATION] INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS AND ACCIDENTS FOR FWHOS, 373°F FWT 104.2% POWER, 109.6% FLOW (INITIAL CYCLE)*

1. Thermal Power Level, MWt Analysis Value	3994 (104.2% rated)*
2. Steam Flow, mlb per hr Analysis Value	16.10
3. Core Flow, mlb per hr	123.3
4. Feedwater Flow Rate, mlb per hr Analysis Value	16.10
5. Feedwater Temperature, °F	373
6. Vessel dome pressure, psig	1020
7. Core exit pressure, psig	1031
8. Turbine Bypass Capacity, % NBR	35
9. Core Coolant Inlet Enthalpy Btu per lb	523
10. Turbine Inlet Pressure, psig	944
11. Fuel Lattice	8x8R
12. Core Leakage Flow, %	10.65
13. Required MCPR Operating Limit First Core	1.18
14. MCPR Safety Limit for Incidents of Moderate Frequency	
First Core	1.06
Reload Cores	1.07
15. Doppler Coefficient (-)¢/°F Analysis Data	0.132(a)
16. Void Coefficient (-)¢/% Rated Voids	
Analysis Data for Power Increase Events	14.0 (a)
Analysis Data for Power Decrease Events	4.0 (a)
17. Core Average Rated Void Fraction, %	38.0
18. Jet Pump Ratio, M	2.32
19. Safety/Relief Valve Capacity, % NBR	
@1145 psig	102.4
Manufacturer	Dikker
Quantity Installed	20
20. Relief Function Delay, seconds	0.4
21. Relief Function Response Time Constant, sec.	0.1

TABLE 15B.2-2:[HISTORICAL INFORMATION] INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS AND ACCIDENTS FOR FWHOS, 373°F FWT 104.2% POWER, 109.6% FLOW (INITIAL CYCLE)* (CONTINUED)

22. Setpoints for Safety/Relief Valves	
Safety Function, psig	1175,1185,1195,1205, 1215
Relief Function, psig	1145,1155,1165,1175
23. Number of Valve Groupings Simulated	
Safety Function, No.	5
Relief Function, No.	4
24. High Flux Trip, % NBR	
Analysis Setpoint (122x1.042), % NBR	127.1
25. High Pressure Scram Setpoint, psig	1126
26. Vessel Level Trips, Feet Above Separator Skirt Bottom	
Level 8 - (L8), feet	5.88
Level 4 - (L4), feet	4.03
Level 3 - (L3), feet	2.16
Level 2 - (L2), feet	(-) 2.182
27. APRM Thermal Trip	
Setpoint, % NBR	118.8
28. RPT Delay, seconds	0.19
29. RPT Inertia Time Constant for Analysis, sec.	5
30. Total Steamline Volume, ft^3	4358

(a) These values for Reference 2 analysis only. Reference 1 values are calculated within the code.

TABLE 15B.2-3:[HISTORICAL INFORMATION] SUMMARY OF GGNS TRANSIENT PEAK VALUES RESULTS - FWHOS - EOC[a]

Transient	Core Flow (% NBR)	Peak Neutron Flux (% NBR)	Peak Dome Pressure (psig)	Peak Vessel Pressure (psig)	Peak Steam-line Pressure (psig)	Fdwtr Temp. (°F)
Load Rejection With Bypass Failure	109.6[b]	162	1194	1224	1195	373
Feedwater Controller Failure, Max. Demand	109.6[b]	120.3	1150	1175	1149	373

(a) Initial power is 104.2% of initially licensed NBR for analysis
(b) Maximum achievable core flow for the given feedwater temperature

TABLE 15B.2-4:[HISTORICAL INFORMATION] SUMMARY OF TRANSIENT PEAK VALUES RESULTS - FWHOS 2000 MWD/T BEFORE EOC1[a]

Transient	Core Flow (% NBR)	Peak Neutron Flux (% NBR)	Peak Dome Pressure (psig)	Peak Vessel Pressure (psig)	Peak Steam-line Pressure (psig)	Fdwtr Temp. (°F)
Load Rejection With Bypass Failure	109.6[b]	194.3	1185	1216	1189	373
Feedwater Controller Failure, Max. Demand	109.6[b]	118.2	1136	1160	1135	373

(a) Initial power is 104.2% of initially licensed NBR for analysis
(b) Maximum achievable core flow for the given feedwater temperature

Table 15B.2-5:[HISTORICAL INFORMATION] SUMMARY OF CPR RESULTS - FWHOS - EOC1

Transient	Core Flow (% NBR)	ICPR[b]	ΔCPR	MCPR	Feedwater Temperature (°F)
Load Rejection With Bypass Failure	109.6[a]	1.18	0.05	1.13	373
Feedwater Controller Failure, Max. Demand	109.6[a]	1.18	0.11	1.07	373

(a) Maximum achievable core flow for the given feedwater temperature
(b) Based on initial core safety limit of 1.06, for reload cores 0.01 must be added.
(c) Analyses based on original licensed power of 3833 MW.

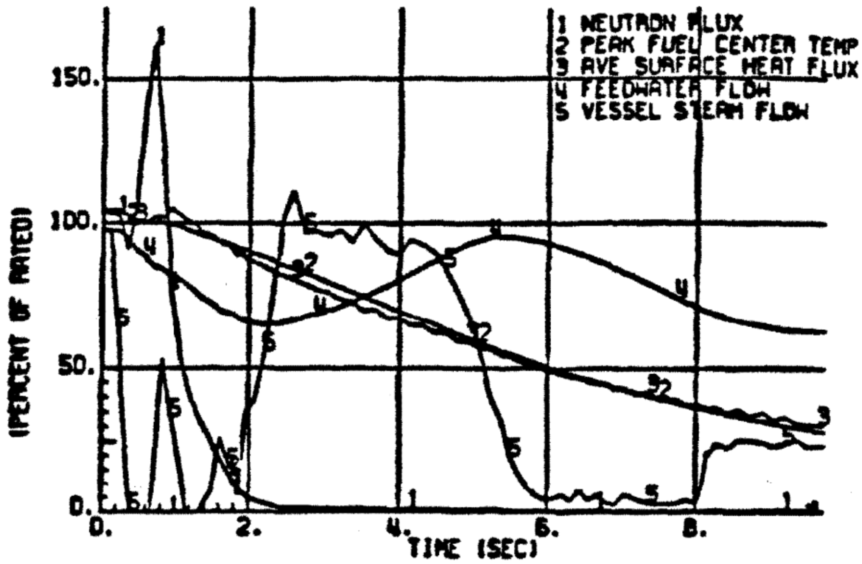
NOTE: Option A adders included.

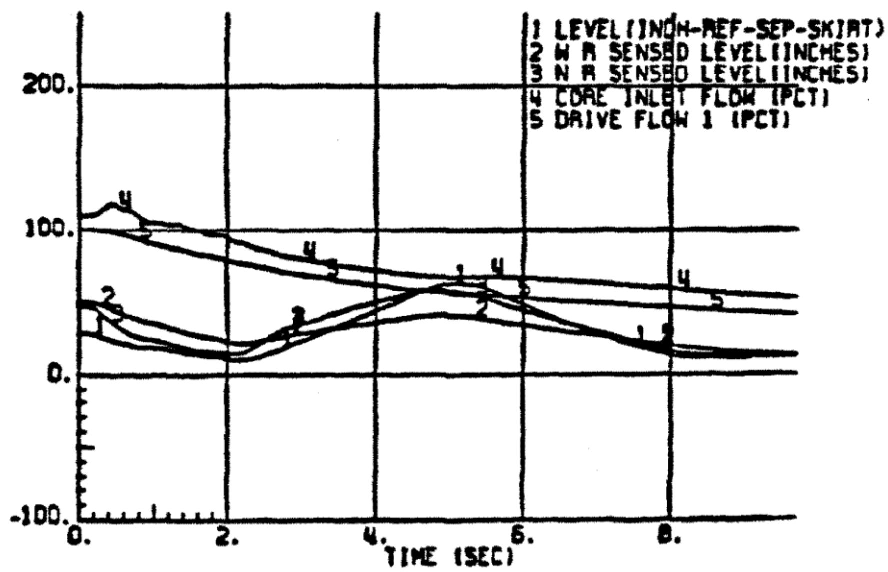
Table 15B.2-6: [HISTORICAL INFORMATION] SUMMARY OF CPR RESULTS - FWHOS - 2000 MWD/T BEFORE EOC1

Transient	Core Flow (% NBR)	ICPR[b]	ΔCPR	MCPR	Feedwater Temperature (°F)
Load Rejection With Bypass Failure	109.6[a]	1.18	0.05	1.13	373
Feedwater Controller Failure, Max. Demand	109.6[a]	1.18	0.11	1.07	373

(a) Maximum achievable core flow for the given feedwater temperature
(b) Based on initial core safety limit of 1.06, for reload cores 0.01 must be added.
(c) Analyses based on original licensed power of 3833 MW.

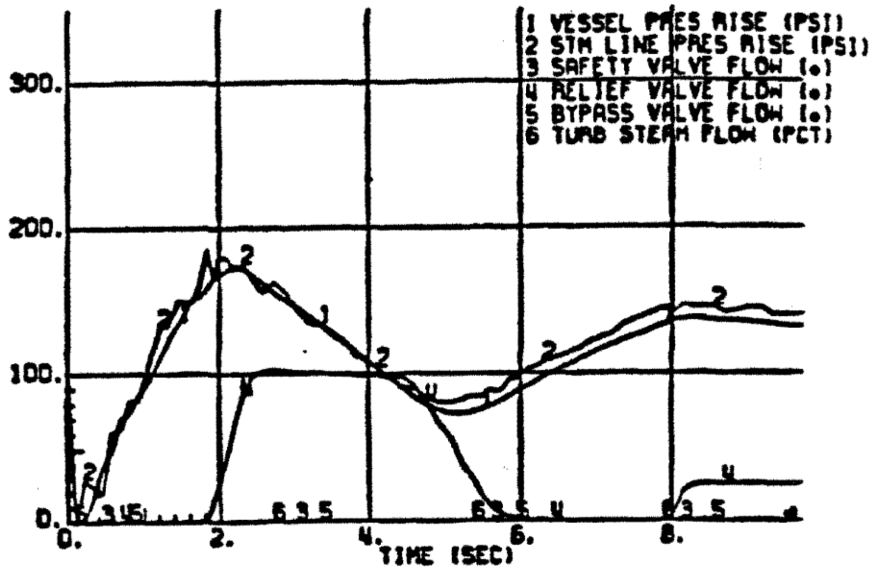
NOTE: Option A adders included.

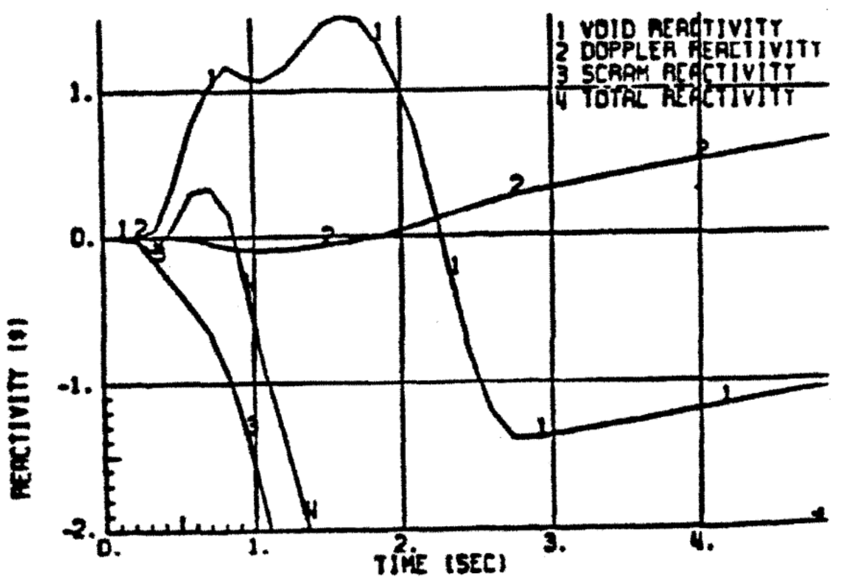




GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER, 109.6% FLOW, 373° F TFW EOC1 [HISTORICAL INFORMATION] FIGURE 15B.2-1 SHEET 1 OF 2

Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.





GRAND GULF NUCLEAR STATION UNIT 1 *UPDATED FINAL SAFETY ANALYSIS REPORT*	LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER, 109.6% FLOW, *373° F TFW EOC1* [HISTORICAL INFORMATION] FIGURE 15B.2-1 SHEET 2 OF 2

Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

Figure 15B.2-2
(Sheets 1 of 2 thru 2 of 2)

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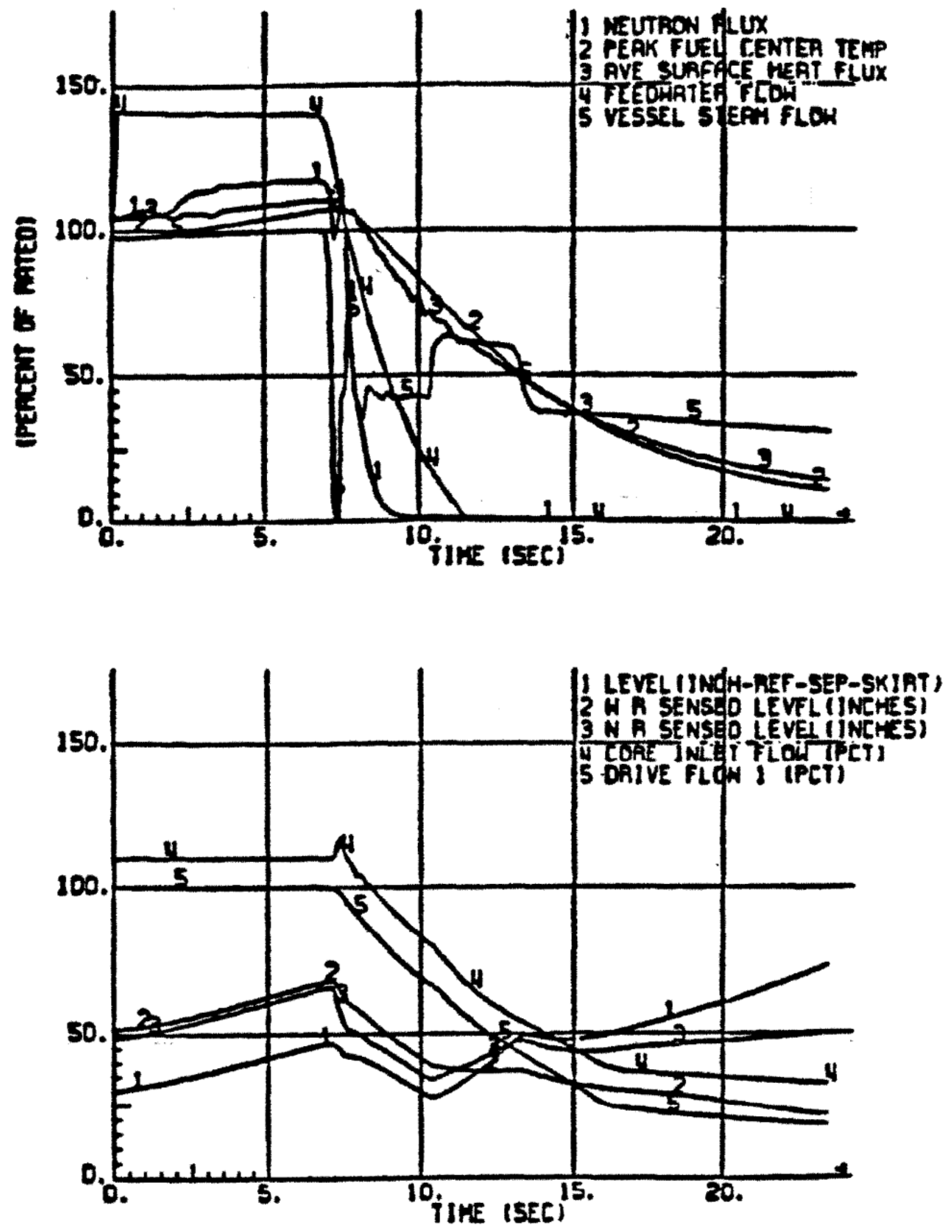
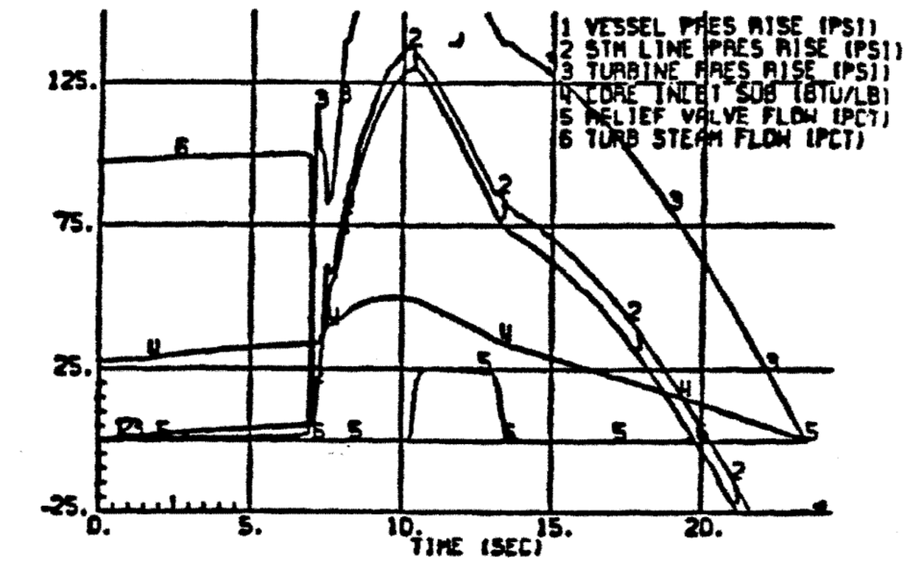
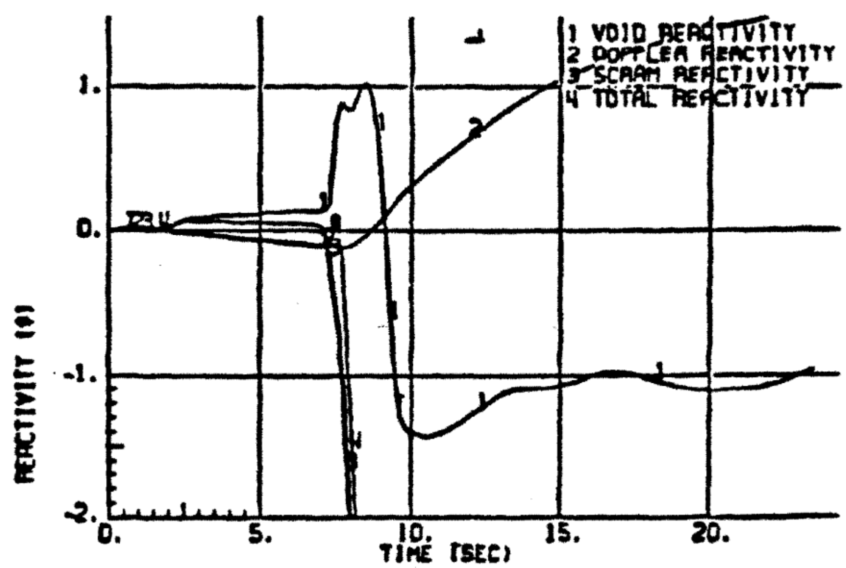


Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE 104.2% POWER, 109.6% FLOW, 373° F TFW EOC1 [HISTORICAL INFORMATION] FIGURE 15B.2-3 SHEET 1 OF 2





GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE 104.2% POWER, 109.6% FLOW, 373° F TFW EOC1 [HISTORICAL INFORMATION] FIGURE 15B.2-3 SHEET 2 OF 2

Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

Figure 15B.2-4
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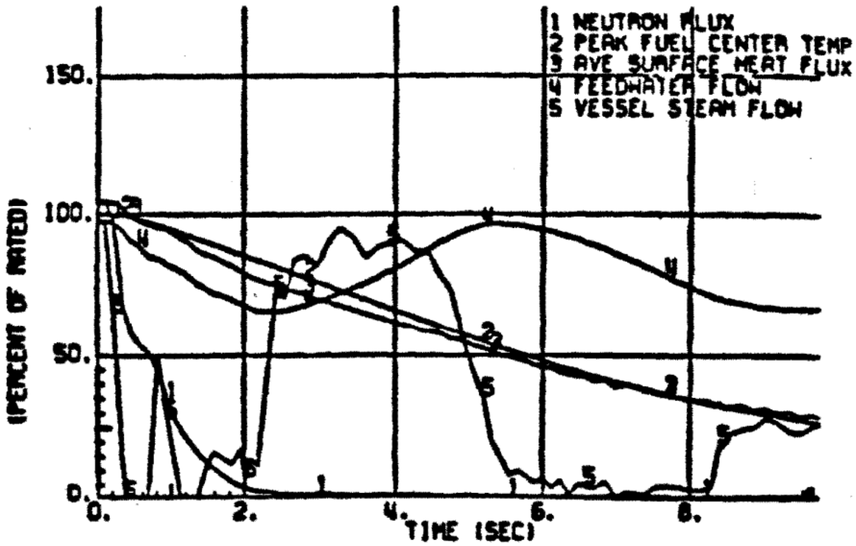
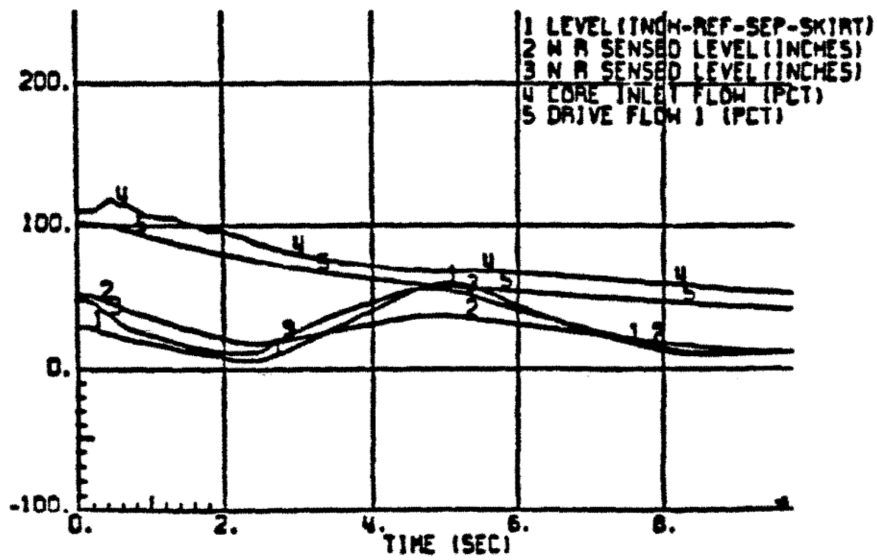
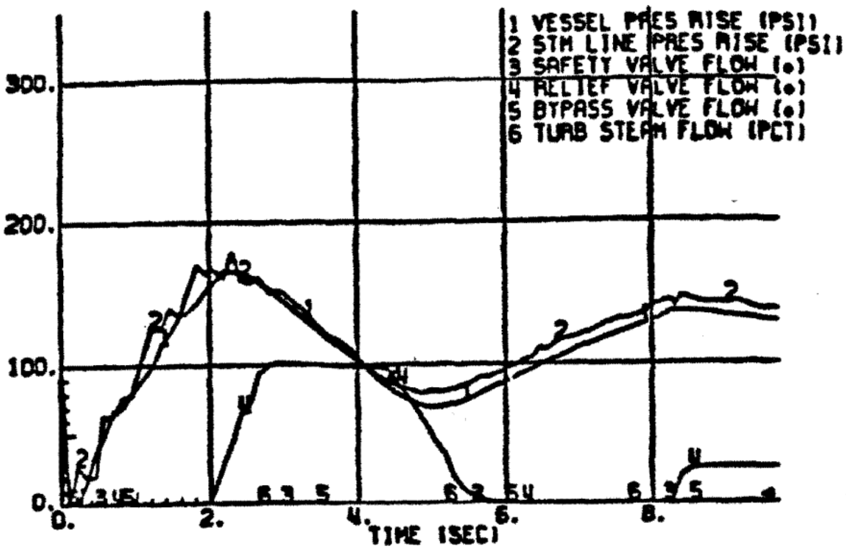
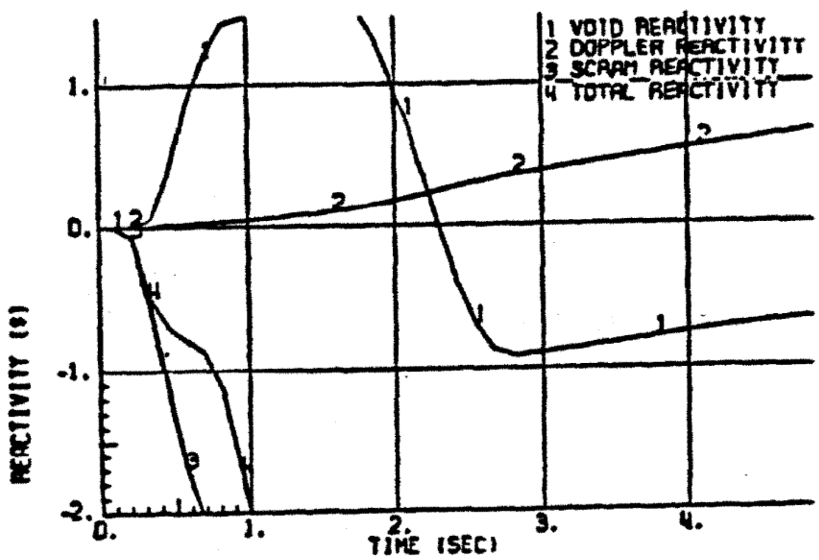




Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	GENERATOR LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER, 109.6% FLOW, 373 F TFW EOC1-2K MWD/T [HISTORICAL INFORMATION] FIGURE 15.B.2-5 SHEET 1 OF 2





GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	GENERATOR LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER, 109.6% FLOW, 373° F TFW EOC1-2K MWD/T [HISTORICAL INFORMATION] FIGURE 15B.2-5 SHEET 2 OF 2

Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

Figure 15B.2-6
(Sheets 1 of 2 thru 2 of 2)

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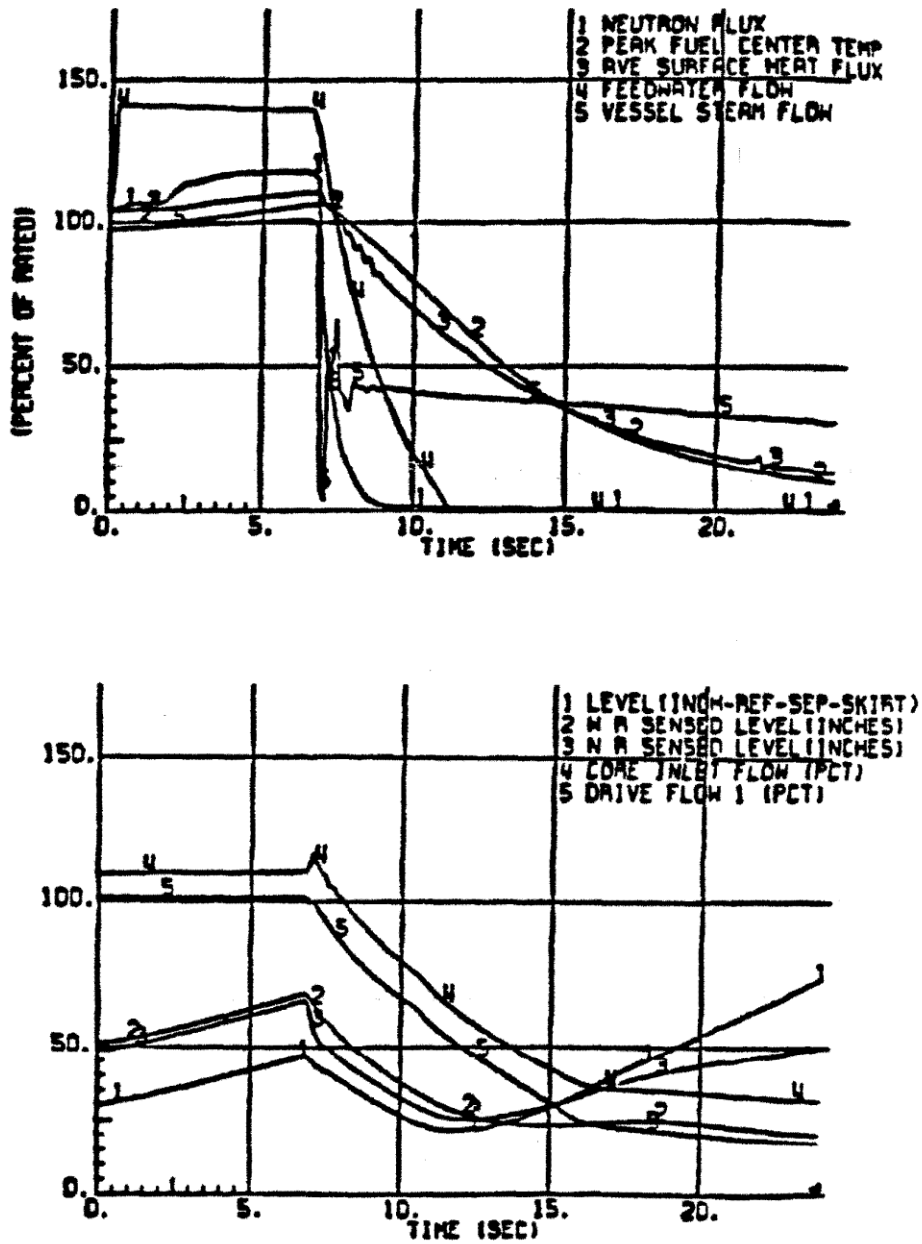
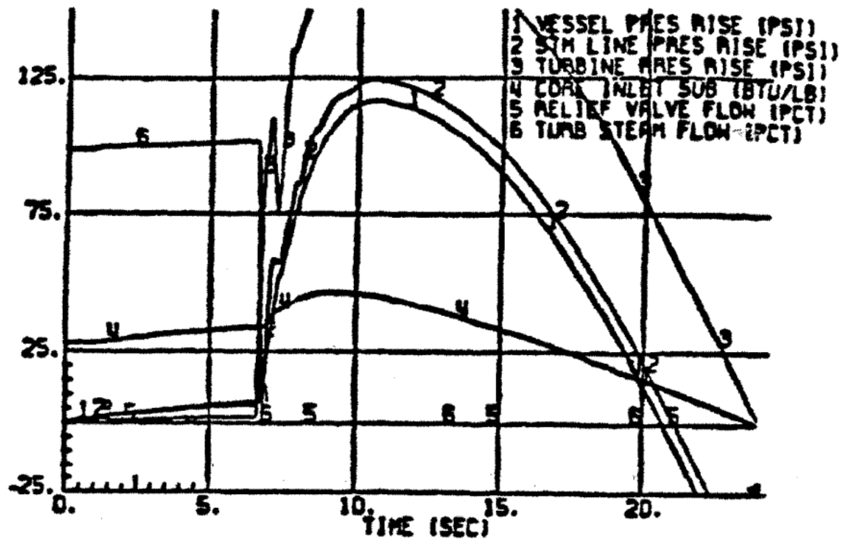
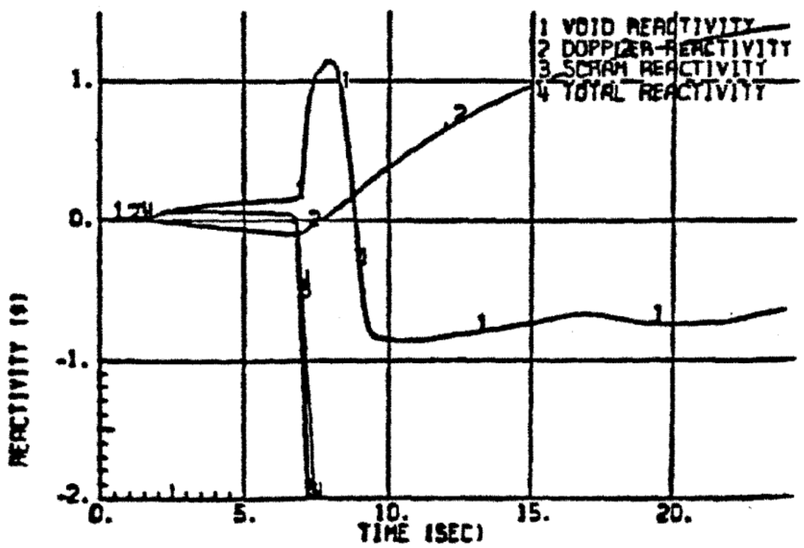


GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE 104.2% POWER, 109.6% FLOW, 373° F TFW EOC1-2K MWD/T [HISTORICAL INFORMATION] FIGURE 15B.2-7 SHEET 1 OF 2

Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.





GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE BYPASS FAILURE 104.2% POWER, 109.6% FLOW, 373° F TFW EOC1-2K MWD/T [HISTORICAL INFORMATION] FIGURE 15B.2-7 SHEET 2 OF 2

Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

Figure 15B.2-8
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15B.3 STABILITY ANALYSIS

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [General Design Criterion 12 (10CFR50, Appendix A) states that power oscillations which result in exceeding specified acceptable fuel design limits are either not possible or can be readily and reliably detected and suppressed. Reference 5 provides stability compliance criteria for GEH fueled BWRs operating in the vicinity of limit cycles. Analyses in Reference 5 demonstrate that for neutron flux limit cycle oscillations just below the 120% neutron flux scram setpoint, fuel design limits are not exceeded for those GEH BWR fuel designs contained in General Electric - Hitachi (GEH) Standard Application for Reactor Fuel (GESTAR, Reference 6).

For demonstration of compliance with GDC 12, the generic analyses of Reference 5 are independent of stability margin since the reactor is already assumed to be in limit cycle oscillations (no stability margin). This implicitly covers any variations in stability margins caused by FWHOS operation. The fuel performance during limit cycle oscillations is characteristically dependent on the fuel design and certain system features (e.g., high neutron flux scram setpoint, inlet orifice diameter of channel) and as such it is possible to determine the acceptability of fuel design independent of plant and cycle parameters. The effects of any changes caused by FWHOS operation (e.g., power distribution inlet subcooling) are covered by the bounding analyses performed in Reference 5. Therefore, the stability compliance criteria of Reference 5 are satisfied for operation in the FWHOS mode.]

15B.4 LOSS OF COOLANT ACCIDENT ANALYSIS

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [A Loss of Coolant Accident Analysis was performed for GGNS with FWHOS operation. Reduction of feedwater temperature results in increased subcooling in the vessel thus increasing the mass flow rate out of a LOCA break. However, an increase in initial total system mass and a delay in lower plenum flashing also occur. They act together to decrease the impact of increased flow out of the recirculation line break. As a result of this offsetting effect, the peak cladding temperature (PCT) was shown to be lower than the 2098°F value reported for GGNS and below the 2200°F 10CFR50.46 cladding temperature limit.]

15B.5 CONTAINMENT RESPONSE ANALYSIS

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [The impact of FWHOS on the containment LOCA response was evaluated. Both Main Steam Line (MSL) break and recirculation line break were analyzed over the entire operation power/flow region. Even though the reduced feedwater temperature increases the subcooling of the coolant, the mass flow rate from the postulated recirculation pipe break also increases, but is limited by the critical flow of the break. The final outcome is that the peak drywell and containment pressures under the FWHOS conditions are bounded by the design values in Chapter 6.]

15B.6 ACOUSTIC LOAD AND FLOW INDUCED LOADS IMPACT ON INTERNALS

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [Acoustic loads are loads on vessel internals created by a sudden LOCA. Acoustic loading is proportional to total pressure wave amplitude in the vessel due to LOCA.

Loads are created on the shroud, shroud support and jet pumps due to high velocity flow in the downcomer in a postulated recirculation line break. These flow induced loads are affected by the critical mass flux rate out of the break. The reactor internals most impacted by acoustic and flow induced loads under FWHOS operation are the shroud, shroud support and jet pump. The impact on these components were evaluated over the operating power flow region. FWHOS increases subcooling thus reduces critical flow out of the break. However, FWHOS also increases density. The analyses concluded that these components have enough design margin to handle the loading during FWHOS operation.]

15B.7 FEEDWATER NOZZLE FATIGUE USAGE

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [An evaluation was performed on the feedwater nozzle in GGNS for FWHOS operation. Assuming a full single 18 month cycle operation with feedwater heater out of service based on an 80% capacity factor would result in 438 full power days operation per cycle. This will result in an additional 0.0214 fatigue usage factor over 40 years of continuous FWHOS operation. An evaluation was also performed assuming end of cycle operation with feedwater temperature between 420°F and 250°F for 41 full power days per year for 40 years. The resultant fatigue usage factor increases by 0.001. The total fatigue usage factor will still be less than 0.8, which is below the limit of 1.0.

The above assumption of 40 years of continuous FWHOS operation is extremely conservative. The nozzle fatigue is expected to be much less than the results presented above.]

15B.8 FEEDWATER SPARGER IMPACT EVALUATION

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [An evaluation was performed to examine the impact of FWHOS operation on the feedwater sparger for GGNS. Cases were analyzed to determine the number of days allowable per year (for 40 years) for FWHOS operation at 370°F without exceeding the feedwater sparger fatigue usage factor limit of 1.0. Results of this study are presented in Table 15B.8-1. This value is sensitive to the temperature reduction step used in FFWTR operation. This reduction in sparger lifetime is mainly of economic concern.]

TABLE 15B.8-1:[HISTORICAL INFORMATION] SUMMARY OF FEEDWATER SPARGER FATIGUE ANALYSIS RESULTS FOR FWHOS OPERATION

FFWTR To 250°F in 41 days for 18-month cycle for 40 years	Allowable Number of Days per Year** For FWHOS operation for 40 Years At FWT Of 370°F
No FFWTR	256
3 Step*	127
7 Step*	144

*3 Step FFWTR is ~3 - 50°F Step

*7 Step FFWTR is ~7 - 25°F Step

** This evaluation assumes 70% capacity factor. Allowable number of days which results in a feedwater sparger fatigue usage factor of 1.0.

15B.9 REACTOR PROTECTION SYSTEM SETPOINT

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [GGNS's turbine stop and control valve closure scram functions have a low power limit at 40% NB rated power with 420°F feedwater. A given core power based on 420°F feedwater will not produce the same steam flow as the same core power based on 370°F rated feedwater. Turbine steam flow characteristics change when feedwater temperature is reduced. Thus, it is necessary to readjust turbine stop and control valve scram bypass setpoints for FWHOS operation.

At reactor power levels where significant amounts of steam are being generated, the fast closure of turbine stop or control valves will result in rapid reactor vessel pressurization. When pressure increases, power increases, especially if the bypass valves fail to open. For this reason, scram occurs on turbine stop valve position and control valve fast closure to provide margin to the core thermal-hydraulic safety limit. At low power levels, high neutron flux scram and vessel pressure scram and other normal scram functions are sufficient to provide the safety limit margin even with stop valve or control valve sudden closures. The required lower bound for stop valve position and control valve fast closure scram is 40% of NB rated power. This is equivalent to ~30% of the first-stage pressure (in psia) that would exist at turbine valves wide-open (VWO) steam flow conditions. Turbine first-stage pressure is the parameter used to enable the turbine valve closure scram functions. Therefore, below ~40% power, the turbine stop valve or control valve scram functions are not enabled.

As feedwater temperature is reduced, steam flow decreases. Since the FFWTR process maintains rated core thermal power, the steam flow reduction means that the turbine first-stage pressure versus power relationship is altered. A new setpoint is established for the trip units prior to commencement of each FWHOS operation at each operating cycle. The recommended setpoints for the turbine stop or control valve RPS scram function are 21.0 ± 0.5% of calibrated span for 420°F to 370°F rated feedwater temperature operation.]

15B.10 MISCELLANEOUS IMPACT EVALUATION

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

15B.10.1 Feedwater System Piping

[HISTORICAL INFORMATION] [A standard stress analysis was performed on the feedwater system piping up to the first feedwater guide lug outside the containment for feedwater temperature at 250°F. Results of the study show that with the additional FWHOS operations, the feedwater piping fatigue usage factor still meets the allowable limit of 1.0.]

15B.10.2 Impact on Anticipated Transient Without Scram (ATWS)

[HISTORICAL INFORMATION] [An impact evaluation was performed which shows that reducing feedwater temperature helps to reduce the consequences of an ATWS event. The worst ATWS event, MSIVC, was used to evaluate the FWHOS impact. As a result of reduced feedwater temperature, steam flow and core average void fraction are reduced. The lower steam flow rate is produced because more of the core heat is needed to heat up the colder moderator in the core. Therefore, less steam is generated at its rated power as feedwater temperature is decreased, a case less severe than when the plant is operating with feedwater temperature at 420°F.]

15B.10.3 Annulus Pressurization Load (APL) Impact

[HISTORICAL INFORMATION] [A boundary analysis was performed to determine the impact of FWHOS operation on annulus pressurization loads (APL). It is found that FWHOS has a small impact on annulus pressurization loads. The FWHOS APL is bounded by the normal operation APL limits.]

15B.10.4 Fuel Mechanical Performance

[HISTORICAL INFORMATION] [Evaluations were performed to determine the acceptability of GGNS FWHOS operation on GE-6 fuel rod and assembly thermal/mechanical performance. Component pressure differential and fuel rod overpower values were determined for anticipated operational occurrences initiated from FWHOS conditions. These values were found to be bounded by those applied in the fuel rod and assembly design bases and therefore, GGNS FWHOS operation is acceptable and consistent with fuel design basis.]

15B.11 REFERENCES

1. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154 Oct. 1978.

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

3. "Three Dimensional BWR Core Simulator," NEDO-20953-A, January 1977.

4. Letter, J. S. Charnley (GE) to F. J. Miraglia (NRC), "Loss of Feedwater Heating Analysis," July 5, 1983 (MFN-125-83).

5. "Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria," NEDE-22277-P, December 1982.

6. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, January 1982.

7. General Electric Company Report "GGNS Feedwater Heater(s) Out of Service Analysis," March 1986.

8. ECH-NE-18-00022, Revision 1, "Supplemental Reload Licensing Report for Grand Gulf Nuclear Station Reload 21 Cycle 22," July 2018

APPENDIX 15C RECIRCULATION SYSTEMS SINGLE-LOOP OPERATION

15C.1 INTRODUCTION AND SUMMARY

Single-loop operation (SLO) at reduced power is highly desirable in the event recirculation pump or other component maintenance renders one loop inoperative. SLO is considered an operational flexibility that is allowed at GGNS in accordance with the Technical Requirements Manual LCO TR3.4.1 and plant operating procedures. SLO is restricted to a reactor power of 2705 MWt and a flow of 60.9 Mlb/hr. The initial cycle analysis performed by the NSSS vendor for SLO operation was bounded by the maximum extended operating domain (MEOD) and was unchanged by EPU. For EPU the absolute power limit for SLO remains the same, requiring a proportional reduction in the percent of rated power at the uprated power level. SLO operation is not allowed in the MELLLA+ operating region. Refer to Appendix 15D for a discussion of the power/flow operating domains.

To justify SLO, accidents and abnormal operational transients associated with power operations, as presented in Sections 6.2 and 6.3 and the main text of Chapter 15.0, are reviewed for the single-loop case with only one pump in operation. This appendix presents the results of this safety evaluation for the initial cycle operation of the Grand Gulf Nuclear Station (GGNS) with single recirculation loop inoperable. For core reload cycles, the effects on plant operating limits are discussed in the appropriate sections of the main body of the UFSAR. The results of the reload evaluations are contained in the COLR and the Supplemental Reload Licensing Report.

[HISTORICAL INFORMATION] [The evaluation described in this appendix is performed for GE-6 fueled GGNS on an initial cycle basis and is applicable to GE-6 fueled normal annual 12 month initial cycle operation. The conditions are those of continued operation in the operating domain defined in Figure 15D.3-2 of Appendix 15D up to a maximum power of 70.6% of the initially licensed value.

Increased uncertainties in the core total flow and Traversing In-Core Probe (TIP) readings resulted in a 0.01 incremental increase in the Minimum Critical Power Ratio (MCPR) fuel cladding integrity safety limit during single-loop operation. No increase in rated MCPR operating limit and no change in the power dependent and flow dependent MCPR limit ($MCPR_f$ and $MCPR_p$) are required because all abnormal operational transients analyzed for single-loop operation indicated there is more than enough MCPR margin to compensate for this increase in MCPR safety limit. The recirculation flow rate dependent rod block and scram setpoint equation given in the TRM are adjusted for one-pump operation.

Thermal-hydraulic stability was evaluated for its adequacy with respect to General Design Criteria 12 (10CFR50, Appendix A). It is shown that SLO satisfies this stability criterion. It is further shown that the increase in neutron noise observed during SLO is independent of system stability margin.

To prevent potential control oscillations from occurring in the recirculation flow control system, the flow control should be in master manual for single-loop operation.

The limiting Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) reduction factor for single-loop operation is calculated to be 0.86.

The containment response for a Design Basis Accident (DBA) recirculation line break with single-loop operation is bounded by the rated power two-loop operation analysis presented in Section 6.2. This conclusion covers all single-loop operation power/flow conditions.

The impact of single loop operation on the Anticipated Transient Without Scram (ATWS) analysis was evaluated. It is found that all ATWS acceptance criteria are met during SLO.

The fuel thermal and mechanical duty for transient events occurring during SLO is found to be bounded by the fuel design bases. The Average Power Range Monitor (APRM) fluctuation should not exceed a flux amplitude of ±15% of rated and the core plate differential pressure fluctuation should not exceed 3.2 psi peak to peak to be consistent with the fuel rod and assembly design bases.

A recirculation pump drive flow limit will be imposed for SLO. The highest drive flow tested during the startup test program at GGNS that meets acceptable vessel internal vibration criteria will be the drive flow limit for SLO.]

15C.2 MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [Except for core total flow and TIP reading, the uncertainties used in the statistical analysis to determine the MCPR fuel cladding integrity safety limit are not dependent on whether coolant flow is provided by one or two recirculation pumps. Uncertainties used in the two-loop operation analysis are documented in the FSAR. A 6% core flow measurement uncertainty has been established for single-loop operation (compared to 2.5% for two-loop operation). As shown below, this value conservatively reflects the one standard deviation (one sigma) accuracy of the core flow measurement system documented in Reference 1. The random noise component of the TIP reading uncertainty was revised for single recirculation loop operation to reflect the operating plant test results given in Subsection 15C.2.2. This revision resulted in a single-loop operation process computer effective TIP uncertainty of 6.8% of initial cores and 9.1% for reload cores. Comparable two-loop process computer uncertainty values are 6.3% for initial cores and 8.7% for reload cores. The net effect of these two revised uncertainties is a 0.01 incremental increase in the required MCPR fuel cladding integrity safety limit.]

15C.2.1 Core Flow Uncertainty

15C.2.1.1 Core Flow Measurement During Single-Loop Operation

[HISTORICAL INFORMATION] [The jet pump core flow measurement system is calibrated to measure core flow when both sets of jet pumps are in forward flow; total core flow is the sum of the indicated loop flows. For single-loop operation, however, some inactive jet pumps will be backflowing (at active pump flow above approximately 36%). Therefore, the measured flow in the backflowing jet pumps must be subtracted from the measured flow in the active loop to obtain the total core flow. In addition, the jet pump coefficient is different for reverse flow than for forward flow, and the measurement of reverse flow must be modified to account for this difference.

In single-loop operation, the total core flow is derived by the following formula:

$$\text{Total Core Flow} = \text{Active Loop Indicated Flow} - C \ \text{Inactive Loop Flow}$$

Where C (= 0.95) is defined as the ratio of "Inactive Loop True Flow" to "Inactive Loop Indicated Flow". "Loop Indicated Flow" is the flow measured by the jet pump "single-tap" loop flow summers and indicators, which are set to read forward flow correctly.

The 0.95 factor was the result of a conservative analysis to appropriately modify the single-tap flow coefficient for reverse flow.* If a more exact, less conservative core flow is required, special in-reactor calibration tests would have to be made. Such calibration tests would involve: calibrating core support plate P versus core flow during one-pump and two-pump operation along with 100% flow control line and calculating the correct value of C based on the core support plate P and the loop flow indicator readings.]

15C.2.1.2 Core Flow Uncertainty Analysis

[HISTORICAL INFORMATION] [The uncertainty analysis procedure used to establish the core flow uncertainty for one-pump operation is essentially the same as for two-pump operation, with some exceptions. The core flow uncertainty analysis is described in Reference 1. The analysis of one-pump core flow uncertainty is summarized below.

For single-loop operation, the total core flow can be expressed as follows (refer to Figure 15C.2-1):

$$W_C = W_A - W_I$$

where:

W_C = total core flow,

W_A = active loop flow, and

W_I = inactive loop (true) flow.

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*The analytical expected value of the "C" coefficient for GGNS is ~0.82.

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

$$W_c^2 = W_{sys}^2 + \left(\frac{1}{1-a}\right)^2 W_{A_{rand}}^2 + \left(\frac{a}{1-a}\right)^2 W_{I_{rand}}^2 + C^2$$

where:

W_C	=	uncertainty of total core flow;
W_{sys}	=	uncertainty systematic to both loops;
$W_{A_{rand}}$	=	random uncertainty of active loop only;
$W_{I_{rand}}$	=	random uncertainty of inactive loop only;
C	=	uncertainty of "C" coefficient; and
a	=	ratio of inactive loop flow (W_I) to active loop flow (W_A).

From an uncertainty analysis, the conservative, bounding values of W_{sys}' $W_A{}'_{rand}$ $W_I{}'_{rand}$ and C are 1.6%, 2.6%,

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

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

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

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

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

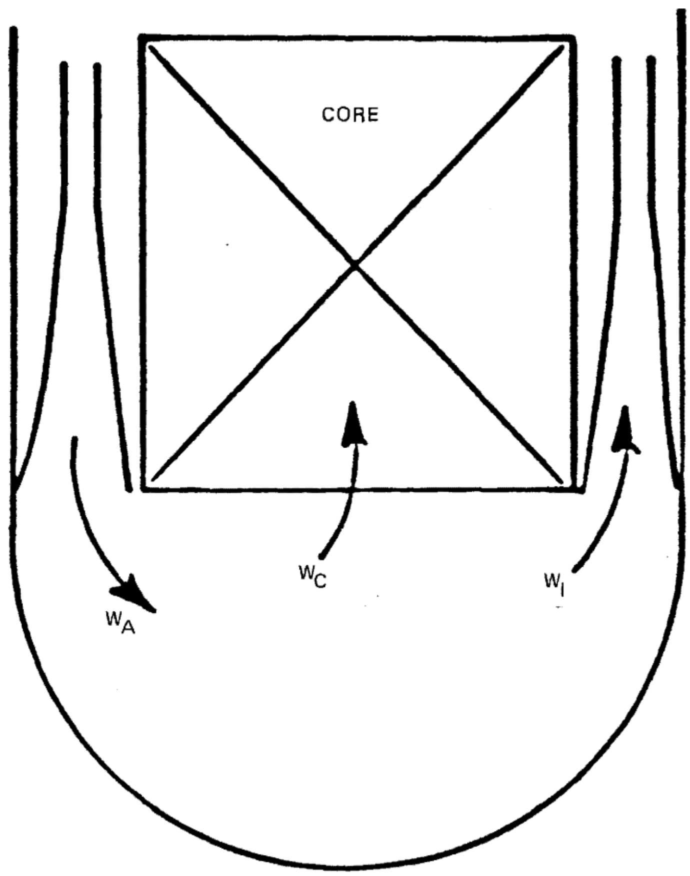
In summary, core flow during one-pump operation is measured in a conservative way and its uncertainty has been conservatively evaluated.]

15C.2.2 Tip Reading Uncertainty

[HISTORICAL INFORMATION] [To ascertain the TIP noise uncertainty for single recirculation loop operation, a test was performed at an operating BWR. The test was performed at a power level 59.3% of rated with a single recirculation pump in operation (core flow 46.3% of rated). A rotationally symmetric control rod pattern existed during the test.

Five consecutive traverses were made with each of five TIP machines, giving a total of 25 traverses. Analysis of this data resulted in a nodal TIP noise of 2.85%. Use of this TIP noise value as a component of the process computer total uncertainty results in a one-sigma process computer total effective TIP uncertainty value for single-loop operation of 6.8% for initial cores and 9.1% for reload cores. The results of the analysis are directly applicable to GGNS because the data collected are typical random neutron, electronic and boiling noise during SLO for a BWR.]

*This flow split ratio varies from about 0.13 to 0.36. The 0.36 value is a conservative bounding value. The analytical expected value of the flow split ratio for GGNS is ~0.28.



W_C = Total Core Flow

W_A = Active Loop Flow

W_I = Inactive Loop Flow

SYSTEM ENERGY RESOURCES, INC. GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	ILLUSTRATION OF SINGLE RECIRCULATION LOOP OPERATION FLOWS [HISTORICAL INFORMATION] FIGURE 15C.2-1

15C.3 MCPR OPERATING LIMIT

15C.3.1 Abnormal Operating Transients

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [Operating with one recirculation loop results in a maximum power output which is about 30% below that which is attainable for two-pump operation. Therefore, the consequences of abnormal operation transients from one-loop operation will be considerably less severe than those analyzed from a two-loop operational mode. For pressurization, flow increase, flow decrease, and cold water injection transients, results presented in the FSAR bound both the thermal and overpressure consequences of one-loop operation.

Figure 15C.3-1 shows the consequences of a typical pressurization transient (turbine trip) as a function of power level. As can be seen, the consequences of one-loop operation are considerably less because of the associated reduction in operating power level.

The consequences of flow decrease transients are also bounded by the full power analysis. A single pump trip from one-loop operation is less severe than a two-pump trip from full power because of the reduced initial power level.

The worst flow increase transient results from recirculation flow controller failure, and the worst cold water injection transient results from the loss of feedwater heater. For the former, the $MCPR_f$ curve is derived from a postulated event involving runout of both recirculation loops. This condition produces the maximum possible power increase and hence maximum ΔMCPR for transients initiated from less than rated power and flow. When operating with only one recirculation loop, the flow and power increase associated with this failure with only one loop will be less than that associated with both loops; therefore, the $MCPR_f$ curve derived with the two-pump assumption is conservative for single-loop operation. The latter event, loss of feedwater heating, is generally the most severe cold water increase event with respect to increase in core power. This event is caused by positive reactivity insertion from core inlet subcooling and it is relatively insensitive to initial power level. A generic statistical loss of feedwater heater analysis using different initial power levels and other core design parameters concluded one-pump operation with lower initial power level is conservatively bounded by the full power two-pump analysis. Inadvertent restart of the idle recirculation pump has been analyzed in the FSAR and is still applicable for single-loop operation.

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

In the following sections, three of the most limiting transients of core flow increase, pressurization, and flow decrease events are analyzed for single-loop operation. They are, respectively:

a. feedwater flow controller failure (maximum demand), (FWCF)

b. generator load rejection with bypass failure, (LRNBP), and

c. one pump seizure accident. (PS)

The plant initial conditions are given in Table 15C.3-1.]

15C.3.1.1 Feedwater Controller Failure - Maximum Demand

15C.3.1.1.1 Core and System Performance

[HISTORICAL INFORMATION] [Mathematical Model

The computer model described in Reference 2 was used to simulate this event.

Input Parameters and Initial Conditions

The analysis has been performed with the plant conditions tabulated in Table 15C.3-1, except the initial vessel water level is at level setpoint L4 for conservatism. By lowering the initial water level, more cold feedwater will be injected before Level 8 s reached resulting in higher heat fluxes.

End of cycle (all rods out) scram characteristics are assumed. The safety/relief valve action is conservatively assumed to occur with higher than nominal setpoints. The transient is simulated by programming an upper limit failure in the feedwater system such that 130% of rated feedwater flow occurs at the design pressure of 1065 psig.

Results

The simulated feedwater controller transient is shown in Figure 15C.3-2 for the case of 70.6% of the initially licensed power 54.1% core flow. The high-water level turbine trip and feedwater pump trip are initiated at approximately 4.2 seconds. Scram occurs simultaneously from Level 8, and limits the peak neutron flux. MCPR is considerably above the safety limit so no fuel failure due to boiling transition is predicted. The turbine bypass system opens to limit peak pressure in the steamline near the safety valves to 1045 psig and the pressure at the bottom of the vessel to about 1059 psig.

Consideration of Uncertainties

All systems used for protection in this event were assumed to have the poorest allowable response (e.g., relief setpoints, scram stroke time, etc.). Expected plant behavior is, therefore, expected to lead to a less severe transient.]

15C.3.1.1.2 Barrier Performance

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

15C.3.1.1.3 Radiological Consequences

[HISTORICAL INFORMATION] [The consequences of this event do not result in any calculated fuel failures; however, radioactive steam is discharged to the suppression pool as a result of SRV activation.]

15C.3.1.2 Generator Load Rejection With Bypass Failure

15C.3.1.2.1 Core and System Performance

[HISTORICAL INFORMATION] [Mathematical Model

The computer model described in Reference 2 was used to simulate this event.

Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15C.3-1.

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

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

Auxiliary power is independent of any turbine generator overspeed effects and is continuously supplied at rated frequency, assuming automatic fast transfer to auxiliary power supplies.

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

Results

The simulated generator load rejection without bypass is shown in Figure 15C.3-3.

Table 15C.3-2 shows for the case of bypass failure, peak neutron flux reaches about 70.7% of the initially licensed rated and peak steamline pressure at the valves reaches 1167 psig. The peak nuclear system pressure reaches 1179 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1375 psig. The calculated MCPR is 1.41, which is well above the safety limit.

Consideration of Uncertainties

The full-stroke closure rate of the turbine control valve of 0.15 second is conservative. Typically, the actual closure rate is approximately 0.2 second.The less time it takes to close, the more severe the pressurization effect.

All systems used for protection in this event were assumed to have the poorest allowable response (e.g., relief setpoints, scram stroke time, etc.). Expected plant behavior is, therefore, expected to reduce the actual severity of the transient.]

15C.3.1.2.2 Barrier Performance

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

15C.3.1.2.3 Radiological Consequences

[HISTORICAL INFORMATION] [The consequences of this event do not result in any calculated fuel failures; however, radioactivity is nevertheless discharged to the suppression pool as a result of SRV activation.]

15C.3.1.3 Recirculation Pump Seizure Accident

15C.3.1.3.1 Core and System Performance

[HISTORICAL INFORMATION] [Mathematical Model

The computer model described in Reference 3 was used to simulate this event.

Input Parameters and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in Table 15C.3-1. For the purpose of evaluating consequences to the fuel thermal limits, this transient event is assumed to occur as a consequence of an unspecified, instantaneous stoppage of the active recirculation pump shaft while the reactor is operating at ~71% of the initially licensed NB rated power under single-loop operation. Also, the reactor is assumed to be operating at thermally limiting conditions.

The void coefficient is adjusted to the most conservative value; that is, the least negative value in Table 15C.3-1.

Results

Figure 15C.3-4 presents the results of the accident. Core coolant flow drops rapidly, reaching a minimum value of 26% rated at about 1.3 seconds. The minimum CPR value during the transient is 1.24 and poses no threats to thermal limits.]

15C.3.1.3.2 Barrier Performance

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

15C.3.1.3.3 Radiological Consequences

[HISTORICAL INFORMATION] [The consequences of this event do not result in any calculated fuel failures.]

15C.3.1.4 Summary and Conclusions

[HISTORICAL INFORMATION] [The transient peak value results are summarized in Table 15C.3-3. The Critical Power Ratio (CPR) results are summarized in Table 15C.3-3. This table indicates that for the transient events analyzed here, the MCPRs for all transients are above the single-loop operation safety limit value of 1.07. It is concluded the thermal margin safety limits established for two-pump operation are also applicable to single-loop operation conditions.

For pressurization, Table 15C.3-2 indicates the peak pressures are below the ASME code value of 1375 psig. Hence, it is concluded the pressure barrier integrity is maintained under single-loop operation conditions.]

15C.3.2 Rod Withdrawal Error

[HISTORICAL INFORMATION] [The rod withdrawal error (RWE) transient for two-loop operation documented in the main text of this chapter employs a statistical evaluation of the minimum critical power ratio (MCPR) and linear heat generation rate (LHGR) response to the withdrawal of ganged control rods for both rated and off-rated conditions. The required MCPR limit protection for the event is provided by the rod withdrawal limits (RWL) system. Since this analysis covered all off-rated conditions in the power/flow operating map, single-loop operation is bounded by the current technical specification.

The Average Power Range Monitor (APRM) rod block system provides additional alarms and rod blocks when power levels are grossly exceeded. Modification of the APRM rod block equation (below) is required to maintain the two loop rod block versus power relationship when in one loop operation. The Option III stability trip function has been installed. Refer to Sections 4.4.4.6, 7.1.2.1, and 7.2.1 for additional information regarding the Option III stability trip function.

One-pump operation results in backflow through 12 of the 24 jet pumps while the flow is being supplied into the lower plenum from the 12 active jet pumps. Because of the backflow through the inactive jet pumps, the present rod block equation was conservatively modified for use during one-pump operation because the direct active-loop flow measurement may not indicate actual flow above about 36% core flow without correction.

A procedure has been established for correcting the APRM rod block equation to account for the discrepancy between actual flow and indicated flow in the active loop. This preserves the original relationship between APRM rod block and actual effective drive flow when operating with a single loop.

The two-pump rod block equation is:

$$RB = mW + RB_{100} - m(100)$$

The one-pump equation becomes:

$$RB = mW + RB_{100} - m(100) - m\Delta w$$

where

ΔW = difference between two-loop and single-loop effective drive flow at the same core flow.

RB = power at rod block in %;

m = flow reference slope

W = drive flow in % of rated.

RB_{100} = top level rod block at 100% flow.

If the rod block setpoint (RB_{100}) is changed, the equation must be recalculated using the new value.

The APRM scram trip settings are flow biased in the same manner as the APRM rod block setting. Therefore, the APRM scram trip settings are subject to the same procedural changes as the rod block settings discussed above.]

15C.3.3 Operating MCPR Limit

[HISTORICAL INFORMATION] [For single-loop operation, the operating MCPR limit remains unchanged from the normal two-loop operation limit. Although the increased uncertainties in core total flow and TIP readings resulted in a 0.01 incremental increase in MCPR fuel cladding integrity safety limit during single-loop operation (Section 15C.2), the limiting transients have been analyzed to indicate that there is more than enough MCPR margin during single-loop operation to compensate for this increase in safety limit. For single loop operation at off-rated conditions, the steady-state operating MCPR limit is established by the $MCPR_p$ and $MCPR_f$ curves. This ensures the 99.9% statistical limit requirement is always satisfied for any postulated abnormal operational occurrence. The abnormal operating transients analyzed concluded that current power dependent $MCPR_p$ limits are bounding for single loop operation. Since the maximum core flow runout during single loop operation is only about 54% of rated, the current flow dependent $MCPR_f$ limits which are generated based on the flow runout up to rated core flow are also adequate to protect the flow runout events during single loop operation.]

TABLE 15C.3-1:INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS AND ACCIDENTS FOR SINGLE-LOOP OPERATION (INITIAL CYCLE[HISTORICAL INFORMATION])*

1.	Thermal Power Level Analysis Value, MWt	2708 (70.6% rated)*
2.	Steam Flow, lb/hr	11.06x10
3.	Core Flow, lb/hr	60.9x10 (54.1% Rated)
4.	Feedwater Flow Rate, lb/sec	3072
5.	Feedwater Temperature, °F	386
6.	Vessel Dome Pressure, psig	981
7.	Vessel Core Pressure, psig	985
8.	Turbine Bypass Capacity, % NBR	35
9.	Core Coolant Inlet Enthalpy Btu/lb	509.5
10.	Turbine Inlet Pressure, psig	946
11.	Fuel Lattice	8x8R
12.	Core Leakage Flow, %	10.65
13.	Required MCPR Operating Limit	1.41(a)
14.	MCPR Safety Limit for Incidents of Moderate Frequency	
	First Core	1.07
	Reload Core	1.08
15.	Doppler Coefficient (-)¢/°F Analysis Data	0.132(b)
16.	Void Coefficient (-)¢/% Rated Voids	
	Analysis Data for Power	
	Decrease Events	4.0(b)
	Analysis Data for Power	
	Increase Events	14.0(b)
17.	Core Average Void Fraction, %	41.9(b)
18.	Jet Pump Ratio, M	3.521
19.	Safety/Relief Valve Capacity, % NBR	
	@1145 psig	102.4
	Manufacturer	DIKKER
	Quantity Installed	20

TABLE 15C.3-1:INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS AND ACCIDENTS FOR SINGLE-LOOP OPERATION (INITIAL CYCLE[HISTORICAL INFORMATION])* (CONTINUED)

20. Relief Function Delay, Seconds	0.4
21. Relief Function Response, Seconds	0.1
22. Setpoints for Safety/Relief Valves	
Safety Function, psig	1175,1185,1195 1205,1215
Relief Function, psig	1145,1155,1165,1175
23. Number of Valve Groupings Simulated	
Safety Function, No.	5
Relief Function, No.	4
24. High Flux Trip, % NBR	
Analysis Setpoint (1.22 x 1.042),	% NBR127.2
25. High Pressure Scram Setpoint, psig	1126
26. Vessel Level Trips, Feet Above	
Separator Skirt Bottom	
Level 8 - (L8), Feet	5.88
Level 4 - (L4), Feet	4.03
Level 3 - (L3), Feet	2.16
Level 2 - (L2), Feet	-2.182
27. APRM Thermal trip	
Setpoint, % NBR @ 100% Core Flow	118.8
28. RPT Delay, seconds	0.19
29. RPT Inertia Time Constant for Analysis, secs.	5
30. Total steamline volume, ft^3	4358

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[a] Operation operating limit is given by $MCPR_f$ for a core flow of 54.1%.

[b] Parameters used in Reference 3 analysis only. Reference 2 values are calculated within the code for end of Cycle 1 condition. These are rated condition values.

Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.

TABLE 15C.3-2:SUMMARY OF TRANSIENT PEAK VALUE RESULTS SINGLE-LOOP OPERATION (INITIAL CYCLE [HISTORICAL INFORMATION]**)**

PARAGRAPH	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX (% NBR)	MAXIMUM DOME PRESSURE (psig)	MAXIMUM VESSEL PRESSURE (psig)	MAXIMUM STEAMLINE PRESSURE (psig)	FREQUENCY* Category
		Initial Condition	70.6	981	998	974	N/A
15C.3.1.1	15C.3-2	Feedwater flow Controller Failure (Maximum Demand)	79.2	1045	1059	1045	a
15C.3.1.2	15C.3-3	Generator Load Rejection With Bypass Failure	70.7	1166	1179	1167	b
15C.3.1.3	15C.3-4	Seizure of Active Recirculation Pump	70.6	984	998	976	c

*a = Moderate frequency incident; b = infrequent; c = limiting faults

Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.

TABLE 15C.3-3: SUMMARY OF CRITICAL POWER RATIO RESULTS - SINGLE-LOOP OPERATION (INITIAL CYCLE [HISTORICAL INFORMATION]**)**

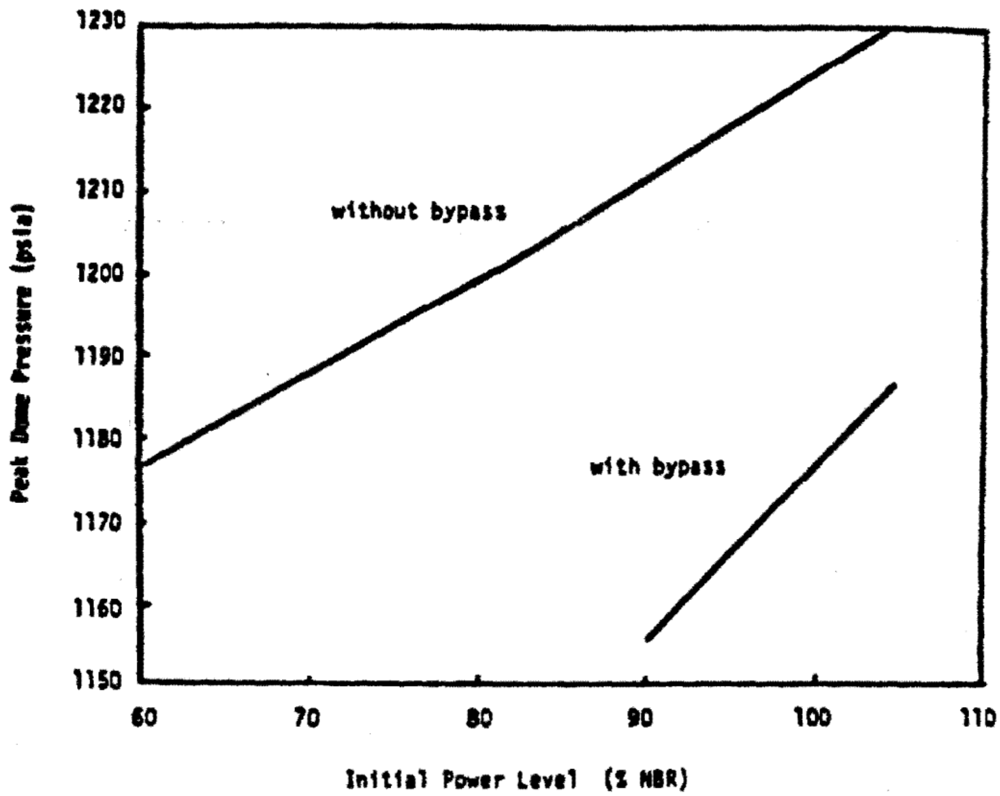
	FWCF	**LRNBT**	**PS**
Initial Operating Condition (% power/% flow)	70.6/ 54.1	70.6/ 54.1	70.6/ 54.1
Required Two Loop Initial MCPR Operating Limit at SLO Condition	1.41	1.41 [a]	1.41
ΔCPR	0.07 [a]	0.00 [b]	0.17
Transient MCPR at SLO	1.34	1.41	1.24
SLMCPR at SLO	1.07	1.07	1.07
Margin Above SLMCPR	0.27	0.34	0.17
Frequency Category	Moderate frequent incident	Infrequent incident	Limiting fault

---

(a) value includes option A adder

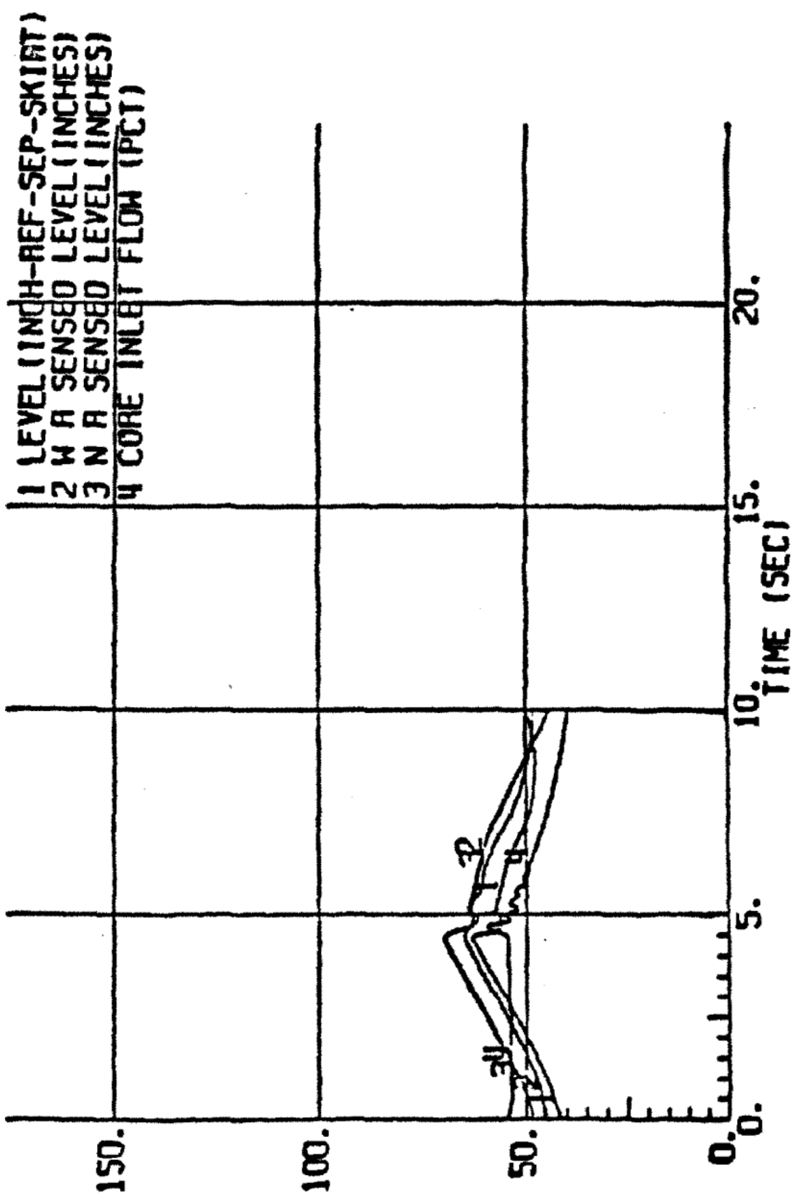
(b) ΔCPR is less than 0.002.

Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.



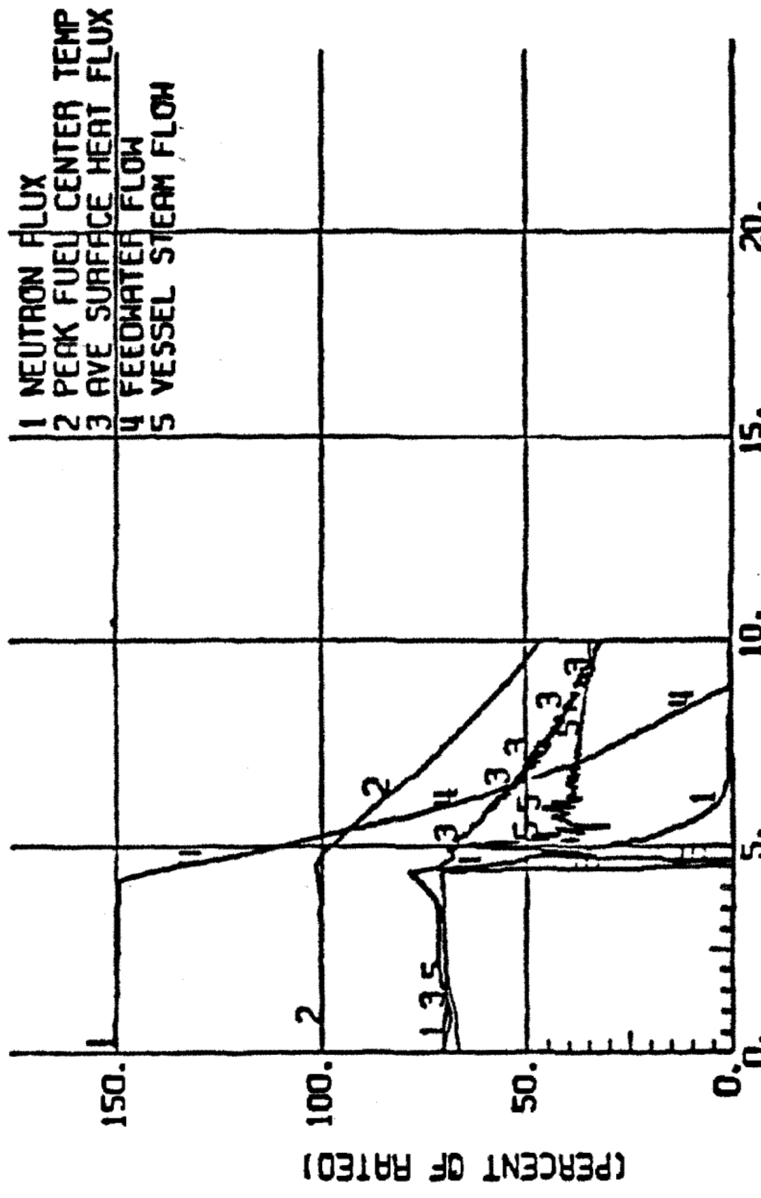
Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	PEAK DOME PRESSURE VERSUS INITIAL POWER LEVEL, TURBINE TRIP AT EOEC (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15C.3-1



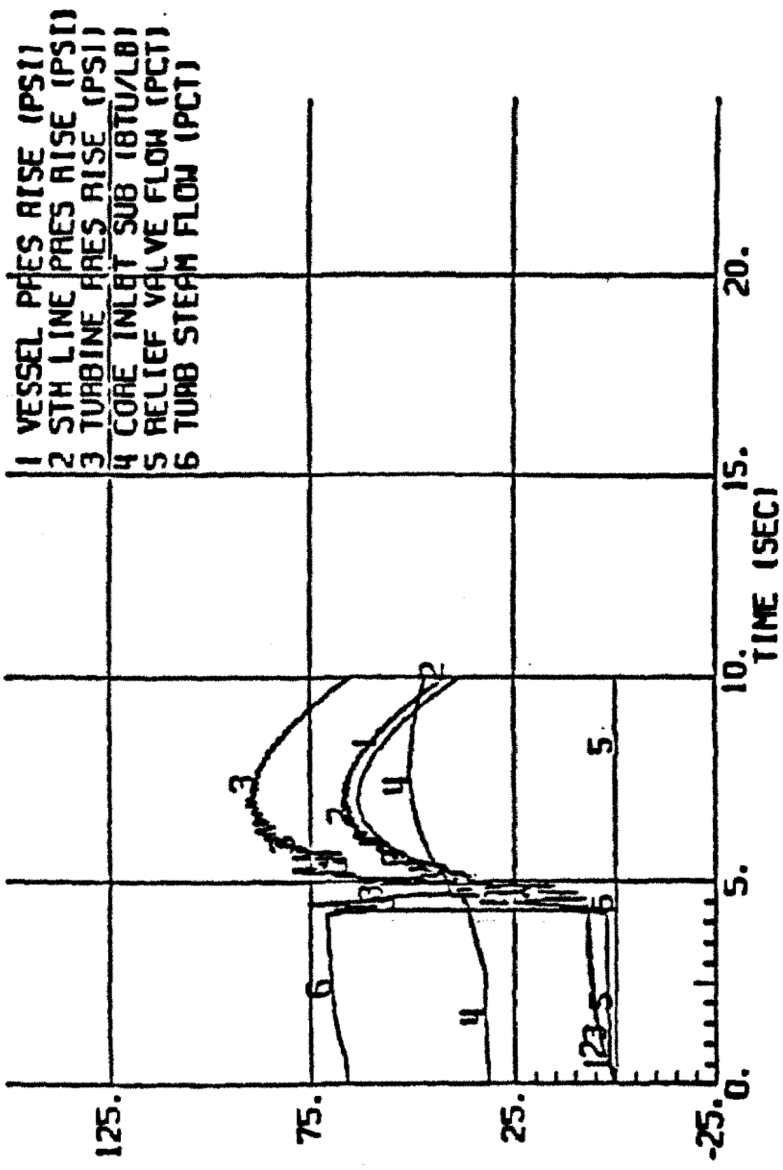
Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE MAXIMUM DEMAND SINGLE LOOP OPERATION 71% POWER, 54% FLOW, (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15C.3-2 SHEET 1 OF 4



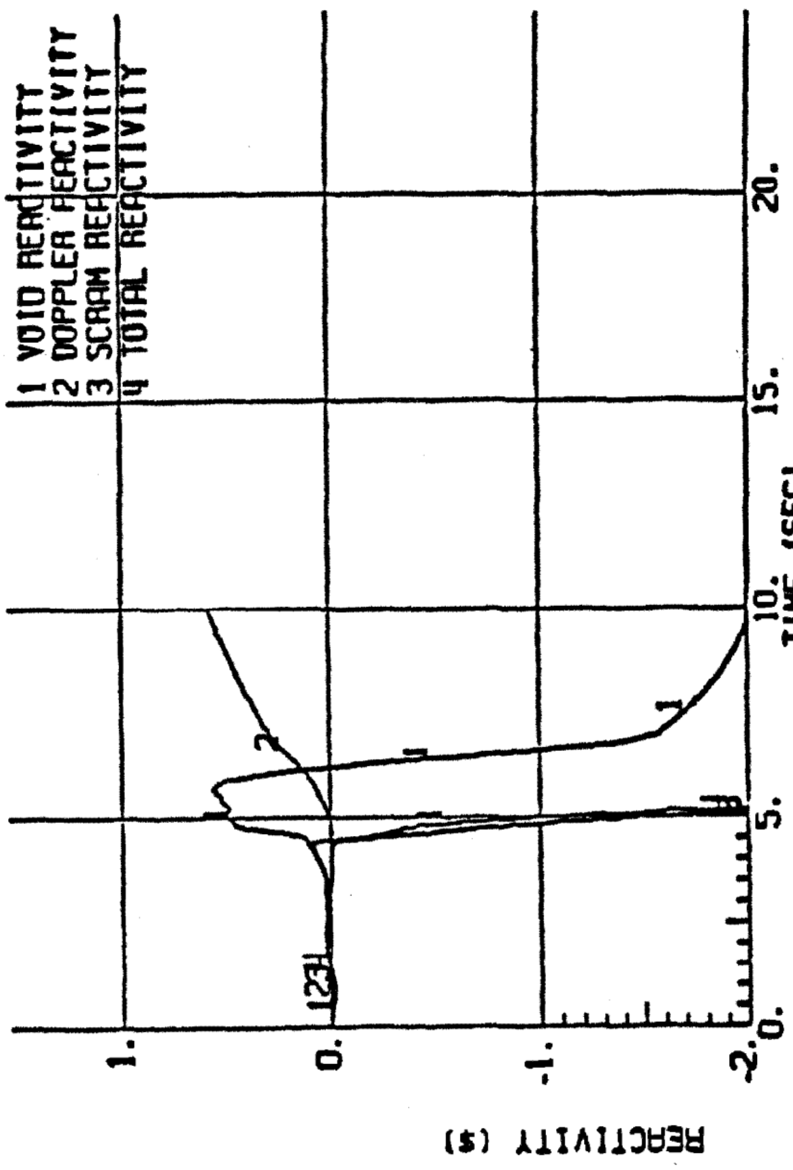
Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE MAXIMUM DEMAND SINGLE LOOP OPERATION 71% POWER, 54% FLOW, (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15C.3-2 SHEET 2 OF 4



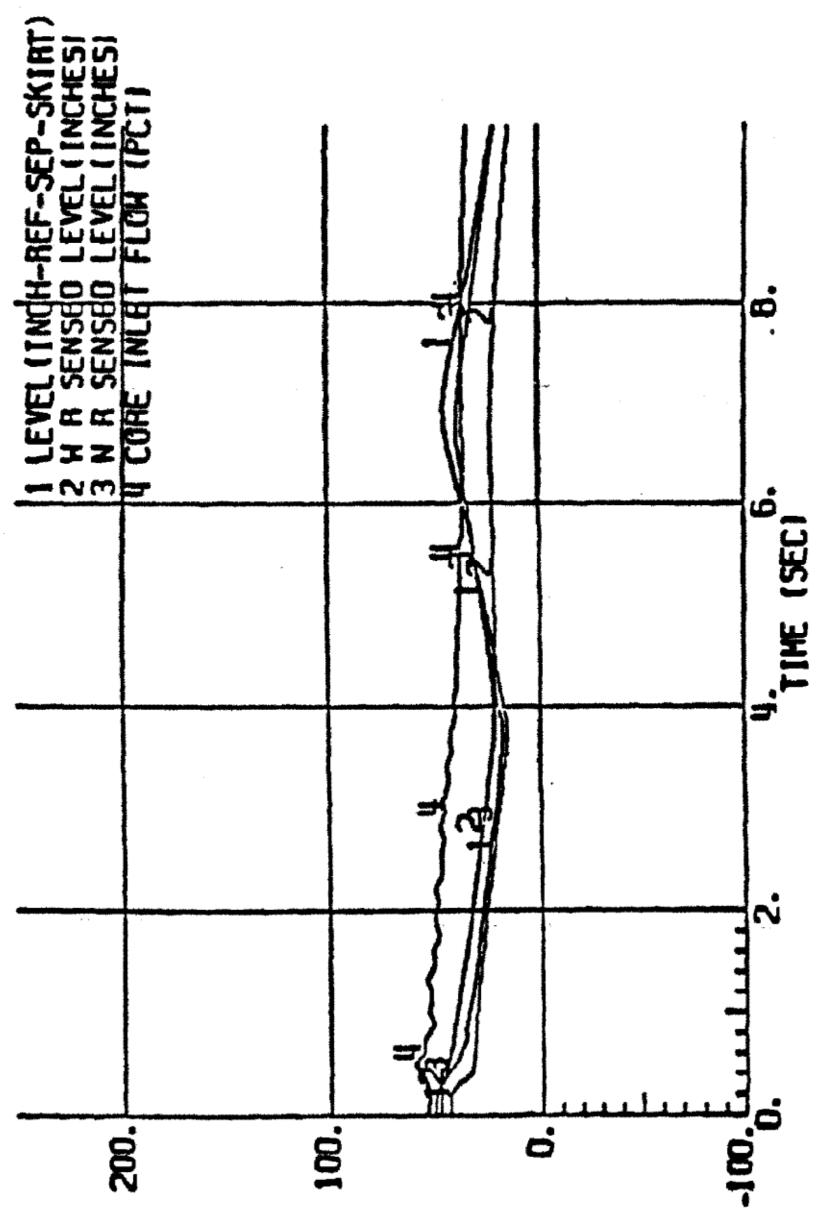
Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE MAXIMUM DEMAND SINGLE LOOP OPERATION 71% POWER, 54% FLOW, (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15C.3-2 SHEET 3 OF 4



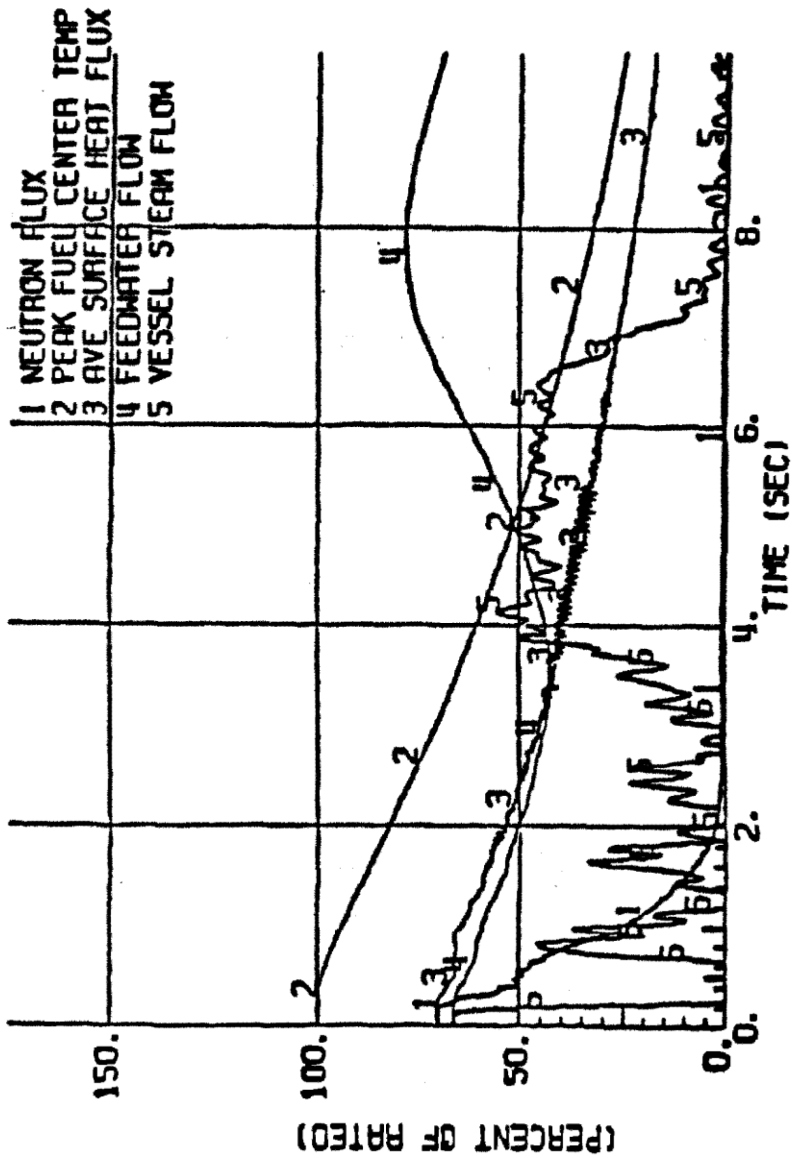
Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE MAXIMUM DEMAND SINGLE LOOP OPERATION 71% POWER, 54% FLOW, (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15C.3-2 SHEET 4 OF 4



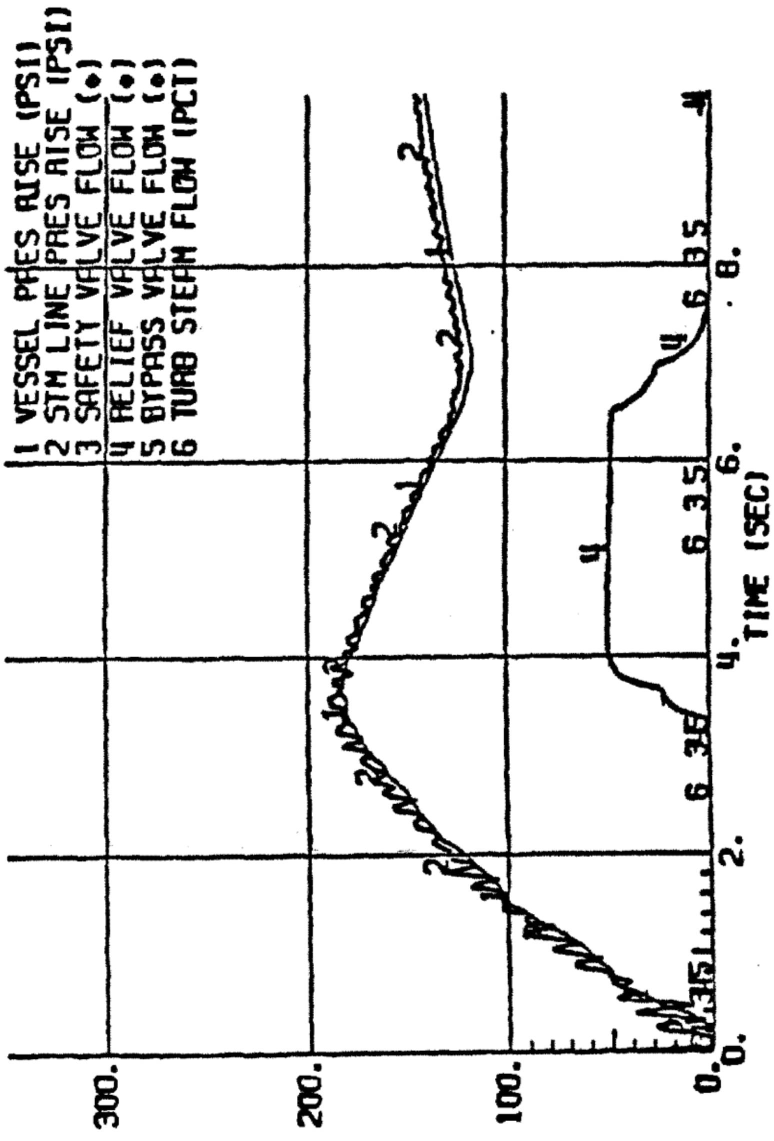
Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	GENERATOR LOAD REJECTION BYPASS FAILURE SINGLE LOOP OPERATION 71% POWER, 54% FLOW, (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15C.3-3 SHEET 1 OF 4



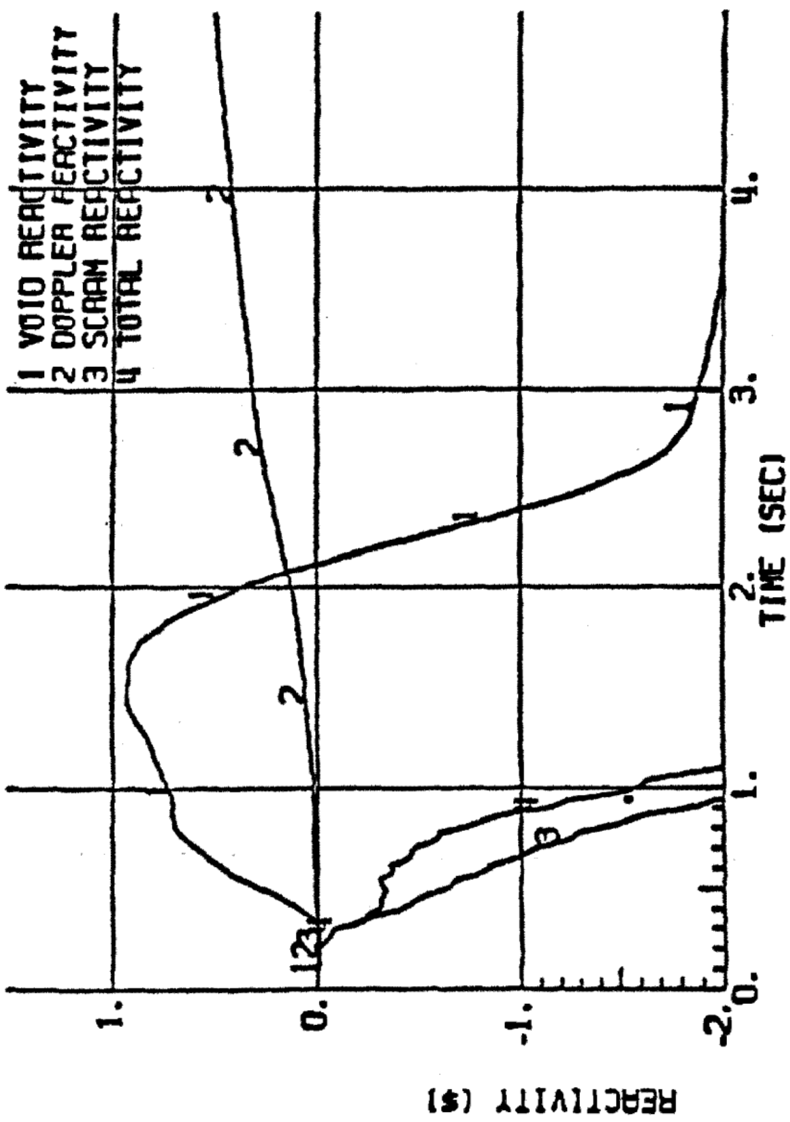
Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	GENERATOR LOAD REJECTION BYPASS FAILURE SINGLE LOOP OPERATION 71% POWER, 54% FLOW, (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15C.3-3 SHEET 2 OF 4



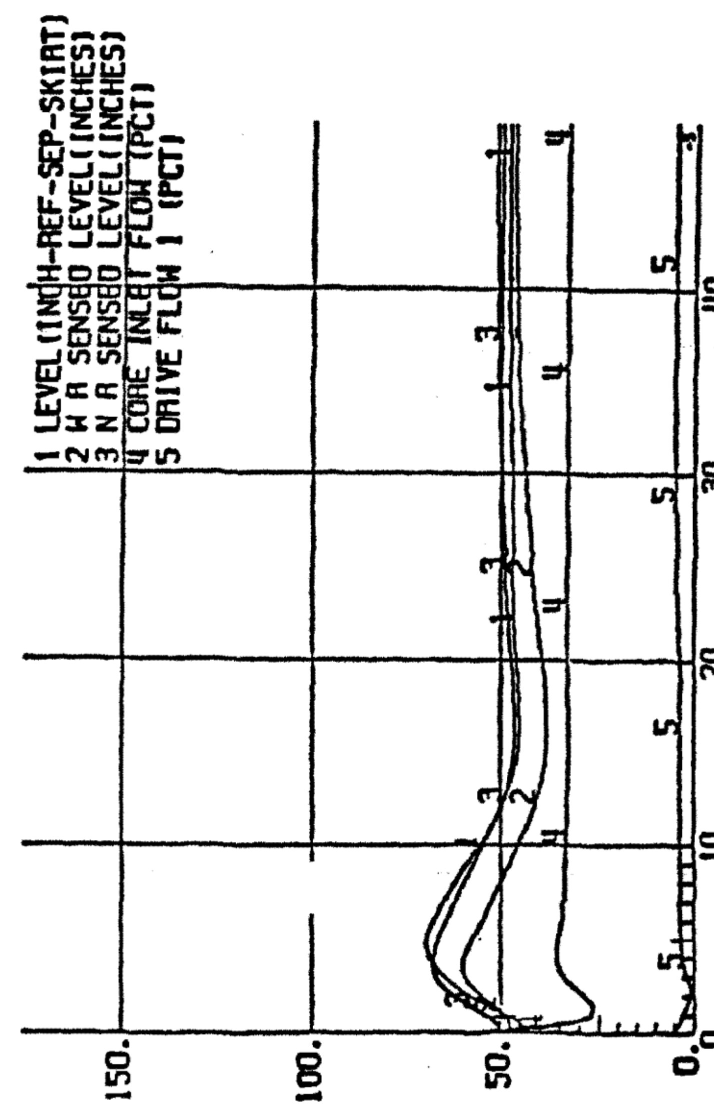
Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	GENERATOR LOAD REJECTION BYPASS FAILURE SINGLE LOOP OPERATION 71% POWER, 54% FLOW; (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15C.3-3 SHEET 3 OF 4



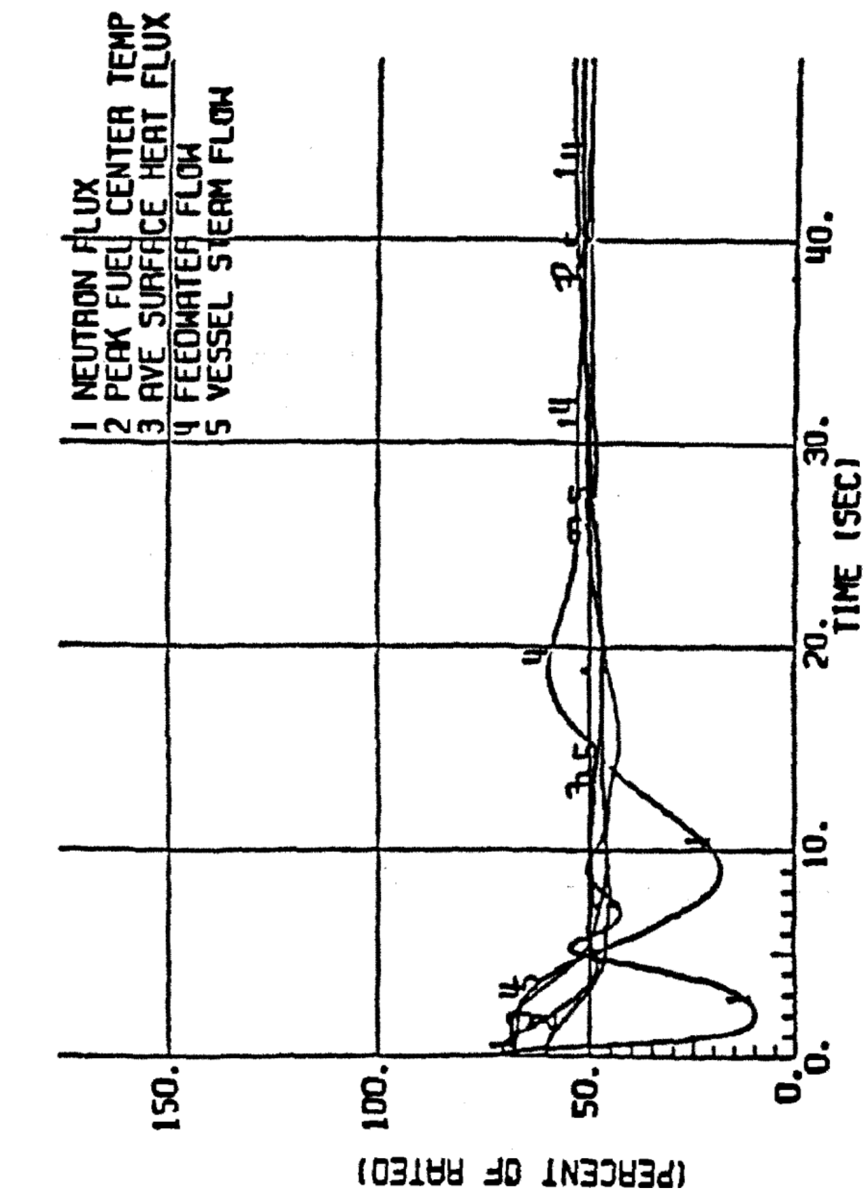
Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	GENERATOR LOAD REJECTION BYPASS FAILURE SINGLE LOOP OPERATION 71% POWER, 54% FLOW, (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15C.3-3 SHEET 4 OF 4



Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

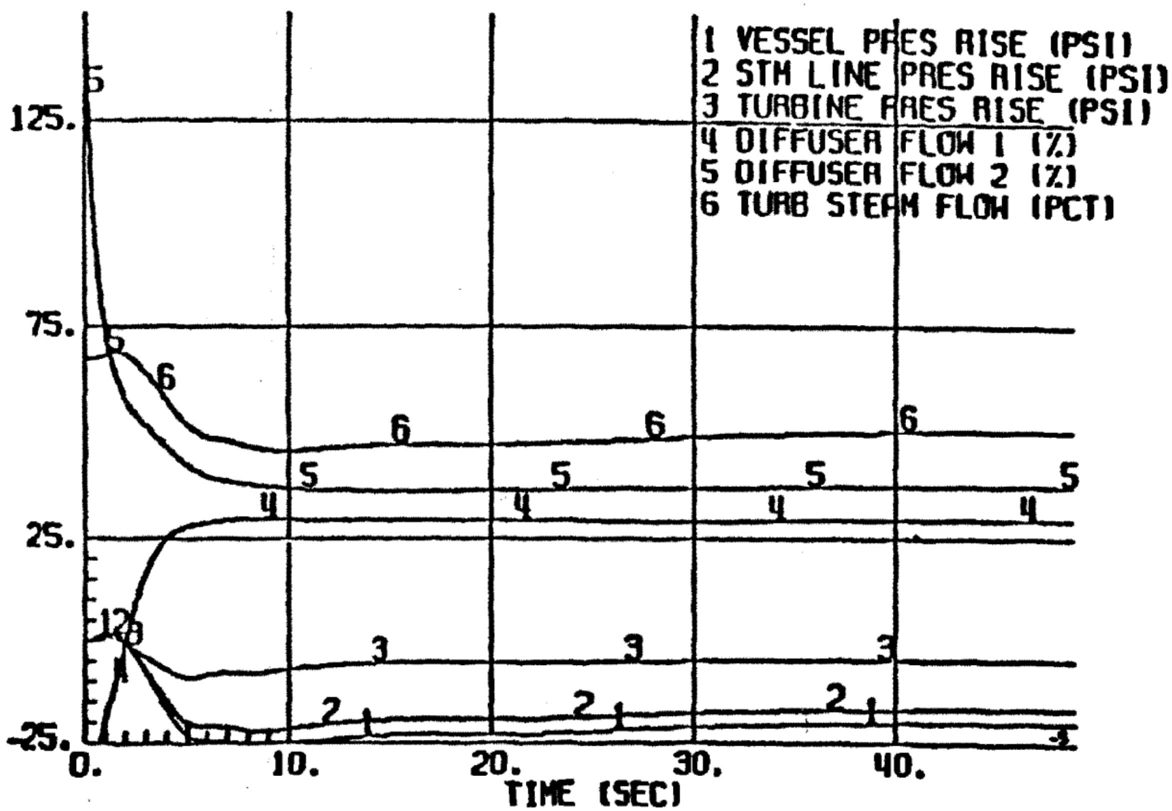
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	PUMP SEIZURE SINGLE LOOP OPERATION 71% POWER, 54% FLOW, (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15C.3-4 SHEET 1 OF 4



Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

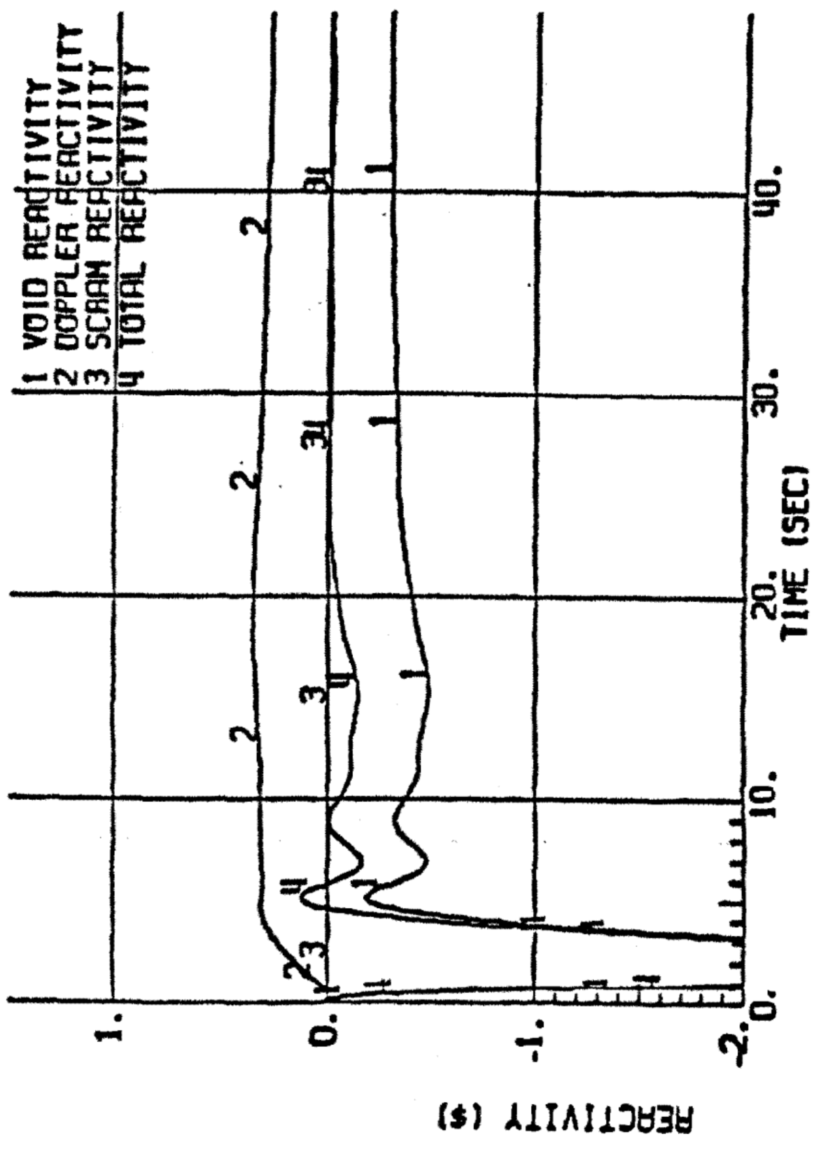
GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

PUMP SEIZURE
SINGLE LOOP OPERATION
71% POWER, 54% FLOW,
(INITIAL CYCLE) [HISTORICAL INFORMATION]
FIGURE 15C.3-4 SHEET 2 OF 4



Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	PUMP SEIZURE SINGLE LOOP OPERATION 71% POWER, 54% FLOW, (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15C.3-4 SHEET 3 OF 4



Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	PUMP SEIZURE SINGLE LOOP OPERATION 71% POWER, 54% FLOW, (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15C.3-4 SHEET 4 OF 4

15C.4 STABILITY ANALYSIS

15C.4.1 Phenomena

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [The least stable power/flow condition attainable under normal operating conditions (both reactor coolant system recirculation loops in operation) occurs at minimum flow and the highest achievable power level. For all operating conditions, the least stable power/flow condition may correspond to operation with one or both recirculation loops not in operation. The primary contributing factors to the stability performance with one or both recirculation loops not in service are the power/flow ratio and the recirculation loop characteristics. At natural circulation flow the highest power/ flow ratio is achieved.At forced circulation with one recirculation loop not in operation, the reactor core stability may be influenced by the inactive recirculation loop. As core flow increases in SLO, the inactive loop forward flow decreases because the natural circulation driving head decreases with increasing core flow. The reduced flow in the inactive loop reduces the resistance that the recirculation loops impose on reactor core flow perturbations thereby adding a destabilizing effect. At the same time the increased core flow results in a lower power/flow ratio which is a stabilizing effect. These two countering effects may result in decreased stability margin (higher decay ratio) initially as core flow is increased (from minimum) in SLO and then an increase in stability margin (lower decay ratio) as core flow is increased further and reverse flow in the inactive loop is established.

As core flow is increased further during SLO and substantial reverse flow is established in the inactive loop an increase in jet pump flow, core flow and neutron noise is observed. A cross flow is established in the annular downcomer region near the jet pump suction entrance caused by the reverse flow of the inactive recirculation loop. This cross flow interacts with the jet pump cross flow interacts with the jet pump suction flow of the active recirculation loop and increases the jet pump flow noise. This effect increases the total core flow noise which tends to drive the neutron flux noise.]

To determine if the increased noise was being caused by reduced stability margin as SLO core flow was increased, an evaluation was performed which phenomenologically accounts for single loop operation effects on stability (Reference 4). The model predictions were initially compared to test data and showed very good agreement for both two loop and single loop test conditions. An evaluation was performed to determine the effect of reverse flow on stability during SLO. With increasing reverse flow, SLO exhibited slightly lower decay ratios than two loop operation. However, at low core flow conditions with no reverse flow, SLO was slightly less stable. This is consistent with observed behavior at stability tests at operating BWRs (Reference 5).

In addition to the above analyses, the cross flow established during reverse flow conditions was simulated analytically and shown to cause an increase in the individual and total jet pump flow noise, which is consistent with tests data (Reference 4). The results of these analyses and tests indicate that the stability characteristics are not significantly different from two loop operation. At low core flows, SLO may be slightly less stable than two loop operation but as core flow is increased and reverse flow is established the stability performance is similar. At even higher core flows with substantial reverse flow in the inactive recirculation loop, the effects of cross flow on the flow noise results in an increase in system noise (jet pump, core flow and neutron flux noise).]

15C.4.2 Compliance to Stability Criteria

[HISTORICAL INFORMATION] [Consistent with the philosophy applied to two loop operation, the stability compliance during single loop operation is demonstrated on a generic basis. Stability acceptance criteria have been established to demonstrate compliance with the requirements set forth in 10CRF50, Appendix A, General Design Criterion (GDC) 12 (Reference 6). A generic analysis which covers those fuels contained in the General Electric Standard Application for Reactor fuel (Reference 7) has been performed. The analysis demonstrates that in the event limit cycle neutron flux oscillations occur within the bounds of safety system intervention, specified acceptable fuel design limits are not exceeded. Since the reactor core is assumed to be in an oscillatory mode, the question of stability margin during SLO is not relevant from a safety standpoint (i.e., the analysis already assumes no stability margin).

The fuel performance during limit cycle oscillations is characteristically dependent on fuel design and certain fixed system features (high neutron flux scram setpoint, channel inlet orifice diameter, etc.). Therefore the acceptability of GE fuel designs independent of plant and cycle parameters has been established. Only those parameters unique to SLO which affect fuel performance need to be evaluated. The major consideration of SLO is the increased Minimum Critical Power Ratio (MCPR) safety limit caused by increased uncertainties in system parameters during SLO. However, the increase in MCPR safety limit (0.01) is well within the margin of the limit cycle analyses (Reference 6) and therefore it is demonstrated that stability compliance criteria are satisfied during single loop operation. Operationally, the effects of higher flow determine the effects on fuel and channel fatigue. However, these are not considered in the compliance to stability criteria but are instead addressed on a plant specific basis. These evaluations are addressed in Section 15C.7.

A Service Information Letter-380, Revision 1 (Reference 8) has been developed to inform plant operators how to recognize and suppress unanticipated oscillations when encountered during plant operation. Evaluation of additional SLO test data taken from an operating BWR in late 1983 has been completed. Results of which have been documented in revision 1 of the Reference 6 report (NEDE-22277-P-1). These efforts combined with the analyses previously documented in References 4 and 6 provide justification that GGNS can operate at the highest achievable power with a single recirculation loop in operation.]

15C.5 LOSS-OF-COOLANT ACCIDENT ANALYSIS

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [An analysis of single recirculation loop operation using the models and assumptions documented in Reference 9 was performed for GGNS. Using this method, SAFE/REFLOOD computer code runs were made for a full spectrum of large break sizes for only the recirculation suction size breaks (most limiting for GGNS). Because the reflood minus uncovery time for the single-loop analysis is similar to the two-loop analysis, the maximum planar linear heat generation rate (MAPLHGR) curves were modified by derived reduction factors for use during one recirculation pump operation.

After a review of the NSSS vendor analyses discussed in this section, the reload fuel vendor found it necessary to evaluate the LOCA during SLO for the current cycle. The MAPLHGR for SLO is determined from the LOCA during SLO analysis. See subsection 6.3.3 for more information.]

15C.5.1 Break Spectrum Analysis

[HISTORICAL INFORMATION] [SAFE/REFLOOD calculations were performed using assumptions given in Section II.A.7.3.1 of Reference 9. Hot node uncovered time (time between uncovery and reflood) for single-loop operation is compared to that for two-loop operation in Figure 15C.5-1.

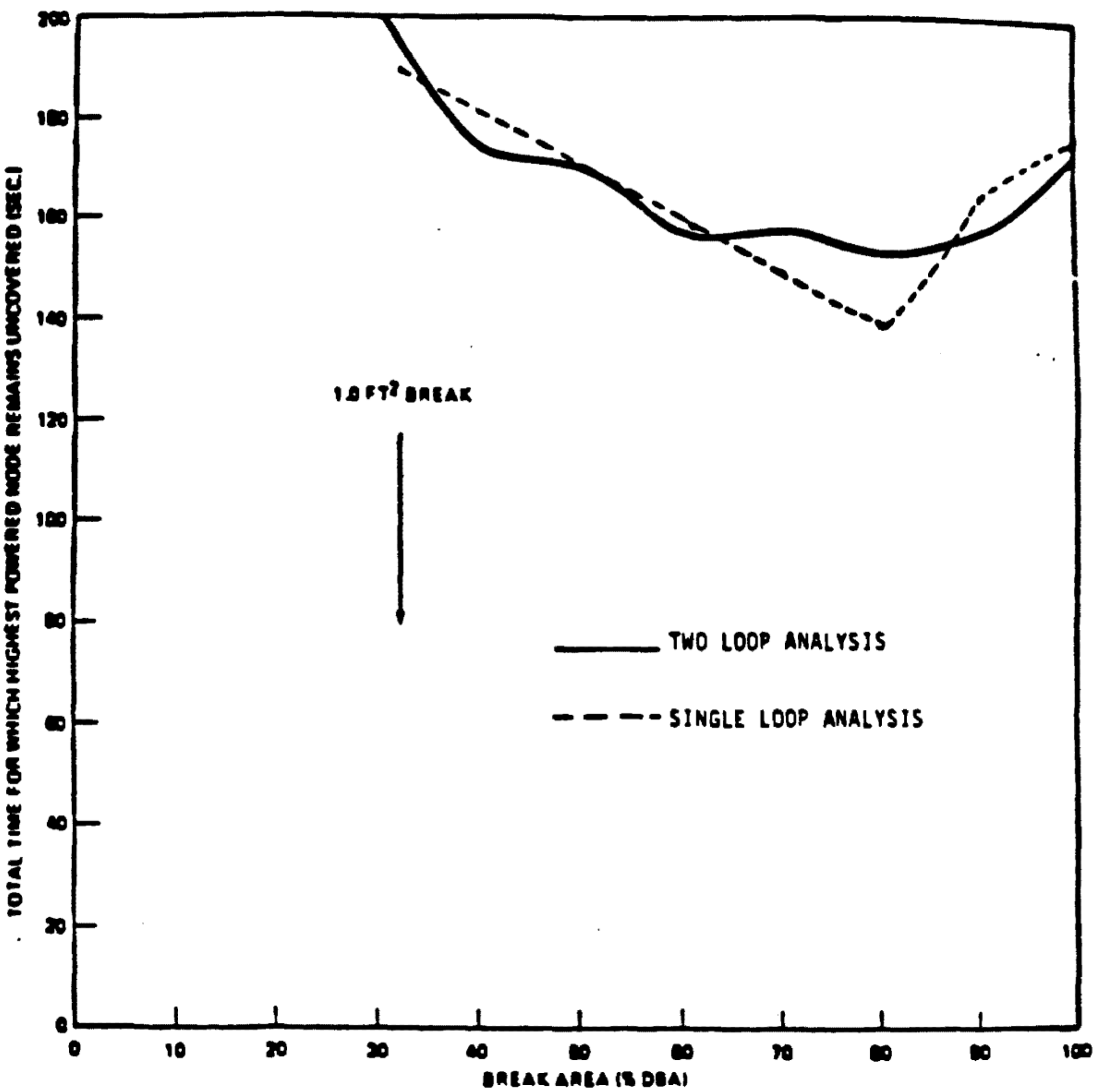
The total uncovered time for two-loop operation is 174 seconds for the 100% DBA suction break. This is the most limiting break for two-loop operation. For single-loop operation, the total uncovered time is 177 seconds and for the 100% DBA suction break. This is the most limiting break for single-loop operation. In both cases, the 1.0 ft^2 suction break has a longer total uncovered time but results in a less severe PCT response due to a later uncovery time.]

15C.5.2 Single-Loop MAPLHGR Determination

[HISTORICAL INFORMATION] [The small differences in uncovered time and reflood time for the limiting break size would result in a small change in the calculated peak cladding temperature. Therefore, as noted in Reference 9, the one and two-loop SAFE/REFLOOD results can be considered similar and the generic alternate procedure described in Section II.A.7.4 of this reference was used to calculate the MAPLHGR reduction factors for single-loop operation. The most limiting single-loop operation MAPLHGR reduction factor (i.e., yielding the lowest MAPLHGR) for GE-6 8x8 retrofit-fuel is 0.86. One-loop operation MAPLHGR values are derived by multiplying the current two-loop MAPLHGR values by the reduction factor (0.86). As discussed in Reference 9, single recirculation loop MAPLHGR values are conservative when calculated in this manner.]

15C.5.3 Small Break Peak Cladding Temperature

[HISTORICAL INFORMATION] [Section II.A.7.4.4.2 of Reference 9 discusses the low sensitivity of the calculated peak cladding temperature (PCT) to the assumptions used in the one-pump operation analysis and the duration of nucleate boiling. As this slight increase (50˚F) in PCT is overwhelmingly offset by the decreased MAPLHGR (equivalent to 300˚F to 500˚F PCT) for one-pump operation, the calculated PCT values for small breaks will be well below the 1404˚F small break PCT value previously reported for GGNS, and significantly below the 2200˚F 10CFR50.46 cladding temperature limit.]



Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	UNCOVERED TIME VERSUS BREAK AREA SUCTION BREAK, LPCS FAILURE (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15C.5-1

15C.6 CONTAINMENT ANALYSIS

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [A single-loop operation containment analysis was performed for GGNS based on a bounding analysis performed for a standard BWR6 plant. The peak wetwell pressure, peak drywell pressure, chugging loads, condensation oscillation and pool and swell containment load responses were estimated over the entire single-loop operation power/flow region.

The analysis shows peak drywell and wetwell pressures for the worst single loop operation condition of 34.5 psia and 21 psia, respectively. The corresponding differential peak drywell and wetwell pressures are 19.8 psig and 6.3 psig which is less than the 22 psig and 9.9 psig reported in Chapter 6. The chugging loads, condensation oscillation download and pool swell velocity evaluated at the worst power/flow condition during single-loop operation were also found to be bounded by the rated power analysis.]

15C.7 MISCELLANEOUS IMPACT EVALUATION

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

15C.7.1 Anticipated Transient Without Scram (ATW) Impact Evaluation

[HISTORICAL INFORMATION] [The principal difference between single loop operation (SLO) and normal two loop operation (TLO) affecting Anticipated Transient Without Scram (ATWS) performance is that of initial reactor conditions. Since the SLO initial power flow condition is less than the rated condition used for TLO ATWS analysis, the transient response is less severe and therefore bounded by the TLO analyses. All ATWS acceptance criteria are met during SLO. Therefore, SLO is an acceptable mode of operation for ATWS considerations.]

15C.7.2 Fuel Mechanical Performance

[HISTORICAL INFORMATION] [The thermal and mechanical duty for the transients analyzed have been evaluated and found to be bounded by the fuel design bases.

It is observed that due to the substantial reverse flow established during SLO both the Average Power Range Monitor (APRM) noise and core plate differential pressure noise are slightly increased. An analysis has been carried out to determine that the APRM fluctuation should not exceed a flux amplitude of ±15% of rated and the core plate differential pressure fluctuation should not exceed 3.2 psi peak to peak to be consistent with the fuel rod and assembly design bases.]

15C.7.3 Vessel Internal Vibration

[HISTORICAL INFORMATION] [A recirculation pump drive flow limit will be imposed for SLO. The highest drive flow tested during the startup test program at GGNS that show acceptable vessel internal vibration criteria will be the drive flow limit for SLO.

A preliminary assessment has been made for the expected reactor vibration level during SLO for GGNS.

Before providing the results of the assessment, it is prudent to define the term "maximum flow" during balanced 2-loop operation and single loop operation. Maximum flow for two-pump balanced operation is equal to rated volumetric core flow at normal reactor operating conditions. Maximum flow for single-pump operation is that flow obtained with the recirculation pump drive flow equal to that required for maximum flow during two-pump balanced operation. For rated reactor water temperature and pressure, this maximum flow for GGNS is about 44,600 gpm.

During the GE BWR-6 jet pump development tests at GE test facility HF2, the reactor internal components were subjected to the maximum flows, as defined above, for both two-pump balanced and single-loop operating conditions. All components were found to be within acceptance limits with the exception of in-core guide tube during single-loop operation. Due to the non-prototypical configuration of the in-core guide tube supports at HF^2, it was decided that no design changes need to be made. Instead, the in-core guide tube was to be monitored for vibration response at the Kuo Sheng 1 plant. Startup tests at the Kuo Sheng 1 plant showed all components, including the in-core guide tube during single-loop operation, to have vibration levels within acceptance limits.

From the above, it can be inferred that the vibration levels of the reactor internal components for GGNS would be expected to be within acceptance limits during single-loop operation with maximum flow as defined above. However, since GGNS reactor internals have extensive instrumentation, final and definitive conclusions can be arrived after vibration data acquisition and data reduction are completed.]

15C.8 REFERENCES

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

2. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154, October 1978.

3. R. B. Linford, "Analytical Methods of Plant Transients Evaluation for the General Electric Boiling Water Reactor," NEDO-10802, April 1973.

4. Letter, H. C. Pfefferlen (GE) to C. O. Thomas (NRC), "Submittal of Response to Stability Action Item from NRC Concerning Single-Loop Operation," September 1983.

5. S. F. Chen and R. O. Niemi, "Vermont Yankee Cycle 8 Stability and Recirculation Pump Trip Test Report," General Electric Company, August 1982 (NEDE-25445, Proprietary Information).

6. G. A. Watford, "Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria," General Electric Company, December 1982 (NEDE-22277-P, Proprietary Information).

7. "General Electric Standard Application for Reload Fuel," General Electric Company, January 1982 (NEDE-24011-P-A-4).

8. "BWR Core Thermal Hydraulic Stability," General Electric Company, February 10, 1984 (Service Information Letter-380, Revision 1).

9. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K Amendment No. 2 - One Recirculation Loop Out-of-Service," NEDO-20566-2 Revision 1, July 1978.

10. General Electric Company Report, "GGNS Feedwater Heater(s) Out-of-Service," March 1986.

APPENDIX 15D MAXIMUM EXTENDED OPERATING DOMAIN

15D.1 DEFINITION OF POWER/FLOW OPERATING DOMAINS

BWRs were originally licensed to operate at rated power and core flow (OLTP, 100% power/flow) along the flow control line. The initial cycle at GGNS was analyzed for a maximum extended load line limit (MELLLA) operating range characterized by the operating state point of reactor thermal power of 100% originally licensed thermal power (OLTP) at 75% of rated core flow. At GGNS this MELLLA operating region was combined with an increased core flow range of 105% of rated core flow into an operating power/flow map referred to as the maximum extended operating domain (MEOD).

When the OLTP was increased as a result of EPU utilizing an extension of the existing MELLLA boundary the minimum core flow was restricted at 100% thermal power to about 93% rated core flow. GGNS has since received the approved License Amendment No. 205 to operate in an expanded operating range called the MELLLA Plus (MELLLA+) operating region. Therefore, this Appendix describes the MEOD as analyzed by the NSSS vendor for the initial cycle as well as the expanded operating domain which includes MELLLA+ analyzed for EPU conditions.

15D.1.1 Definition of MEOD

The initial cycle power/flow operating domain as given in Figure 15D.3-2 can be regarded as a map bounded by the following restrictions:

(1) The 100% rated power condition.

(2) The 105% rated steam flow rod line.

(3) The 100% rated core flow condition.

(4) Low power recirculation system component cavitation restriction.

(5) Minimum core flow restrictions on pump speed FCV position.

15D.1.2 Definition of MELLLA+ Region

When the OLTP was increased through EPU, the existing MELLLA load line was extended to a state point of approximately 115% of the OLTP. This extension of the existing MEOD resulted in a narrow operating window with regards to core flow. MELLLA+ attempts to address the flow control issue by increasing the operating points above the existing MELLLA load line, potentially up to 120% OLTP and down to 80% core flow.; thus creating a 20% flow control window. An illustration of this MELLLA+ boundary area is shown in Figure 4.4-5. The power flow map for the current fuel cycle is defined on a cycle specific basis and is located in the Core Operating Limits Report (COLR).

Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.

Chapter 6 and 15 justify safe operation of the Grand Gulf Nuclear Stations (GGNS) in this defined region by evaluating all normal and abnormal transients and all design basis accidents to prove that all requirements established by the Code of Federal Regulations are met.

TABLE 15D.1-1: DELETED

15D.2 OBJECTIVES OF POWER/FLOW OPERATING DOMAINS

15D.2.1 Objectives of the MEOD

An extended power/flow operating domain is defined relative to the normal operating map in Figure 15D.3-2 for a region satisfying the following:

(1) The additional region is operational beneficial and achievable.

(2) It is safe to operate in this additional region and all requirements in the code of Federal Regulations are met.

(3) The Technical Specifications required to cover operation in the extended region do not restrict operation unduly.

(4) The hardware changes required in this region are not major changes to the GE Boiling Water Reactor (BWR) 6 standard plant definition.

This appendix will show that the operating domain as defined in Figure 15D.3-2 can be safely extended to meet all the above objectives and specifically all requirements in the Code of Federal Regulations.

15D.2.2 Objectives of the MELLLA+ Operating Region

Prior to EPU, GGNS operated in the MEOD operating domain which is characterized by the operating setpoint of reactor thermal power of 100% OLTP at 75% to 105% of rated core flow. Due to operations in EPU conditions and an extension of the MEOD boundary along the flow control line, a flow reduction or recirculation pump trip would revert approximately to the pre-OLTP operation statepoints. Uprating to 115% of the OLTP restricts the core flow to approximately 92.8% of rated flow which results in a reduced core flow range available for flexible operation at the new rated power. The addition of the MELLLA+ operating region increases the available core flow window at the uprated power level from 80% to the maximum licensed core flow. This represents a significant improvement on operating flexibility (Ref. 10).

An expanded power/flow operating domain is defined relative to the MEOD operating map to include a MELLLA+ region as shown in Figure 4.4-5. The current licensed operating domain is maintained in the COLR. This appendix will show that plant operations in the MELLLA+ operating region is a safe expansion of the MEOD and will continue to meet all the objectives and requirements described above for the MEOD.

15D.3 INTRODUCTION & SUMMARY

15D.3.1 Introduction and Summary of MEOD

This appendix presents the results of a safety and impact evaluation for operation of the Grand Gulf Nuclear Station in a modified operating envelop called the Maximum Extended Operating Domain (MEOD). The MEOD region can be utilized to improve the operating flexibility and capacity factor for the Grand Gulf Nuclear Station.

If the rated load line control rod pattern is maintained as core flow is increased, changing equilibrium xenon concentrations will result in less than rated power at rated core flow. In addition, fuel pellet-cladding interaction considerations inhibit withdrawal of control rods at high power levels. The combination of these factors can result in the inability to attain rated core power directly.

The maximum extended operating domain as defined and illustrated in Figure 15D.3-1 permits improved power ascension capability to full power and provides additional flow range at rated power including an increased flow region to compensate for reactivity reduction due to exposure during an operating cycle. This expanded power flow map can be separated into two regions. One is the expanded operation in the lower than 100% core flow region which is termed Maximum Extended Load Line Limit (MELLLA) Region and the other is the expanded region in the higher than 100% core flow region which is termed Increased Core Flow Region (ICFR). The combined MELLLA region and ICFR is termed Maximum Extended Operating Domain (MEOD) in the remaining content of this appendix.

The MEOD analysis consists of three features:

(a) Operation in the MELLLA region

(b) Operation in the ICFR

(c) Elimination of the APRM Total Peaking Factor Setdown Technical Specification Requirement.

The MELLLA region boundary is limited by 75% core flow at 100% of the originally licensed power and its corresponding power/flow constant rod line. This is determined based on a safety and impact evaluation as well as a feasibility study that indicates that this is the highest rod line that is operationally feasible in meeting thermal and reactivity margins. The ICFR is bounded by the 105% core flow line. This ICFR boundary is limited by plant recirculation system capability, acceptable flow induced vibration limit and reactor internal pressure differences plus the impact of fuel bundle lift forces on the vessel internal components.

The limiting normal and abnormal operating transients in Chapter 15 were reevaluated in the MEOD. No change in power dependent operating limit $MCPR_p$ was made when operating in the MEOD. However, a new set of power dependent operating limit $MCPR_p$ was necessary as a result of elimination of the APRM trip setdown requirement. A new set of flow dependent operating limit $MCPR_f$ was also developed for operation in the MEOD. It was also determined that the fuel mechanical limits are met for all transients occurring in the MEOD.

Overpressure protection analyses were also performed in the MEOD. It was concluded that peak vessel pressures for the MEOD conditions are below the ASME code limit. Therefore, adequate pressure protection is present in the MEOD.

The Loss of Coolant Accident and Containment responses as described in Chapter 6 were reevaluated in the MEOD. It was found that the responses are bounded by the current design analysis.

Thermal hydraulic stability was evaluated for its adequacy with respect to the General Design Criterion 12 (10CFR50, Appendix A). It is shown that MEOD operation satisfies this stability criterion.

The effect of increased reactor internal pressure differences, acoustic loads, flow induced loads and fuel bundle lift forces on the reactor internal components and fuel channels due to increased core flow were evaluated to show that the design limits are not exceeded. The effect of increased flow rate on the flow-induced vibration responses of the reactor internals was monitored during startup testing and evaluated to ensure the responses are within acceptable limits for Grand Gulf Station.

Several impact evaluations were also performed to justify operation in the MEOD. This includes impact of the MEOD on Anticipated Transient Without Scram (ATWS), fuel assemblies, fuel channel bypass flow, creep and control blade interference. It was found that all acceptance criteria and design limits are met.

The Average Power Range Monitor (APRM) simulated thermal power scram and rod block configuration are redefined to accommodate operation in the MEOD. The same protection margin as the current configuration is maintained in the new definition.

This appendix also justifies operation of Feedwater Heater(s) Out of Service as described in Appendix 15B in the MEOD (MELLLA region and ICFR). All evaluations described in Appendix 15B were reevaluated or reviewed in the MEOD to ensure that FWHOS operation in this region is safe and feasible.

Finally, this appendix justifies the replacement of the APRM trip setdown requirements by more meaningful power and flow dependent Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits to reduce the need for manual setpoint adjustment and to allow for more direct administration of limits. New power dependent $MCPR_p$ limit requirements are also established.

Lastly, due to the different MCPR limits required as discussed in the main text, Appendix 15C for single loop operation and Appendix 15B for feedwater heater(s) out of service, a new power dependent MCPR multiplier (K_p) is developed to simplify the $MCPR_p$ implementation.

15D.3.2 Objectives of the MELLLA+ Operating Region

Prior to EPU, GGNS operated in the MEOD operating domain which is characterized by the operating setpoint of reactor thermal power of 100% OLTP at 75% to 105% of rated core flow. Due to the new operating conditions associated with EPU, and an extension of the MEOD boundary along the flow control line, a flow reduction or recirculation pump trip would revert approximately to the pre-OLTP operation statepoints. Uprating to 115% of the OLTP then restricts the core flow to approximately 92.8% of rated flow which results in a reduced core flow range available for flexible operation at the new rated power.

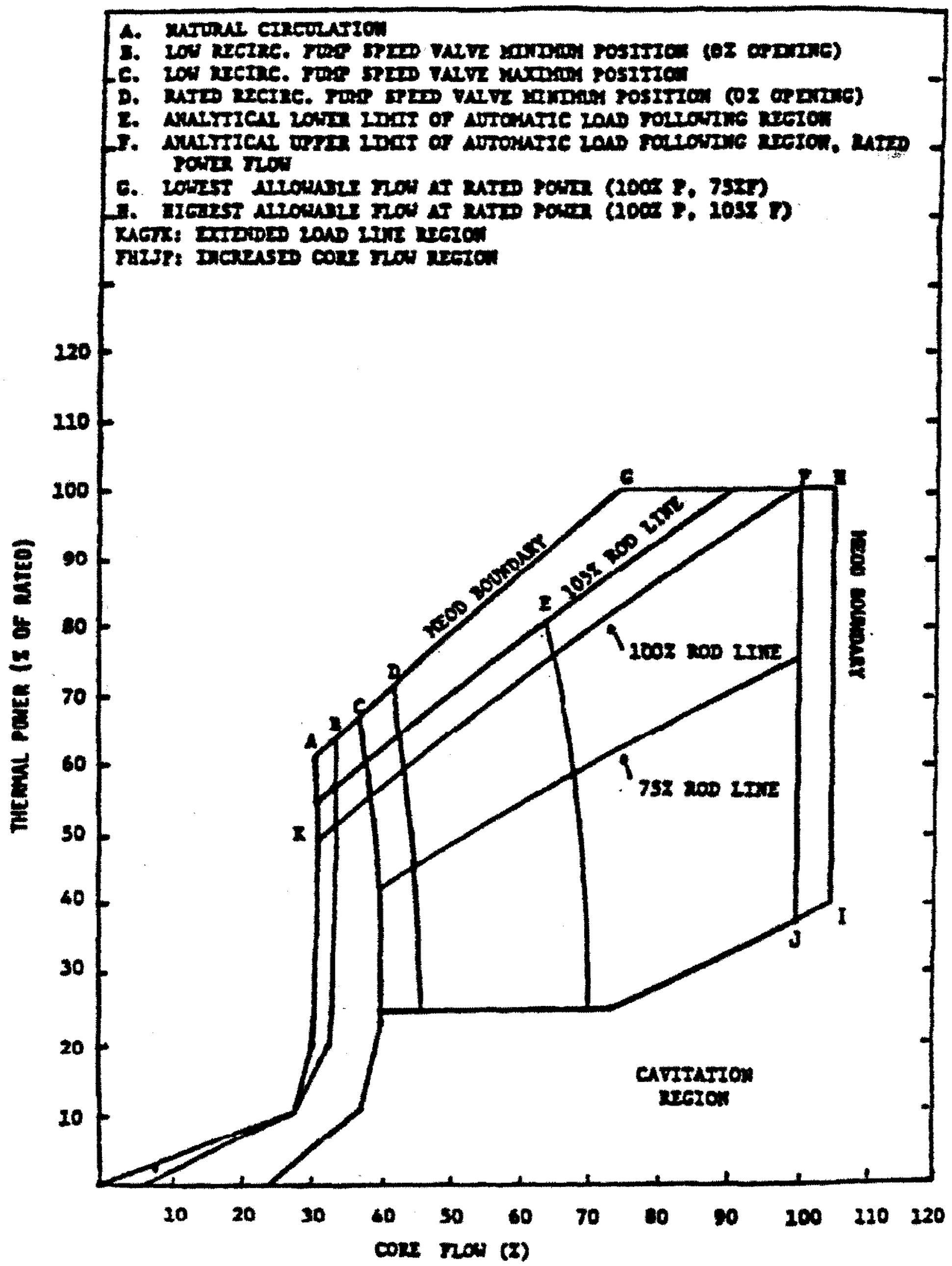
Day-to-day operation of the reactor requires that reactivity balance be maintained to accommodate fuel burn-up. The BWR operators have typically two options to maintain this reactivity balance: (a) control rod movements or (b) flow adjustments. Because of the strong void reactivity feedback and its distributed effect all over the core, flow adjustments are the preferred reactivity control method. Control rod movements are typically performed a few times during the cycle to accomplish larger reactivity changes and the desired burn-up profiles. Because of the strong local power changes that may result from control rod motion and its local effect on the fuel, control rod movements should be performed very slowly and at a reduced power level; otherwise, fuel clad failures may occur.

The preferred reactivity control method is to set up a target control rod pattern at a low power level, increase the power to full licensed conditions and control reactivity by increasing flow over a period of several months. When the burn-up reactivity can no longer be adjusted using flow, the power level is reduced, the next target control rod sequence is achieved, the power is increased back to the licensed level, and flow control continues to maintain power. Following EPU the flow-control window can be very small, therefore, reactor operators are forced to either move control rods very often or allow power changes as bum-up takes place. In a typical EPU reactor, the control rods must be repositioned almost on a weekly basis to maintain power at the licensed level.

MELLLA+ attempts to address this flow control issue at GGNS by increasing the operating range to up to approximately 115% OLTP and 80% flow; thus creating a 20% flow control window. Hence, GGNS operations in the MELLLA+ range require significantly lower number of control rod movements than without MELLLA+. This represents a significant improvement on operating flexibility. It also provides safer operation, because reducing the number of control rod manipulations; (a) minimizes the likelihood of fuel failures and (b) reduces the likelihood of accidents initiated by reactor maneuvers required to achieve an operating condition where control rods can be extracted.

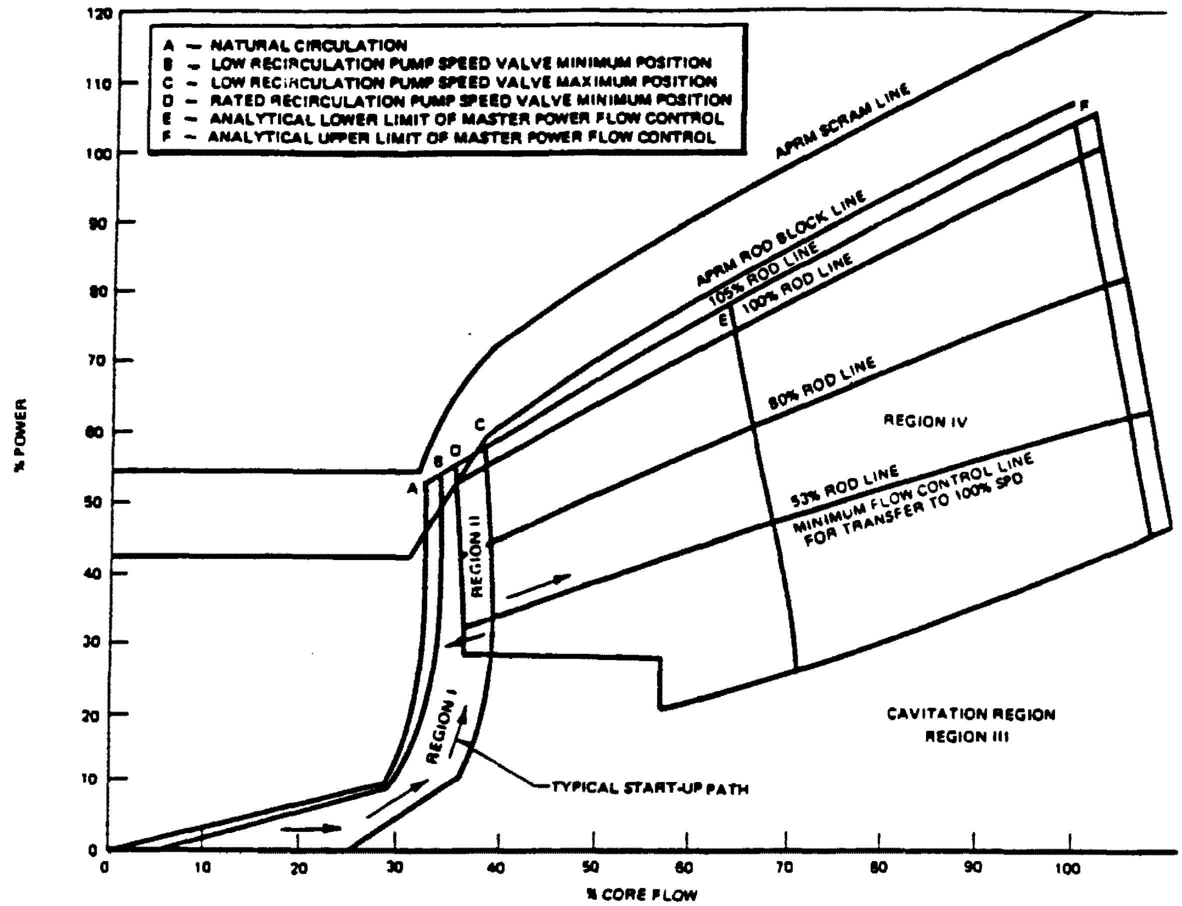
A secondary benefit from MELLLA+ operation is spectral shifting. Operation at high power-to-flow ratios results in high void fractions, and the reduced water-moderation of neutrons increases the neutron average energy. At higher neutron energies, the Uranium-238 absorption cross section increases, and more

Plutonium-239 is produced. Since Plutonium-239 is a fissile isotope, it increases the core reactivity and, essentially, adds production days to the fuel cycle. Towards the end of the cycle, approximately 30% of the nuclear energy is produced by fission of the Plutonium-239 as opposed to Uranium-235 (Ref. 10).



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	MAXIMUM EXTENDED OPERATING DOMAIN POWER/FLOW MAP (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15D.3-1

Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.



NOTE: THE ABOVE FIGURE IS A PRESENTATION OF TYPICAL OPERATING REGIONS FOR A BWR/6 REACTOR.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	POWER FLOW OPERATING MAP FIGURE 15D.3-2

15D.4 MCPR OPERATING LIMIT

15D.4.1 Abnormal Operating Transients

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [All abnormal operating transients described in Chapter 15 were examined for Maximum Extended Operating Domain (MEOD) operation. A bounding analysis is performed using a standard BWR/6 238 size 748 bundle plant with the highest enriched GE-6 fuel type as a basis. The core average power density of this standard BWR 6 plant is almost identical to the 251 size 800 bundle GGNS. The fuel type used in this analysis represents the bounding nature of this analysis. This bounding analysis was performed at various MEOD boundary power/flow conditions of Figure 15D.3-1 at the end of the 18 month equilibrium cycle using both the computer models described in References 1 and 2.

The various power/flow conditions and transients analyzed are tabulated in Table 15D.4-1. The CPR results of this analysis are also tabulated in Table 15D.4-1. This bounding evaluation concluded that the CPR results for all the cases analyzed in the MEOD are bounded by the current power dependent $MCPR_p$ limits.

The following limiting pressurization and cold water injection abnormal operating transients were reevaluated in detail for the Grand Gulf Station to confirm that the bounding analysis performed for equilibrium cycle is bounding for GGNS at Cycle 1:

1. Generator Load Rejection With Bypass Failure (LRNBP)

2. Feedwater Flow Controller Failure (FWCF)

The transients were analyzed at the end of Cycle 1. An initial power of 104.2% of the originally licensed rated power was used. The core flow condition chosen for the analysis were the minimum flow of 73.8% of rated and maximum achievable flow of 108% of rated. Plant heat balance, core coolant hydraulics and nuclear transient parameter data were developed and used in the above transient analysis. The initial condition for these lowest and highest flow analysis point at rated power is presented in Tables 15D.4-2 and 15D.4-3. The computer model described in Reference 1 was used to simulate these events. The transient peak values results and critical power ratio (CPR) results for the cases analyzed at 104.2% of initially licensed power (lowest and highest flow) are summarized in Tables 15D.4-4 and 15D.4-5 respectively. The transient responses are presented in Figures 15D.4-1 to 15D.4-4. The results of this Grand Gulf unique evaluation show that the ΔCPR results for all the cases analyzed are bounded by the bounding BWR/6 Standard Plant analysis presented in Table 15D.4-1 (see additional discussion provided in Addendum 2 to Appendix 15D). Therefore, no change in $MCPR_p$ limits are required for operation in the MEOD. Section 15D.14 provides a new set of $MCPR_p$ limits to support the elimination of APRM trip setdown requirement.

The BWR/6 standard plant bounding overpressure protection transient analysis using the computer model described in Reference 1 is performed at the power flow conditions described in Table 15D.4-1. The bounding MSIV closure flux scram event resulted in a peak pressure of 1273 psi at 110% core flow condition. This result is verified for the GGNS which results in a peak pressure of 1262 psig at the 108% ICF condition. Therefore, it is shown that the peak vessel pressures for the MEOD conditions are below the ASME code limit of 1375 psig. Hence, adequate pressure margin is present in the MEOD.

The lOO°F Loss of Feedwater Heater (LFWH) Transient results described in the main text of this chapter are applicable to the MEOD. A generic statistical LFWH analysis using the computer model described in Reference 3 and methodologies described in Reference 4 was performed utilizing a large data base throughout the power/flow map. It is found that the LFWH responses initiated from all off-rated power/flow conditions (including MEOD) are bounded by the generic rated condition 95% probability 95% confidence values. This generic conservative CPR value is 0.10 which is bounded by the values described in Chapter 15. GGNS plant specific results are presented in Addendum 1 to Appendix 15D.]

15D.4.2 Rod Withdrawal Error

[HISTORICAL INFORMATION] [The rod withdrawal error (RWE) transient documented in Chapter 15 is analyzed using a statistical evaluation of the minimum critical power ratio (MCPR) and linear heat generation rate (LHGR) responses to the withdrawal of ganged control rods throughout the operating power/ flow map including the MEOD region (see Figure 15.B.-9 of GESSAR II 238 Nuclear Island). Therefore, the current MCPR limit is adequate to protect the RWE in the MEOD.]

15D.4.3 Flow Runout Transient

[HISTORICAL INFORMATION] [The current flow dependent MCPR operating limit ($MCPR_f$) was determined based on the slow recirculation flow runout transient event. This curve was generated with some contingent conservative margins in the original design process. This event was reanalyzed, as part of the MEOD program to include the highest rod line for the MELLLA, up to 102.5% maximum flow and the ICFR, up to 107% maximum core flow. This analysis utilized the latest design procedure in which some of the unnecessary original contingent design conservatism were removed. The new flow dependent $MCPR_f$ curves are presented in Figure 15D.4-5. It is also shown that these curves still bound the other flow dependent abnormal transients which are considered to establish the flow dependent MCPR. limits. For additional information refer to Addendum 2 to Appendix 15D.]

15D.4.4 Operating Limit MCPR

[HISTORICAL INFORMATION] [The analyses presented in the above subsections concluded that the current power dependent $MCPR_p$ limits are adequate for operation in the MELLLA and ICFR of MEOD. However, new $MCPR_p$ limits are required to eliminate the APRM trip setdown Technical Specification requirement. The flow dependent $MCPR_f$ curves are to be modified as described in Figure 15D.5-5 for operation in the MEOD. Section 15D.14 describes the set of $MCPR_p$ limits to support the elimination of APRM trip setdown requirements.]

TABLE 15D.4-1: ANALYSIS POWER-FLOW POINTS AND CPR RESULTS FOR BWR/6 BOUNDING TRANSIENT EVALUATION (INITIAL CYCLE) [HISTORICAL INFORMATION]

Power (%)/Flow (%)	Transients	CPR[a]
70/40	LRNBP	0.071
	FWCF	0.095
	CLDLP	0.087
	FCVO	0.208[b]
83/55	LRNBP	0.066
104.2/75	LRNBP	0.076
	FWCF	0.084
	FCVO	0.072
104.2/100	LRNBP	0.110
	FWCF	0.095
104.2/110	LRNBP	0.114
	FWCF	0.097
53.5/116	LRNBP	0.125
	FWCF	0.284

(a) Option A adders included for the Reference 1 analysis.
(b) This transient is covered by $MCPR_f$, not $MCPR_p$.

NOTE:

LRNBP Generator Load Rejection With Bypass Failure
FWCF Feedwater Controller Failure (maximum demand)
CLDLP Cold Loop Startup
FCVO Flow Control Valve Opening

The LRNBP and FWCF transients are analyzed using Reference 1 and the CLDLP and FCVO transients are analyzed using Reference 2.

Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.

TABLE 15D.4-2: INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS AND ACCIDENTS FOR MEOD, 104.2% POWER, 73.8% FLOW (INITIAL CYCLE) [HISTORICAL INFORMATION]

1.	Thermal Power Level, MWt Analysis Value	3994 (104.2% rated)
2.	Steam Flow, mlb per hr Analysis Value	17.22
3.	Core Flow, mlb per hr	83.0
4.	Feedwater Flow Rate, mlb per hr Analysis Value	17.22
5.	Feedwater Temperature, °F	425
6.	Vessel dome pressure, psig	1045
7.	Core exit pressure, psig	1053
8.	Turbine Bypass Capacity, % NBR	35
9.	Core Coolant Inlet Enthalpy Btu per lb	522
10.	Turbine Inlet Pressure, psig	961
11.	Fuel Lattice	8x8R
12.	Core Leakage Flow, %	10.65
13.	Required MCPR Operating Limit First Core	1.27
14.	MCPR Safety Limit for Incidents of Moderate Frequency	
	First Core	1.06
	Reload Cores	1.07
15.	Doppler Coefficient (-)¢/°F Analysis Data	0.132(a)
16.	Void Coefficient (-)¢/% Rated Voids	
	Analysis Data for Power Increase Events	14.0 (a)
	Analysis Data for Power Decrease Events	4.0 (a)

TABLE 15D.4-2: INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS AND ACCIDENTS FOR MEOD, 104.2% POWER, 73.8% FLOW (INITIAL CYCLE) [HISTORICAL INFORMATION] **(CONTINUED)**

17. Core Average Rated Void Fraction, %	48
18. Jet Pump Ratio, M	2.25
19. Safety/Relief Valve Capacity, % NBR	
@1145 psig	102.4
Manufacturer	Dikker
Quantity Installed	20
20. Relief Function Delay, seconds	0.4
21. Relief Function Response Time Constant, sec.	0.1
22. Setpoints for Safety/Relief Valves	
Safety Function, psig	1175,1185,1195,1205, 1215
Relief Function, psig	1145,1155,1165,1175
23. Number of Valve Groupings Simulated	
Safety Function, No.	5
Relief Function, No.	4
24. High Flux Trip, % NBR	
Analysis Setpoint (122x1.042), % NBR	127.1
25. High Pressure Scram Setpoint, psig	1095
26. Vessel Level Trips, Feet Above Separator	
Skirt Bottom	
Level 8 - (L8), feet	5.88
Level 4 - (L4), feet	4.03
Level 3 - (L3), feet	2.16
Level 2 - (L2),	(-) 2.182
27. APRM Thermal Trip	
Setpoint, % NBR	118.8
28. RPT Delay, seconds	0.19
29. RPT Inertia Time Constant for Analysis, sec.	5

TABLE 15D.4-2: INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS AND ACCIDENTS FOR MEOD, 104.2% POWER, 73.8% FLOW (INITIAL CYCLE) [HISTORICAL INFORMATION] **(CONTINUED)**

30. Total Steamline Volume, ft^3	4358

(a) These values for Reference 2 analysis only. Reference 1 values are calculated within the code.

Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.

TABLE 15D.4-3: INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS AND ACCIDENTS FOR MEOD, 104.2% POWER, 108.0% FLOW (INITIAL CYCLE) [HISTORICAL INFORMATION]

1.	Thermal Power Level,MWt Analysis Value	3994 (104.2% rated)
2.	Steam Flow, mlb per hr Analysis Value	17.29
3.	Core Flow, mlb per hr	121.5
4.	Feedwater Flow Rate, mlb per hr Analysis Value	17.29
5.	Feedwater Temperature, °F	425
6.	Vessel dome pressure, psig	1045
7.	Core exit pressure, psig	1056
8.	Turbine Bypass Capacity, % NBR	35
9.	Core Coolant Inlet Enthalpy Btu per lb	532
10.	Turbine Inlet Pressure, psig	959
11.	Fuel Lattice	8x8R
12.	Core Leakage Flow, %	10.65
13.	Required MCPR Operating Limit First Core	1.18
14.	MCPR Safety Limit for Incidents of Moderate Frequency	1.06
15.	Doppler Coefficient (-)¢/°F Analysis Data	0.132(a)
16.	Void Coefficient (-)¢/% Rated Voids	
	Analysis Data for Power Increase Events	14.0 (a)
	Analysis Data for Power Decrease Events	4.0 (a)

TABLE 15D.4-3: INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS AND ACCIDENTS FOR MEOD, 104.2% POWER, 108.0% FLOW (INITIAL CYCLE) [HISTORICAL INFORMATION] **(CONTINUED)**

17. Core Average Rated Void Fraction, %	41.0
18. Jet Pump Ratio, M	2.25
19. Safety/Relief Valve Capacity, % NBR	
@1145 psig	102.4
Manufacturer	Dikker
Quantity Installed	20
20. Relief Function Delay, seconds	0.4
21. Relief Function Response Time Constant, sec.	0.1
22. Setpoints for Safety/Relief Valves	
Safety Function, psig	1175,1185,1195,1205, 1215
Relief Function, psig	1145,1155,1165,1175
23. Number of Valve Groupings Simulated	
Safety Function, No.	5
Relief Function, No.	4
24. High Flux Trip, % NBR	
Analysis Setpoint (122x1.042), % NBR	127.1
25. High Pressure Scram Setpoint, psig	1095
26. Vessel Level Trips, Feet Above Separator	
Skirt Bottom	
Level 8 - (L8), feet	5.88
Level 4 - (L4), feet	4.03
Level 3 - (L3), feet	2.16
Level 2 - (L2), feet	(-) 2.182
27. APRM Thermal Trip	
Setpoint, % NBR	118.8
28. RPT Delay, seconds	0.19

TABLE 15D.4-3: INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS AND ACCIDENTS FOR MEOD, 104.2% POWER, 108.0% FLOW (INITIAL CYCLE) [HISTORICAL INFORMATION] **(CONTINUED)**

29. RPT Inertia Time Constant for Analysis, sec.	5
30. Total Steamline Volume, ft^3	4358

(a) These values for Reference 2 analysis only. Reference 1 values are calculated within the code.

Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.

TABLE 15D.4-4: SUMMARY OF TRANSIENT PEAK VALUE RESULTS - 104.2% POWER MEOD[a] (INITIAL CYCLE) [HISTORICAL INFORMATION]

Transient	**Core Flow (% NBR)**	**Peak Neutron Flux (% NBR)**	**Peak Dome Pressure (psig)**	**Peak Vessel Pressure (psig)**	**Peak Steamline Pressure (psig)**	**Figure**
Load Rejection with Bypass Failure	108[b]	135	1200	1236	1209	15D.4-1
"	73.8	104	1205	1230	1209	15D.4-2
Feedwater Controller Failure, Max. Demand	108[b]	111	1163	1195	1162	15D.4-3
"	73.8	111	1169	1190	1167	15D.4-4

(a) Feedwater is 425°F.

(b) Maximum achievable core flow with 425°F feedwater.

Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.

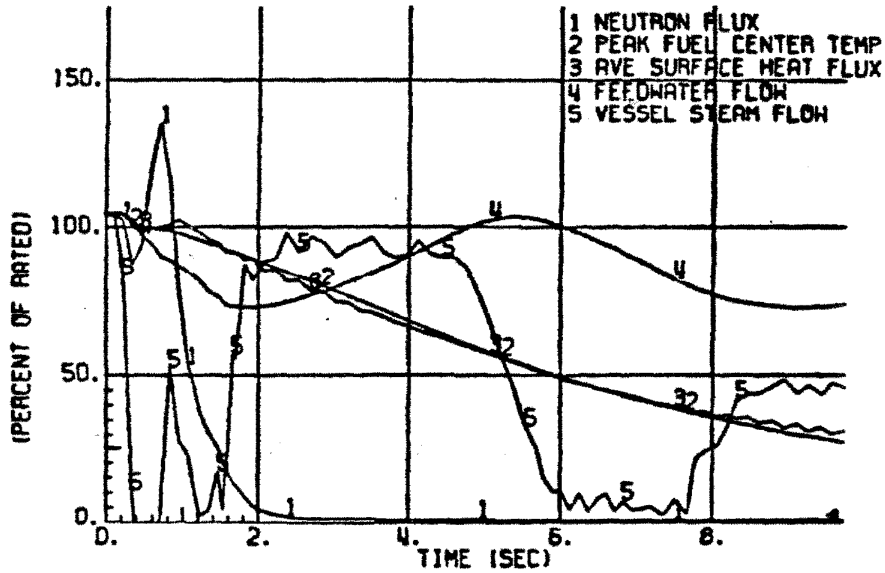
TABLE 15D.4-5: [HISTORICAL INFORMATION] SUMMARY OF CPR RESULTS - 104.2%[a] (INITIAL CYCLE)

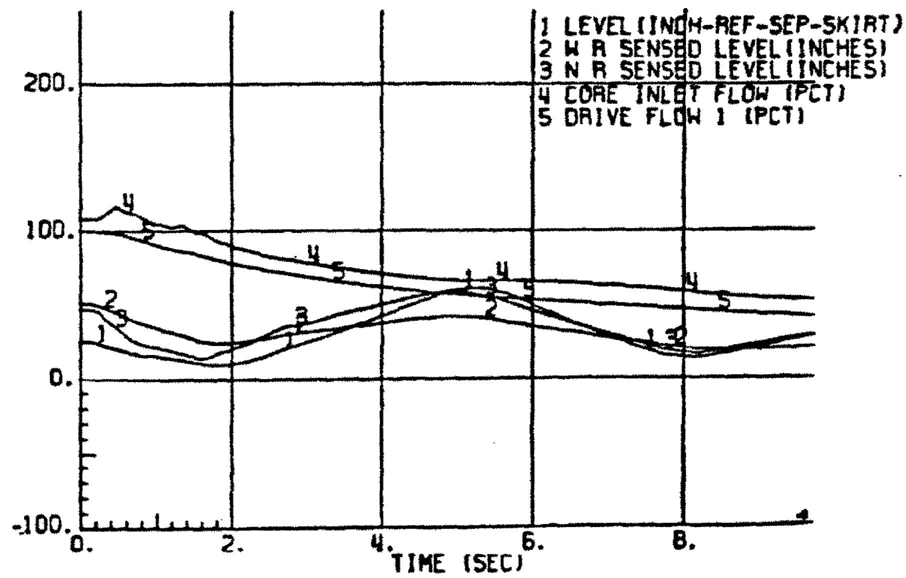
Transient	Core Flow (% NBR)	ICPR	CPR	MCPR
Load Rejection with Bypass Failure	108[b]	1.18	0.05	1.13
"	100.0	1.18	0.05	1.13
"	73.8	1.27	0.05	1.22
Feedwater Controller Failure, Max. Demand	108[b]	1.18	0.09	1.09
"	100.0	1.18	0.09	1.09
"	73.8	1.27	0.09	1.18

(a) Feedwater is 425°F. Option A adders included

(b) Maximum achievable core flow with 425°F feedwater.

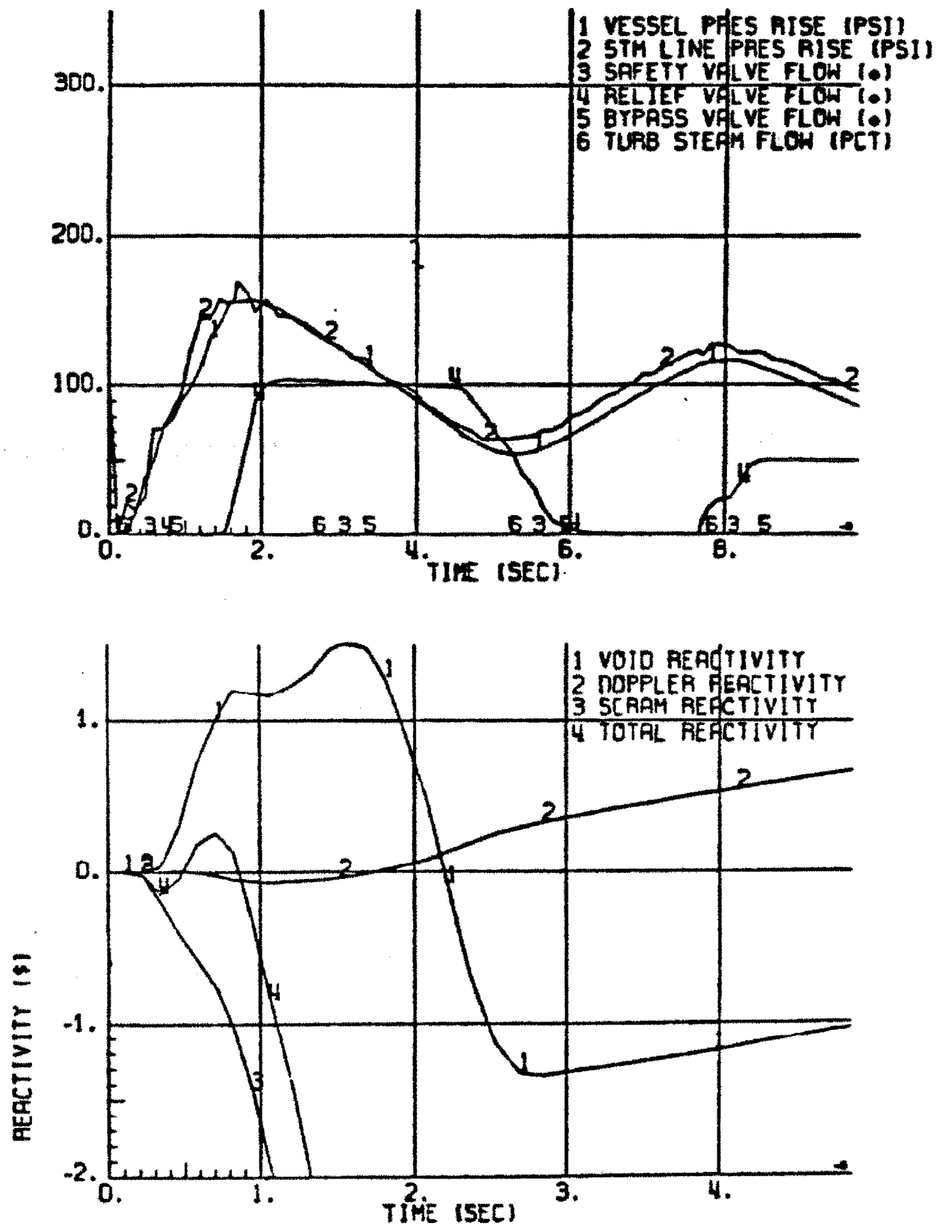
Note: Initial cycle analyses are based on the originally licensed power level of 3833 MW.





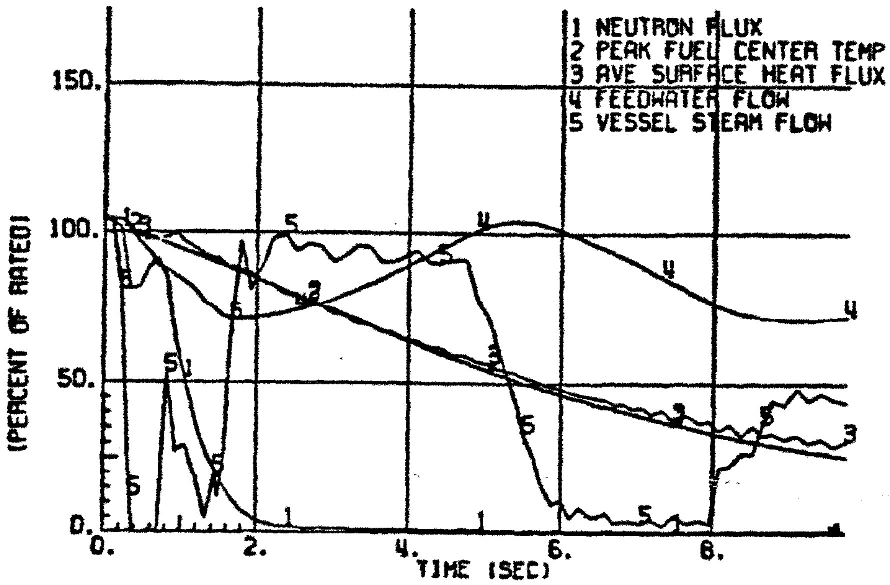
Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

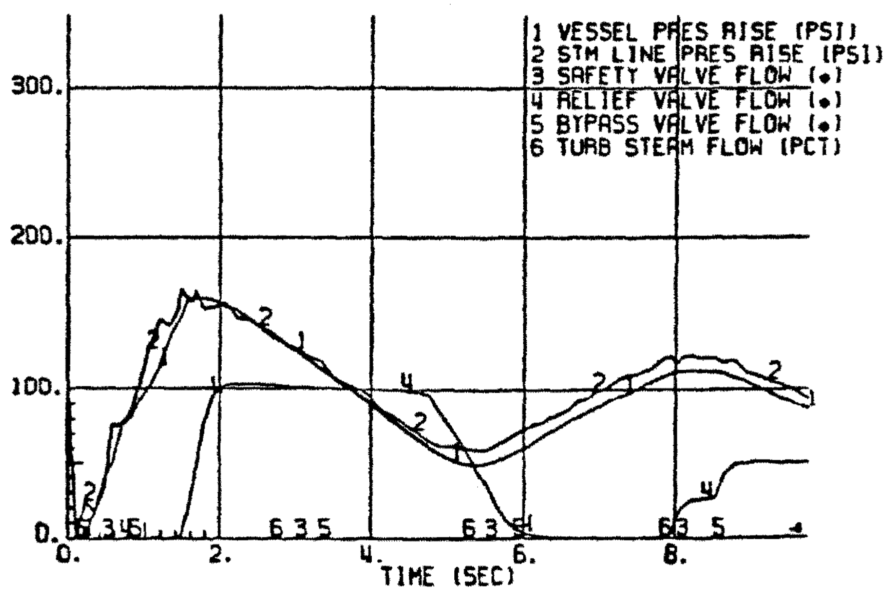
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	GENERATOR LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER 108% FLOW EOC1 [HISTORICAL INFORMATION] FIGURE 15D.4-1 SHEET 1 OF 2



Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

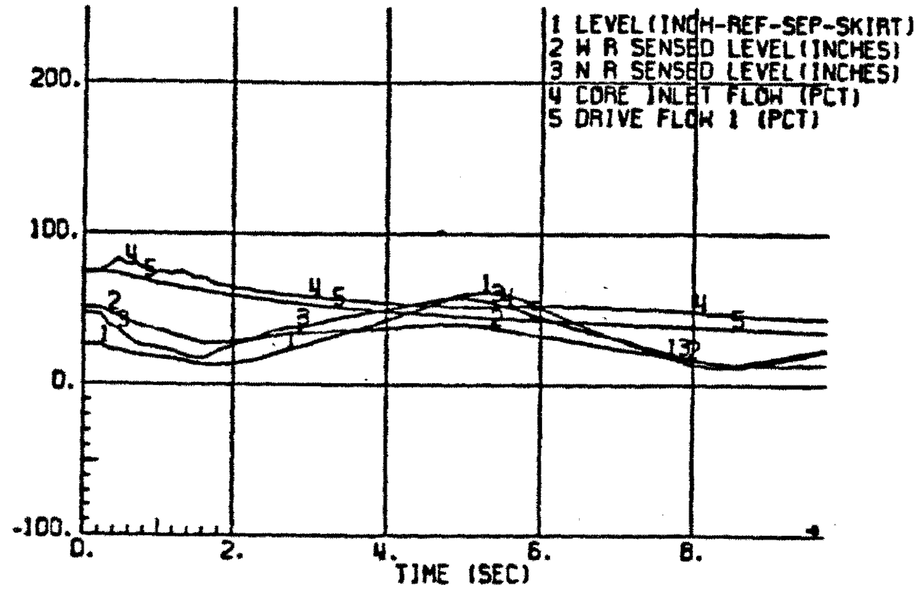
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	GENERATOR LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER 108% FLOW EOC1 [HISTORICAL INFORMATION] FIGURE 15D.4-1 SHEET 2 OF 2

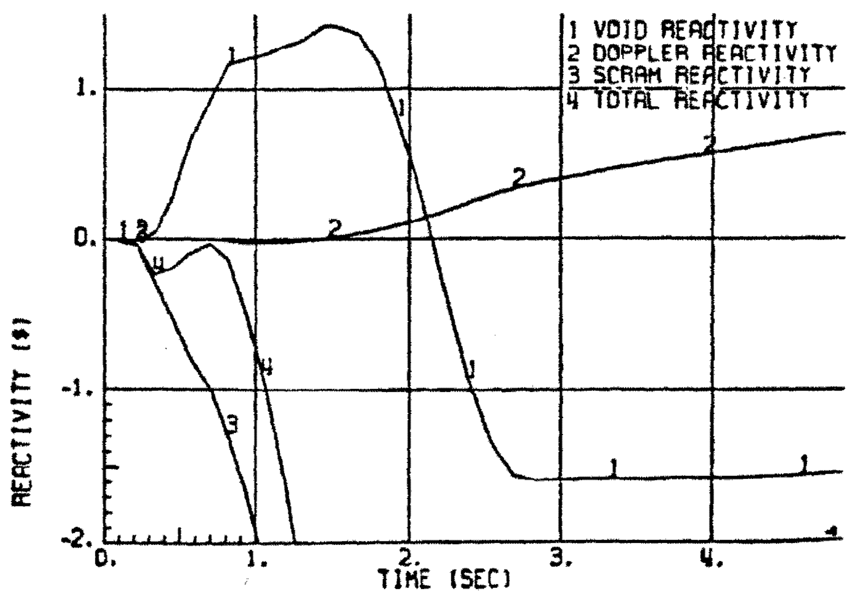




Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

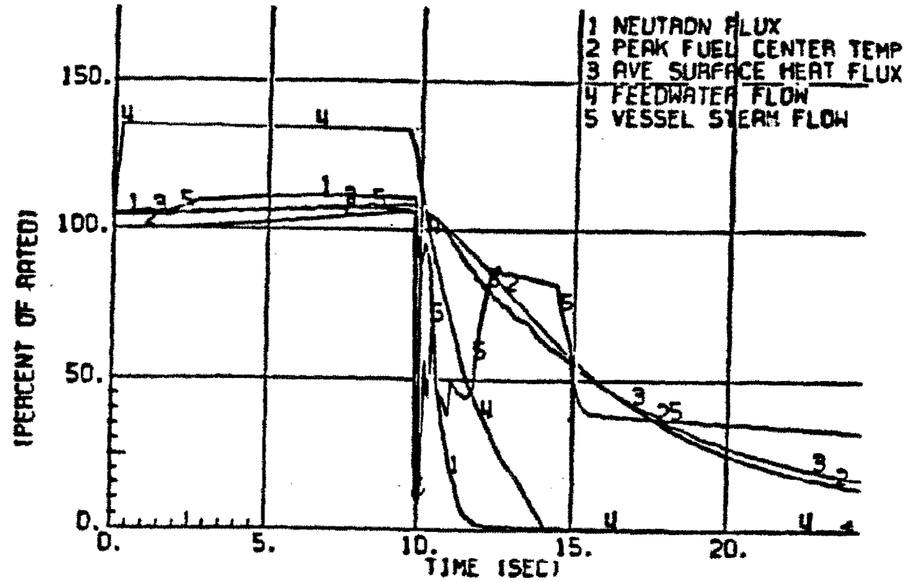
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	GENERATOR LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER, 73.8% FLOW EOC1 [HISTORICAL INFORMATION] FIGURE 15D.4-2 SHEET 1 OF 2

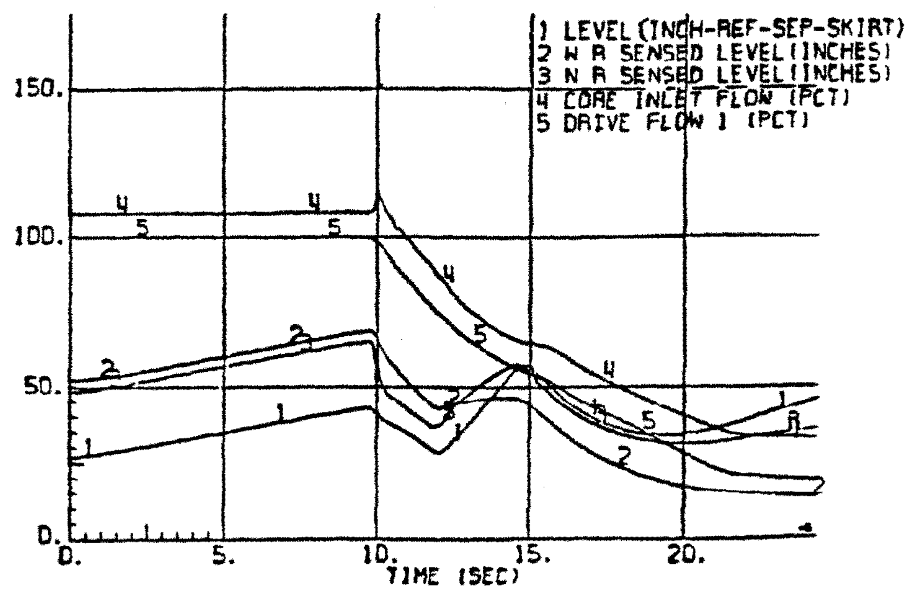




Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

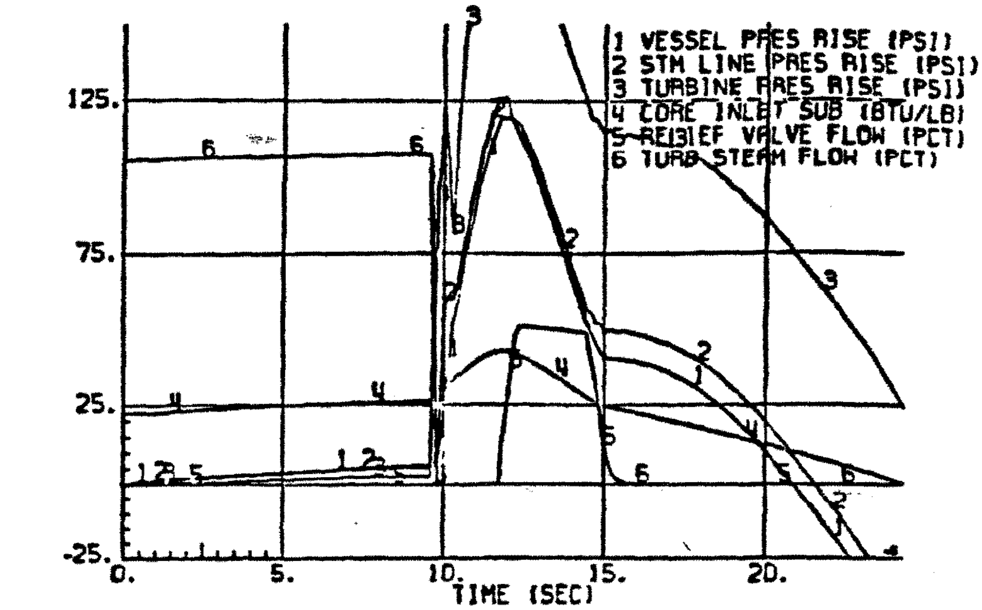
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	GENERATOR LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER, 73.8% FLOW EOC1 [HISTORICAL INFORMATION] FIGURE 15D.4-2 SHEET 2 OF 2

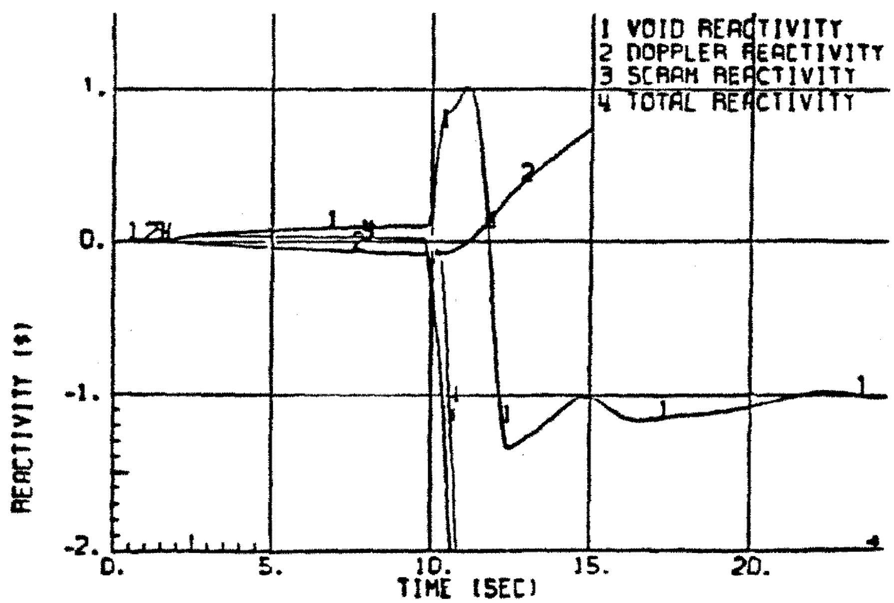




Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

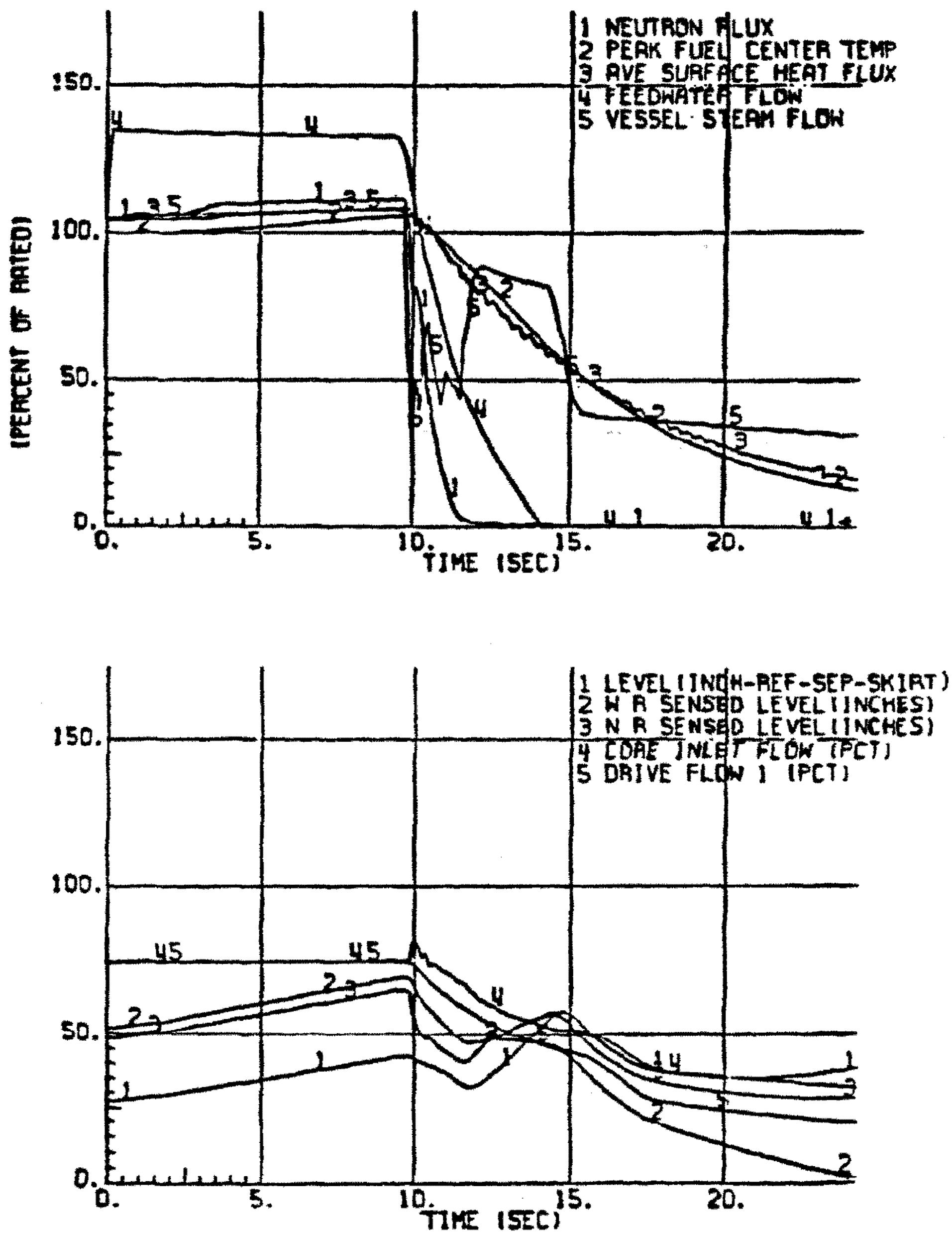
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE 104.2% POWER, 108% FLOW, EOC1 [HISTORICAL INFORMATION] FIGURE 15D.4-3 SHEET 1 OF 2





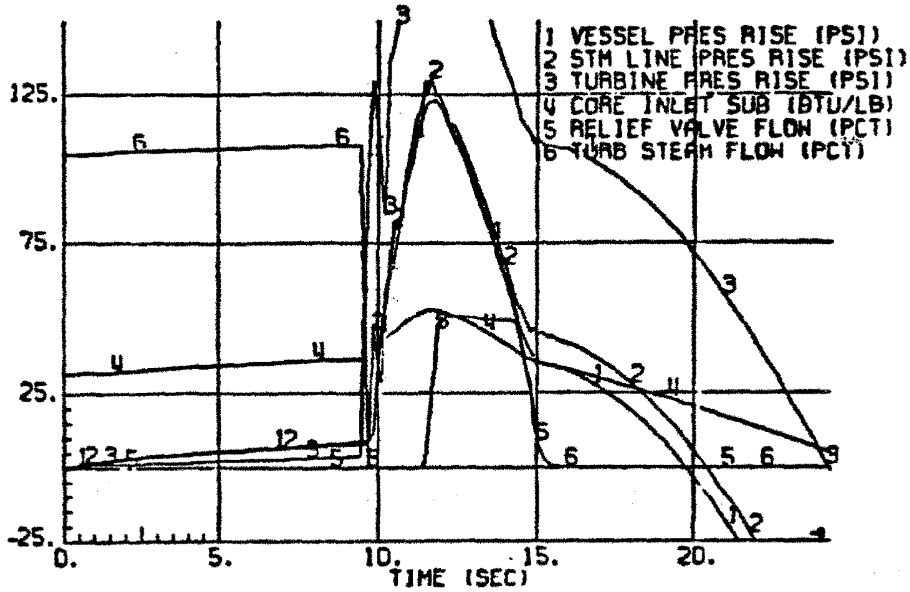
Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

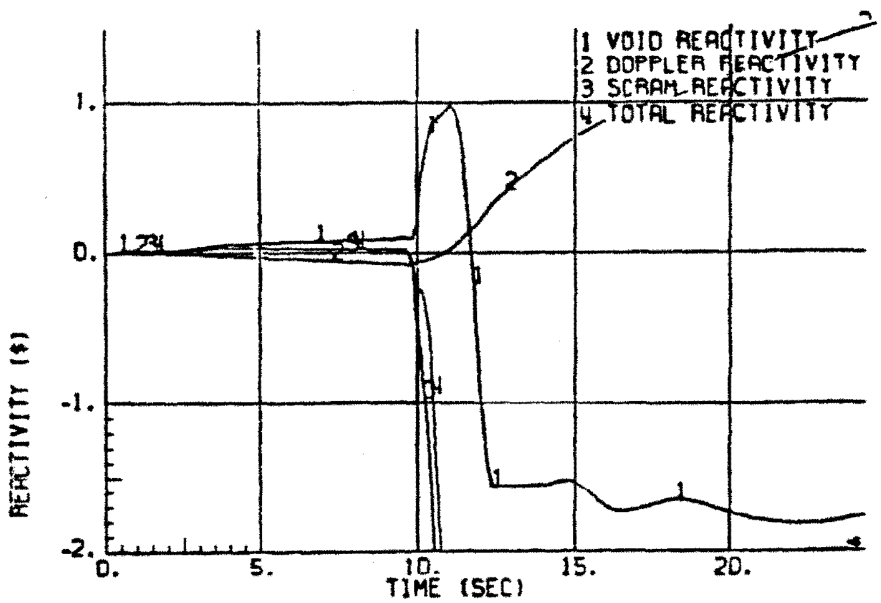
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE 104.2% POWER, 108% FLOW, EOC1 [HISTORICAL INFORMATION] FIGURE 15D.4-3 SHEET 2 OF 2



Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

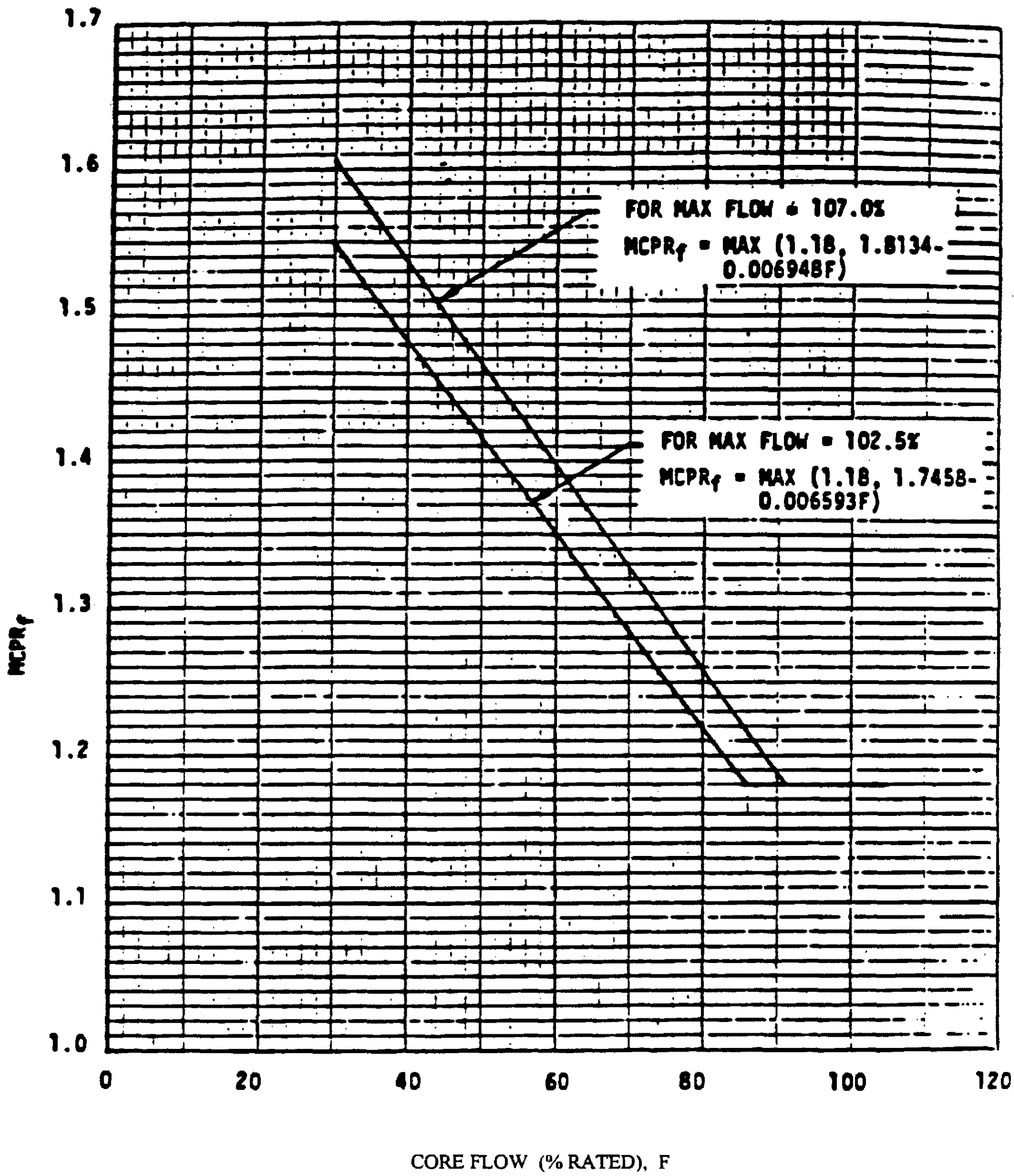
GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	**FEEDWATER CONTROLLER FAILURE** 104.2% POWER, 73.8% FLOW, EOC1 [HISTORICAL INFORMATION] FIGURE 15D.4-4 SHEET 1 OF 2





Note: Initial cycle analyses are based on originally licensed power level of 3833 MW.

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE 104.2% POWER, 73.8% FLOW, EOC1 [HISTORICAL INFORMATION] FIGURE 15D.4-4 SHEET 2 OF 2



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	FLOW DEPENDENT MCPR LIMIT FOR MEOD (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15D.4-5

15D.5 STABILITY ANALYSIS

The General Electric - Hitachi (GEH) Company has established stability criteria to demonstrate compliance to requirements set forth in 10CFR50 Appendix A, General Design Criterion (GDC) 12. These stability compliance criteria consider potential limit cycle response within the limits of safety system intervention and assure that for GEH BWR fuel designs this operating mode does not result in specified acceptable fuel design limits being exceeded. Furthermore, the onset of power oscillations for which corrective actions are necessary is reliably and readily detected and suppressed by operator actions and/or automatic system functions.

The fuel performance during limit cycle oscillations is characteristically dependent on the fuel design and certain fixed system features (high neutron flux scram setpoint, channel inlet orifice diameter, etc.). It is therefore possible to determine the acceptability of fuel designs independent of plant and cycle parameters. The stability compliance of those GEH BWR fuel designs contained in the General Electric Standard Application for Reactor Fuel (GESTARII, Ref. 6) is demonstrated on a generic basis in Reference 5. For reload cores, technical specification restrictions associated with the operating domain are established to ensure core thermal-hydraulic stability. In addition, thermal-hydraulic analyses are done to demonstrate that the stability performance of the reload core is equivalent to the stability performance for the previous cycle.

In response to NRC Generic Letter 94-02, GGNS implemented the BWR Owners' Group Enhanced Option 1-A (E1A) stability solution, and this was subsequently replaced with Option III Stability Solution. With the implementation of MELLLA+, GGNS replaced Option Ill with Detect and Suppress - Confirmation Density (DSS-CD) Stability Solution. The DSS-CD design continues to provide automatic detection and suppression of reactor instability and minimizes reliance on the operator to suppress instability events. The discussion provided below describes the analyses performed by the NSSS vendor for the initial fuel cycle and is retained for historical purposes only. For a detailed discussion of the current thermal-hydraulic stability solution refer to subsection 4.4.4.6.

[HISTORICAL INFORMATION] [For operation in the Maximum Extended Operating Domain (MEOD) the stability margin (defined by the core decay ratio) is reduced at higher powers for a given core flow. Therefore at the limiting condition for stability, natural circulation flow, operation at the maximum extended load line MELLLA will result in a higher decay ratio and therefore lower stability margin. However, the normal realistic operating region has the lowest stability margin at the maximum pump speed/minimum valve position flow (minimum forced circulation) which corresponds to 43% core flow for Grand Gulf (illustrated in Figure 15D.3-1). This increased flow relative to natural circulation results in a significant increase in stability margin for the maximum extended operating domain as demonstrated by tests at operating BWRs. Operation below minimum forced circulation flows can only occur during transients, e.g., two recirculation pump trip. Operation in this region is addressed in a set of GE operating recommendations (Ref. 7).

For demonstration of compliance with GDC 12, the generic stability analysis in Reference 5 is independent of stability margin since the reactor is assumed to be operating in a limit cycle condition (no stability margin). In addition, analyses are performed at various power/flow conditions to demonstrate that fuel design limits are not exceeded during limit cycle operation in any region of the power/flow map. These analyses have shown that the fuel performance is a function of the oscillation amplitude defined by the difference between the high neutron flux scram setpoint and the operating power level.Therefore, higher power levels (MEOD) result in smaller oscillations up to the scram setpoint and subsequently less limiting fuel performance.

The analyses of Reference 5 therefore support operation in the Maximum Extended Operating Domain for those fuel designs contained in Reference 6. As discussed above, in addition to these analyses, GE has issued a set of operating recommendations (Ref. 7) which inform the reactor operator how to recognize and suppress unanticipated oscillations when encountered during plant operation. Together, the analyses and operator recommendations support operation in the MEOD region and demonstrate compliance to GDC-12.]

15D.6 LOSS OF COOLANT ACCIDENT ANALYSES

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION [A bounding BWR/6 Loss of Coolant Accident (LOCA) analysis was performed in the Maximum Extended Operating Domain (MEOD) boundary defined in Figure 15D.3-1. The results were reviewed for GGNS. It is found that the initial cycle MAPLHGR limits presented in Chapter 6 are adequate to cover the entire maximum extended operating domain as defined in Figure 15D.3-1. The flow dependent MCPR operating limits used in the LOCA analysis bound the new flow dependent MCPR limits required for MEOD which is illustrated in Figure 15D.4-5. Therefore, the initial cycle LOCA analysis presented in Chapter 6 is applicable in the power/flow domain defined in this appendix with the initial cycle MAPLHGR limits. For additional information refer to Addendum 2 to Appendix 15D.

The effect of potential limit cycle oscillations on LOCA analyses was examined. A bounding BWR/6 evaluation was performed for a limit cycle oscillation at 90% power and 60% core flow. The maximum power oscillation at this power and core flow was found to be ~7% higher than the initial power. To bound this oscillation, LOCA calculations were performed starting at 100% power and 60% core flow. The results of this analysis showed no early boiling transition prior to jet pump uncovery. Consequently, these conditions produce less severe LOCA analysis results than are obtained at the 105% steam flow/100% core flow condition.]

15D.7 CONTAINMENT ANALYSIS

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [A containment analysis is performed in the MEOD region for Grand Gulf Stations. The peak drywell and wetwell pressures, peak suppression pool temperatures, chugging loads, condensation oscillation and pool swell containment responses were evaluated to bound the entire MEOD region with rated feedwater temperature reduced up to 100°F to provide a bounding analysis to justify feedwater heater(s) out of service operation at 370°F (described in Appendix 15B and Section 15D.13) in the MEOD.

The analysis shows peak drywell pressure for the worst MEOD combined with FWHOS operation condition of 38.0 psia. The corresponding peak drywell pressure is 23.3 psig which is 1.3 psi above the Chapter 6 value of 22.0 psig. However, this value is still below the design limit of 30 psig reported in Chapter 6. The limiting break is switched from main steam line break to the recirculation line break. The peak suppression pool temperatures, chugging loads, condensation oscillation and pool swell boundary loads are all found to be bounded by the rated power analysis in Chapter 6.]

15D.8 LOAD IMPACT ON INTERNALS

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

15D.8.1 Acoustic and Flow Induced Loads

[HISTORICAL INFORMATION] [The acoustic loads are lateral loads on the vessel internals that result from propagation of the decompression wave created by a sudden recirculation-suction-line break. The acoustic loading on the vessel internals is proportional to the total pressure wave amplitude in the vessel recirculation outlet nozzle. The total pressure amplitude is the sum of the initial pressure subcooling plus the experimentally determined pressure undershoot below saturation pressure. A larger downcomer subcooling results in a lower saturation pressure, thereby, having a larger total pressure amplitude and resulting in large acoustic loads. The maximum subcooling in the MEOD was found to be 48.68 BTU/LB.

The high-velocity flow patterns in the downcomer resulting from a recirculation-suction-line break create lateral loads on the shroud and the jet pump. These loads are proportional to the square of the critical mass flux rate out of the break. The additional subcooling in the downcomer resulting from operating in the MEOD leads to an increase in critical flow and, therefore, in flow induced loads.

The reactor internals most impacted by acoustic and flow induced loads are the shroud, shroud support and jet pump. The impact on these components were generically analyzed with a maximum subcooling of 83.6 BTU/LB associated with final feedwater temperature reduction operation. It is found that these components have enough design margin to handle these loadings.]

15D.8.2 Reactor Internal Pressure Difference Loads

[HISTORICAL INFORMATION] [A reactor internals pressure difference analysis is performed for the ICFR of MEOD. The increased reactor internal pressure differences across the reactor internals are generated for the maximum core flow at normal, upset, emergency and faulted conditions as input data for the fuel lift and the reactor internals impact evaluation to ensure the GGNS reactor internals can withstand the increased pressure differences.]

15D.8.3 Bundle Lift Evaluation

[HISTORICAL INFORMATION] [The margin to fuel bundle lift was reevaluated for the ICFR operation for normal, upset and faulted conditions. It was shown that there is enough net fuel lift margin during the worst case faulted event. A probabilistic fuel lift analysis is performed which concluded the fuel lift criteria are met for increased core flow operation.]

15D.8.4 Impact on Reactor Internals

[HISTORICAL INFORMATION] [The impact of increased core flow on the various reactor internal components are evaluated using the differential pressures discussed in Section 15D.8.2 and the forces generated by the probabilistic load combination analysis including fuel lift data discussed in Section 15D.8.3. The reactor internals most affected by pressure under increased core flow conditions are the core plate, guide tube, shroud head, upper shroud, shroud support ring and lower shroud, shroud top guide, fuel channel wall, steam dryer and jet pump. These components are evaluated under normal, upset, emergency and faulted conditions. It is concluded that the pressure differences for these and other components during increased core flow operation produce stresses that are within the allowable design limits.]

15D.9 FLOW INDUCED VIBRATIONS

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [To ensure that the flow-induced vibration response of the reactor internals is acceptable, a single reactor of each product line and size undergoes an extensive vibration test during initial plant startup. After analyzing the results of such tests and assuring that all responses fall within acceptable limits of the established criteria, the reactor is classified as a valid prototype in accordance with Regulatory Guide 1.20. All other reactors of the same product line and size undergo a less vigorous confirmatory test to assure similarity to the base test. Grand Gulf Station is fully instrumented as a prototype BWR/6 251 plant. The vessel internal vibration startup test has been completed for Grand Gulf Station in the ICFR. The results of the vibration test are documented in Reference 8.]

15D.10 IMPACT ON ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [An ATWS performance impact evaluation was performed for GGNS in the MEOD. The ATWS limiting Main Steam Isolation Valve (MSIV) closure event was analyzed in the MEOD region. Analyses were performed initiating from 100% power/75% flow which resulted in a higher power condition following an ATWS event. All peak pressures are below the emergency stress limits. Maximum neutron flux and heat flux, as well as vessel pressure, were also found to be acceptable. Therefore, it is concluded that MEOD operation is acceptable from ATWS requirements standpoint including ATWS stability considerations.]

15D.11 FUEL MECHANICAL PERFORMANCE

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [Evaluations were performed to determine the acceptability of GGNS MEOD operation on GE fuel rod and assembly thermal/mechanical performance. Component pressure differentials (described in Section 15D.8.4) and fuel rod overpower values were determined for anticipated operational occurrences initiated from MEOD conditions. These values were found to be bounded by those applied as the fuel rod and assembly design bases and therefore, GGNS MEOD operation is acceptable and consistent with fuel design bases.

An evaluation was also performed which concluded that fuel channel bypass flow, creep and control blade interference are not impacted by operation in the MEOD.]

15D.12 AVERAGE POWER RANGE MONITOR (APRM) SIMULATED THERMAL POWER (STP) SCRAM AND ROD BLOCK SETPOINT

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor. Refer to Sections 4.4.4.6, 7.1.2.1, and 7.2.1 for additional information regarding the stability trip function.

[HISTORICAL INFORMATION] [In order to allow operation in the Maximum Extended Operating Domain (MEOD), the current Average Power Range Monitor (APRM) Simulated Thermal Power Monitor (STPM) scram and rod block configuration and setpoints are modified to accommodate this region.

This consists of:

(1) Raising the current APRM rod block and STP scram line to higher power setpoints.

(2) Clipping the APRM rod block at high core flow.

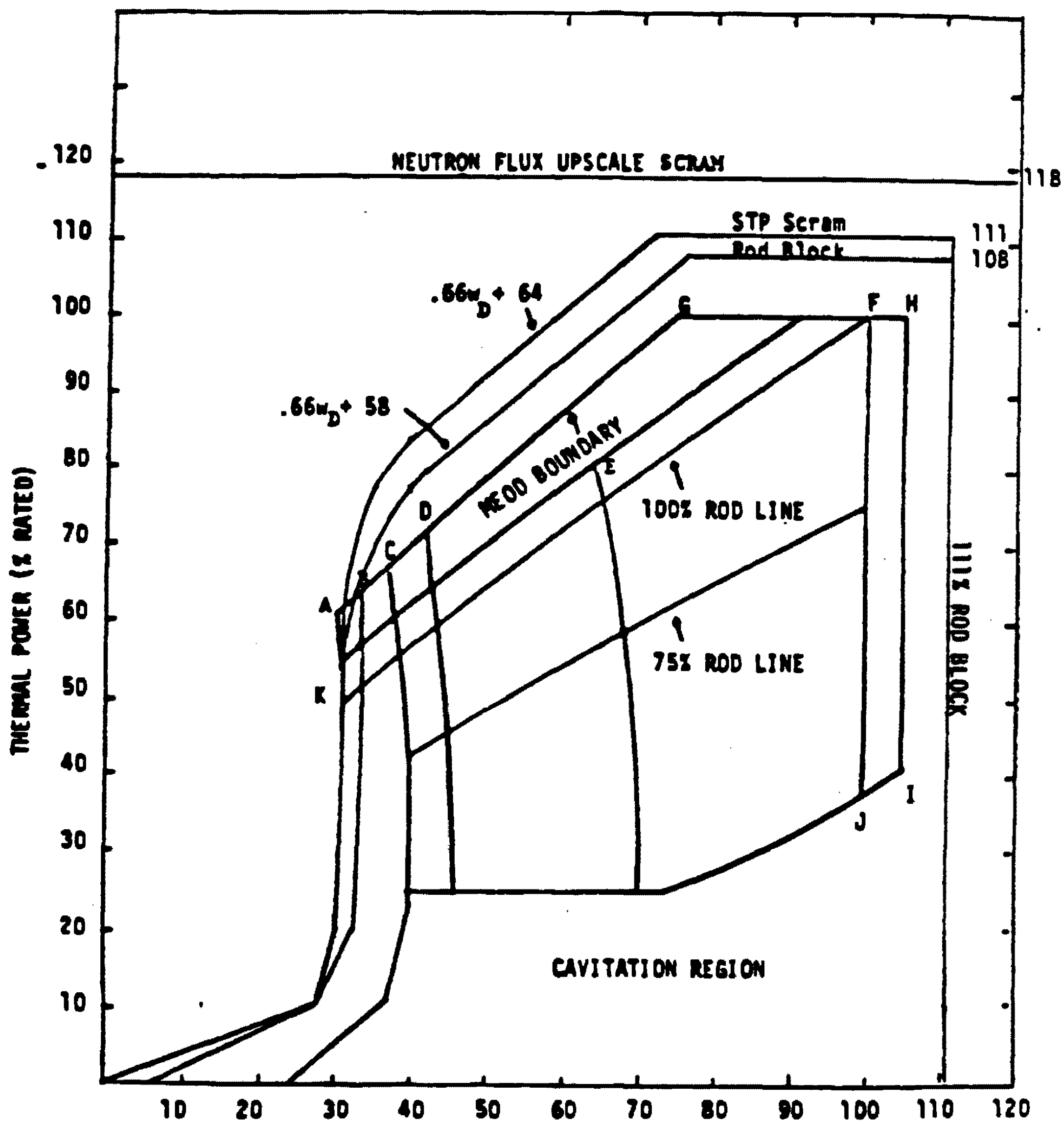
(3) Increasing the High Flow Rod Block setpoint.

The new APRM rod block and STPM scram setpoints for the MEOD are illustrated in Figure 15D.12-1 and tabulated in Table 15D.12-1.

The new setpoints presented in Table 15D.12-1 maintain the same slope (0.66), same clip setpoint (111% for STPM scram and 108% for rod block) at rated power condition and same margin (6%) between the STPM scram and rod block setpoints as the current 100% rod line Technical Specification. Therefore, no loss in rod block warning or scram protection exists due to this setpoint change when operating in the MEOD. These new setpoints are also applicable for single loop operation described in Appendix 15C. The increase of the high flow rod block setpoint from 108% to 111% core flow is one of operational concern. This is to ensure operation in the ICFR will not result in too many unnecessary rod block alarms.]

TABLE 15D.12-1: APRM INSTRUMENTATION SETPOINT FOR MEOD (INITIAL CYCLE) [HISTORICAL INFORMATION]

Functional Unit	Trip-Setpoint	Allowable Values	Analytical Limit
Flow Biased Simulated Thermal Power-High			
a) Flow biased	0.66W + 64% with a max.	0.66W + 67% with a max.	0.66W + 70% with a maximum of
b) High flow clamped	of 111% of rated thermal power	of 113% of rated thermal power	114% of rated thermal power
c) Neutron flux-high	118% of rated thermal power	120% of rated thermal power	122% of rated thermal power
Flow Biased Rod Block			
a) Flow biased	0.66W + 58% with a max.	0.66W + 61% with a max.	
b) High flow clamped	of 108% of rated thermal power	of 110% of rated thermal power	
Reactor Coolant System Recirculation Flow Rod Block			
a) Upscale	111% of rated flow	114% of rated flow	



CORE FLOW (% RATED)
REFER TO FIGURE 15D3-1 FOR DEFINITION OF POINTS A, B, C, D, E, F, G, H, I, J, K

GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	APRM SETPOINT CONFIGURATION FOR MEOD (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15D.12-1

15D.13 FEEDWATER HEATER(S) OUT OF SERVICE IN THE MAXIMUM EXTENDED OPERATING DOMAIN

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [This section presents results from a safety evaluation for the continued operation of GGNS with feedwater heater(s) out of service (FWHOS) during an annual initial cycle with GE-6 fuel in the maximum extended operating domain (MEOD) (Reference 9). This evaluation supports FWHOS operation in the MEOD as illustrated in Figure 15D.3-1. This section supplements the description in Appendix 15B. The evaluation is performed for the GE fueled GGNS on initial Cycle 1 basis and is applicable to 12 month cycle GE6 fueled initial cycle operation. The conditions of operation considered in the evaluation are those of unlimited continued 100% thermal power operation during the standard operation cycle with a maximum feedwater temperature reduction of up to 100°F in the MEOD. This evaluation is used to justify GGNS unlimited continued operation during an operation cycle in the MEOD between 420°F to 370°F feedwater temperature at rated power.

All the impact evaluations described in Appendix 15B were reexamined or reevaluated in the MEOD for FWHOS operation. Conclusions made in Appendix 15B are directly applicable in the MEOD. An operating MCPR limit of 1.18 is adequate for the range of 420°F to 370°F rated feedwater temperature FWHOS operation in the MEOD for GGNS at cycle 1.

It is concluded that with the power dependent MCPR limits described in Appendix 15B and those described in this appendix for MEOD, GGNS can operate with FWHOS down to 370°F in the MEOD during cycle 1.]

15D.13.1 Abnormal Operating Transient

[HISTORICAL INFORMATION] [Two of the limiting abnormal operating transients (Load Rejection with Bypass Failure and Feedwater Controller Failure) were reevaluated for FWHOS with a bounding BWR 6 238 size 748 bundle Standard plant at both 75% and 110% core flow and verified for GGNS at about 110% core flow at 105% steam flow condition. As described in Section 15D.4.1, the core average power density of the Standard BWR 6 plant is almost identical to the 251 size 800 bundle GGNS. The fuel type used in the standard plant analysis represents the bounding nature.

The GGNS specific verification analysis was performed to confirm the bounding analysis for equilibrium cycle is bounding for GGNS cycle 1. All other conditions described in Section 15B.2.1 are assumed in this analysis. The GGNS evaluation was performed at rated feedwater temperature of 370°F at end of cycle 1 and 2000 MWD/T exposure before end of cycle. Section 15B.2.1 describes the justification of these analyzed conditions. The 75% core flow case was not evaluated for GGNS because the bounding BWR 6 analysis performed at 370°F has shown that significant margin existed in this condition.

The GGNS plant specific transient peak value results are summarized in Tables 15D.13-1 and 15D.13-2. The critical power ratio (CPR) results are summarized in Tables 15D.13-3 and 15D.13-4. The transient responses are presented in Figures 15D.13-1, 15D.13-3, 15D.13-5, and 15D.13-7.

The GGNS specific analysis results shown in Tables 15D.13-3 and 15D.13-4 indicate that the ΔCPR at 370°F rated feedwater temperature does not exceeds the current operating limit. For GGNS cycle 1 application, the rated operating limit MCPR value of 1.18 is to be used for the range of rated feedwater temperature down to 370°F. Section 15D.14 describes the set of off-rated $MCPR_p$ limits to support the elimination of APRM trip setdown requirement including operation with FWHOS in the MEOD.]

15D.13.2 Other Evaluations

[HISTORICAL INFORMATION] [All other evaluations described in Appendix 15B are directly applicable to FWHOS in the MEOD. The 100°F Loss of Feedwater Heating is not affected by MEOD because the generic study applies to all power flow conditions. The rod withdrawal error analysis described in Section 15B.2.2 is directly applicable to the MEOD because the generic RWE analysis is performed based on the MEOD. As described in Section 15B.3, the compliance to stability General Design Criterion (GDC) 12 is demonstrated because the generic analyses of Reference 5 are independent of stability margin since the reactor is already assumed to be in limit cycle oscillation. This implicitly covers any variations in stability margins caused by FWHOS operation. Section 15D.5 also provided justification for compliance with GDC 12 in the MEOD region. It is therefore concluded that stability criteria are met for FWHOS in the MEOD because combination of FWHOS and MEOD does not result in any changes in parameters affecting the analysis of Reference 5 as the analyses already assumed limit cycle oscillation. Loss of Coolant Accident responses are shown to be bounded by Chapter 6 initial cycle analysis. Containment analysis and acoustic/flow induced load analysis described in Sections 15D.7 and 15D.8 are analyzed with reduced feedwater temperature associated with final feedwater temperature reduction (FFWTR) operation in the MEOD and are shown to be acceptable. Feedwater nozzle, sparger fatigue and system piping are all independent of power/flow condition. Impact on ATWS, annulus pressurization loads and fuel duty are all shown to be acceptable for FWHOS operation in the MEOD. GGNS plant specific analysis results are presented in Addendum 1 to Appendix 15D.]

TABLE 15D.13-1: SUMMARY OF GGNS TRANSIENT PEAK VALUES RESULTS - FWHOS IN MEOD - EOC1[a]
[HISTORICAL INFORMATION]

Transient	Core Flow (% NBR)	Peak Neutron Flux (% NBR)	Peak Dome Pressure (psig)	Peak Vessel Pressure (psig)	Peak Steamline Pressure (psig)	Fdwtr Temp. (°F)
Load Rejection With Bypass Failure	109.6[b]	162	1194	1224	1195	373
Feedwater Controller Failure, Max. Demand	109.6[b]	120.3	1150	1175	1149	373

(a) Initial power is 104.2% NBR for analysis

(b) Maximum achievable core flow for the given feedwater temperature

TABLE 15D.13-2: SUMMARY OF TRANSIENT PEAK VALUES RESULTS - FWHOS IN MEOD 2000 MWD/T BEFORE EOC1[a] [HISTORICAL INFORMATION]

Transient	Core Flow (% NBR)	Peak Neutron Flux (% NBR)[a]	Peak Dome Pressure (psig)	Peak Vessel Pressure (psig)	Peak Steamline Pressure (psig)	Fdwtr Temp. (°F)
Load Rejection With Bypass Failure	109.6[b]	104.3	1185	1216	1189	373
Feedwater Controller Failure, Max. Demand	109.6[b]	118.2	1136	1160	1135	373

(a) Initial power is 104.2% NBR for analysis

(b) Maximum achievable core flow for the given feedwater temperature

TABLE 15D.13-3: SUMMARY OF CPR RESULTS - FWHOS IN MEOD - EOC1
[HISTORICAL INFORMATION]

Transient	Core Flow (% NBR)	ICPR[b]	ΔCPR	MCPR	Fdwtr Temp. (°F)
Load Rejection With Bypass Failure	109.6[a]	1.18	0.05	1.13	373
Feedwater Controller Failure, Max. Demand	109.6[a]	1.18	0.11	1.07	373

(a) Maximum achievable core flow for the given feedwater temperature.

(b) Based on initial core safety limit of 1.06, for reload cores 0.01 must be added.

NOTE: Option A adders included.

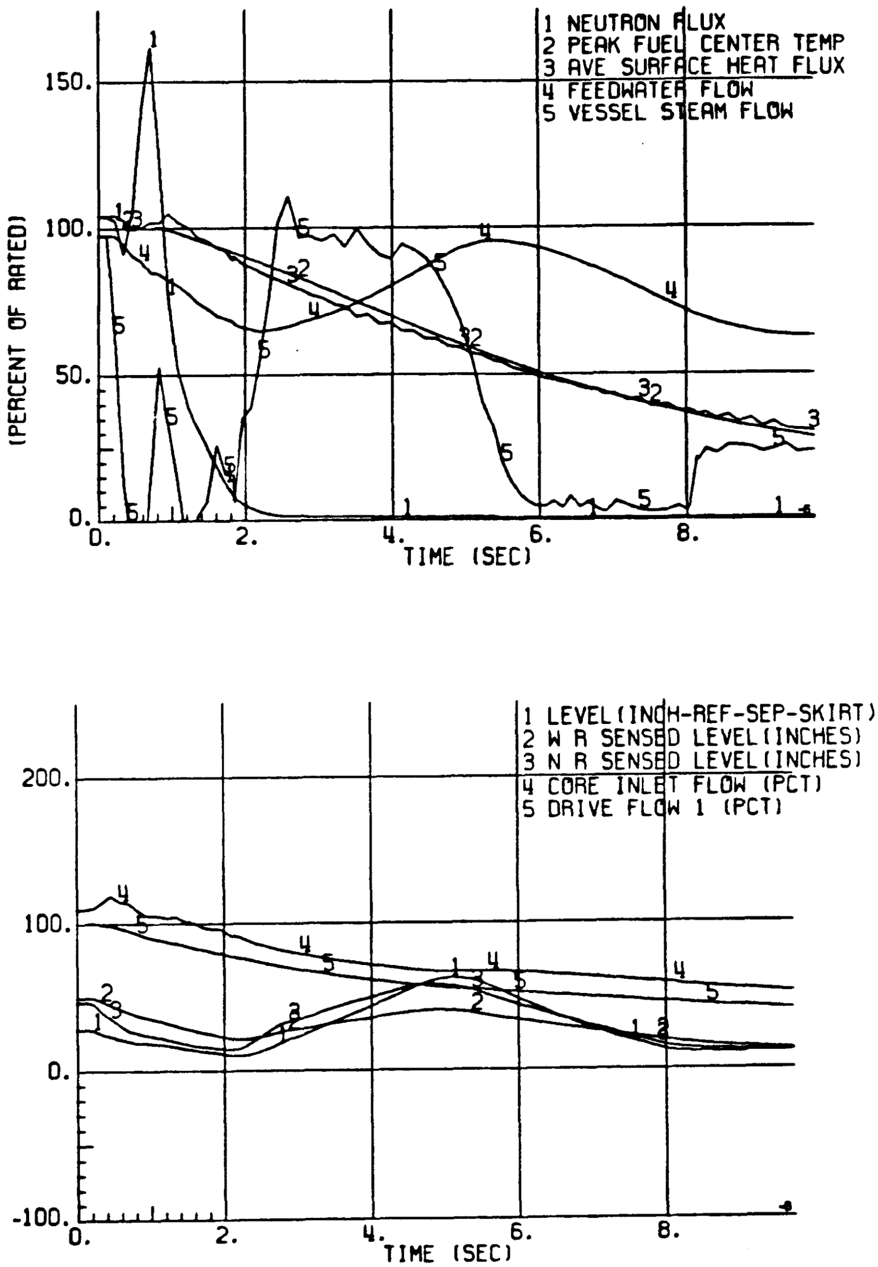
TABLE 15D.13-4: SUMMARY OF CPR RESULTS - FWHOS IN MEOD - 2000 MWD/T BEFORE EOC1
[HISTORICAL INFORMATION]

Transient	Core Flow (% NBR)	ICPR[b]	ΔCPR	MCPR	Fdwtr Temp. (°F)
Load Rejection With Bypass Failure	109.6[a]	1.18	0.05	1.13	373
Feedwater Controller Failure, Max. Demand	109.6[a]	1.18	0.11	1.07	373

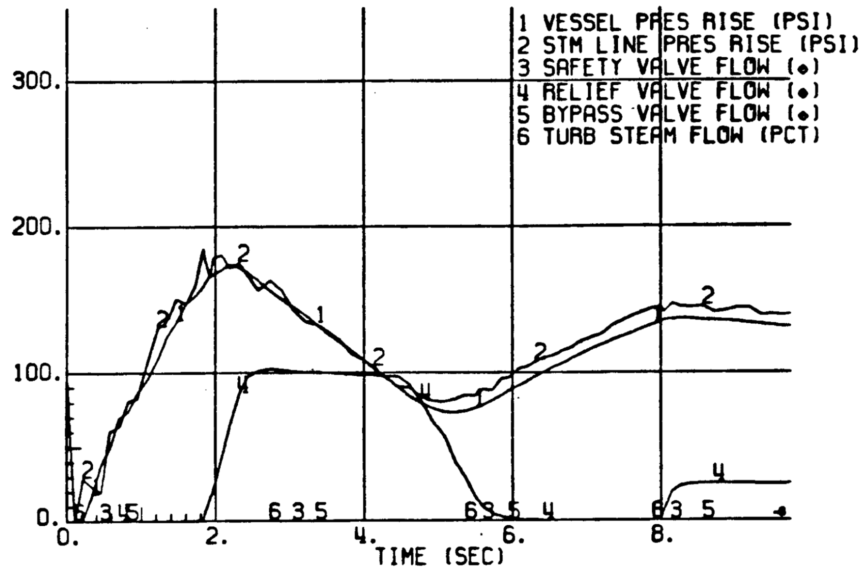
(a) Maximum achievable core flow for the given feedwater temperature.

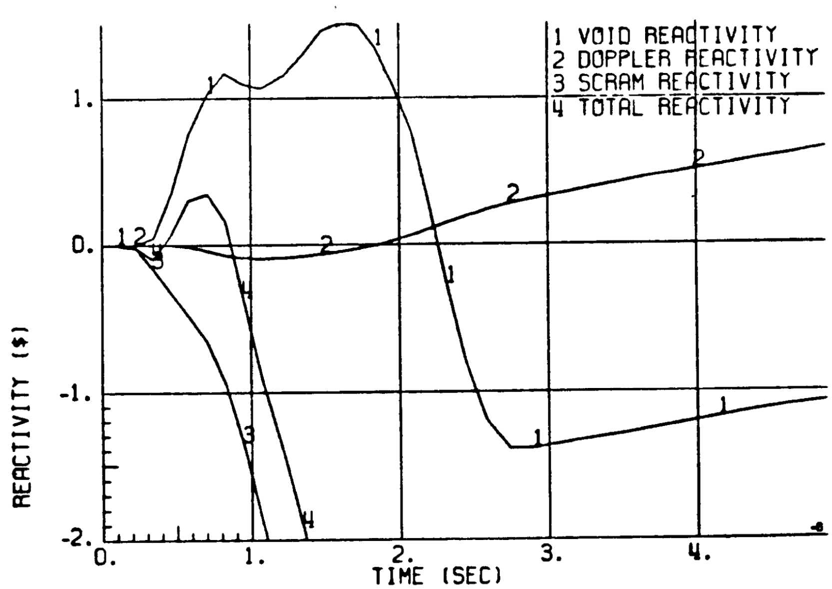
(b) Based on initial core safety limit of 1.06, for reload cores 0.01 must be added.

NOTE: Option A adders included.



SYSTEM ENERGY RESOURCES, INC. GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER, 109.6% FLOW, 373 °F TFW EOC1 [HISTORICAL INFORMATION] FIGURE 15D.13-1 SHEET 1 OF 2





SYSTEM ENERGY RESOURCES, INC. GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER, 109.6% FLOW, 373 °F TFW EOC1 [HISTORICAL INFORMATION] FIGURE 15D.13-1 SHEET 2 OF 2

Figure 15D.13-2
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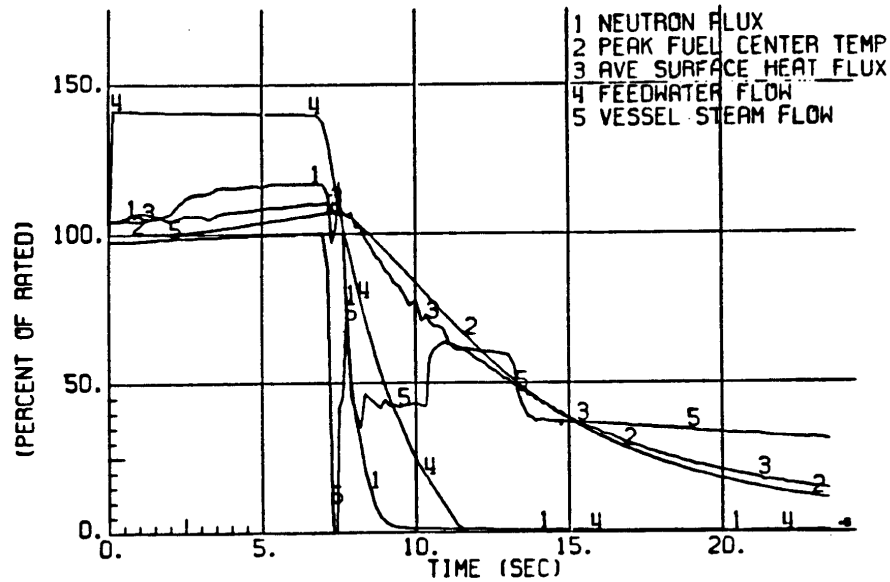
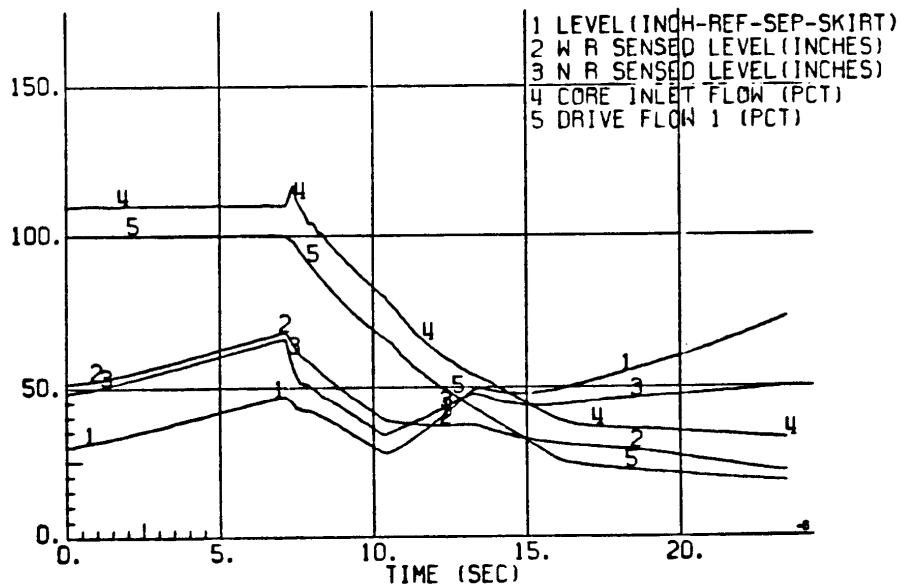
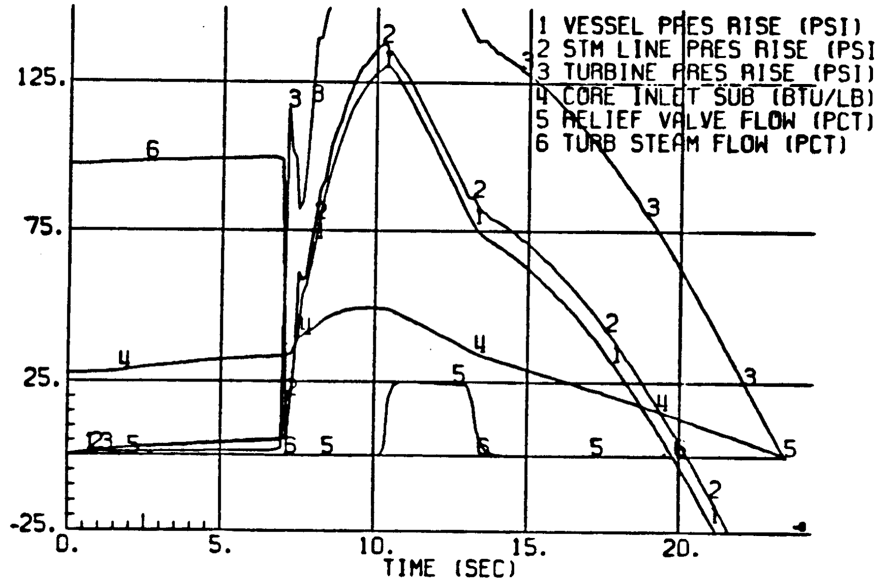
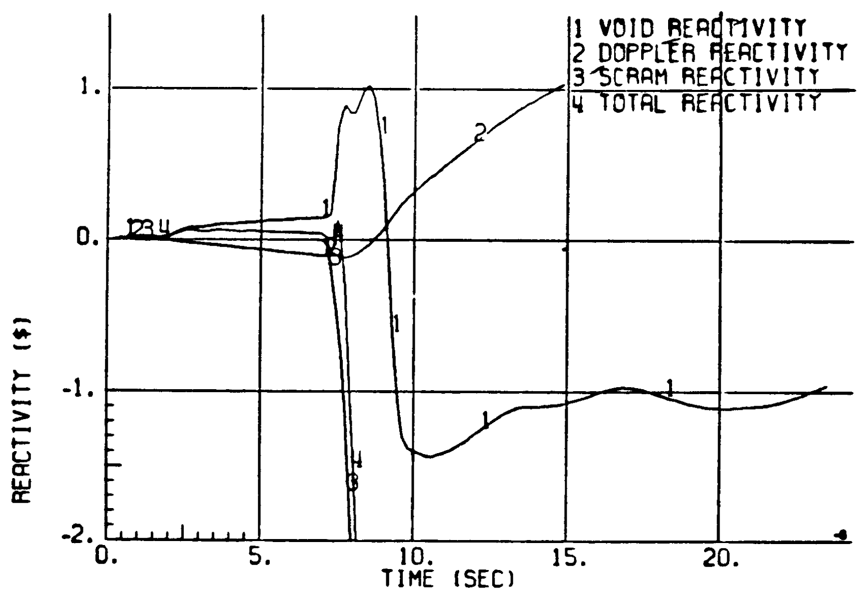




SYSTEM ENERGY RESOURCES, INC. GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE 104.2% POWER, 109.6% FLOW, 373 °F TFW EOC1 [HISTORICAL INFORMATION] FIGURE 15D.13-3 SHEET 1 OF 2





SYSTEM ENERGY RESOURCES, INC. GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE 104.2% POWER, 109.6% FLOW, 373 °F TFW EOC1 [HISTORICAL INFORMATION] FIGURE 15D.13-3 SHEET 2 OF 2

Figure 15D.13-4
(Sheets 1 of 2 thru 2 of 2)

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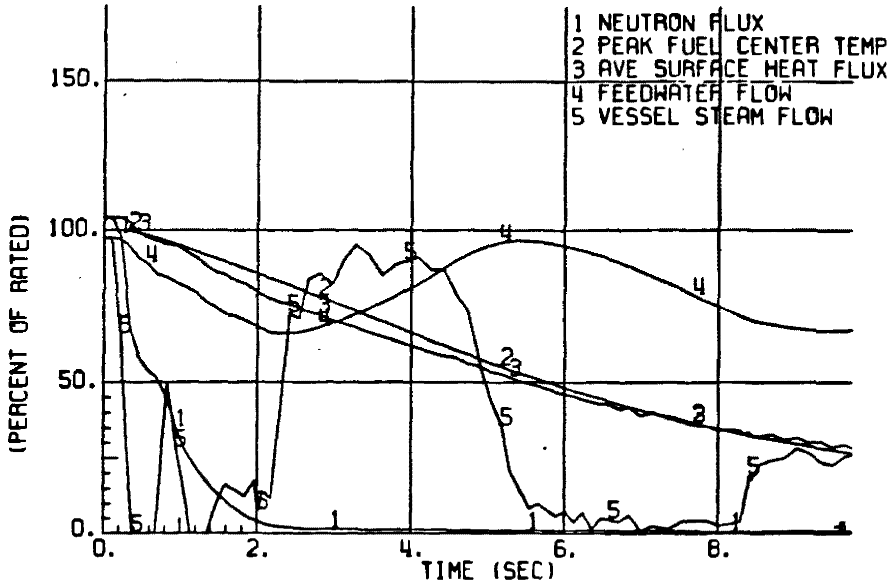
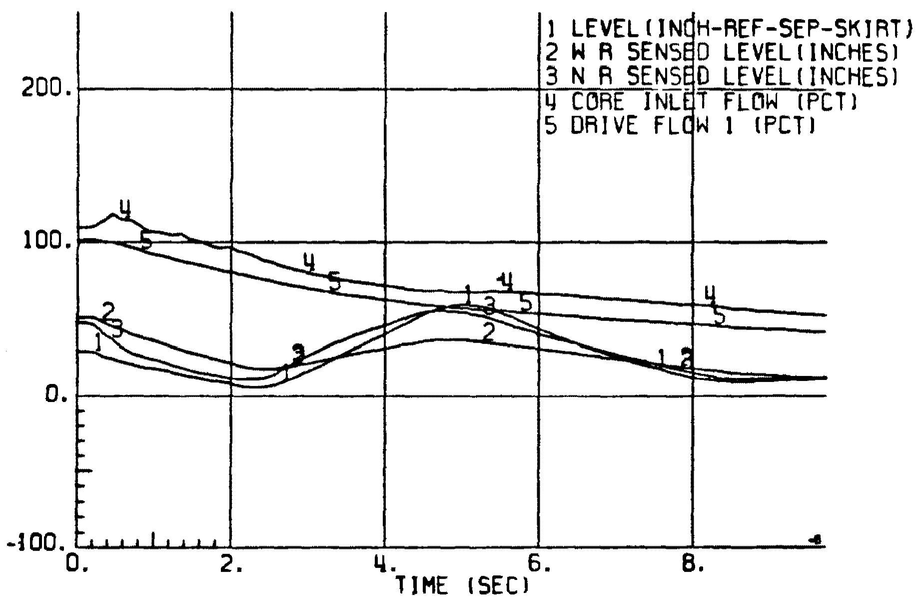
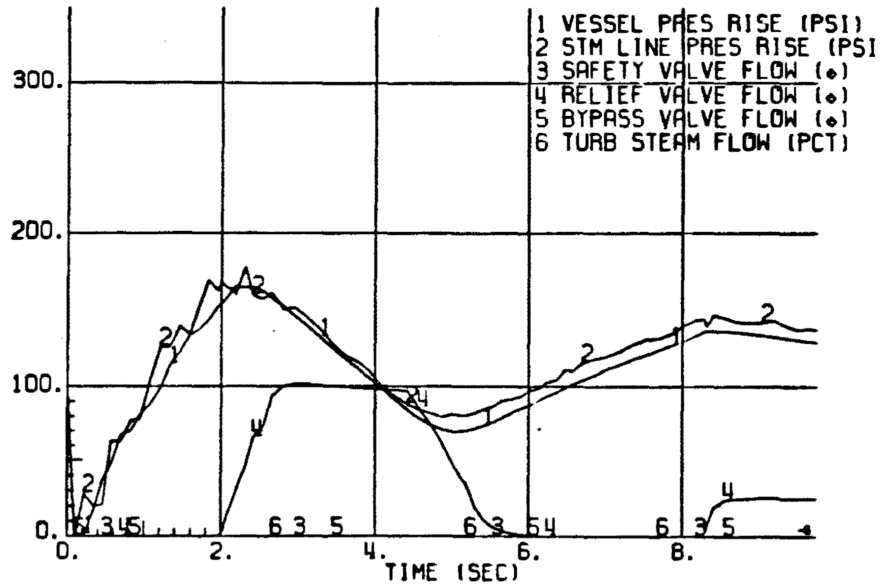
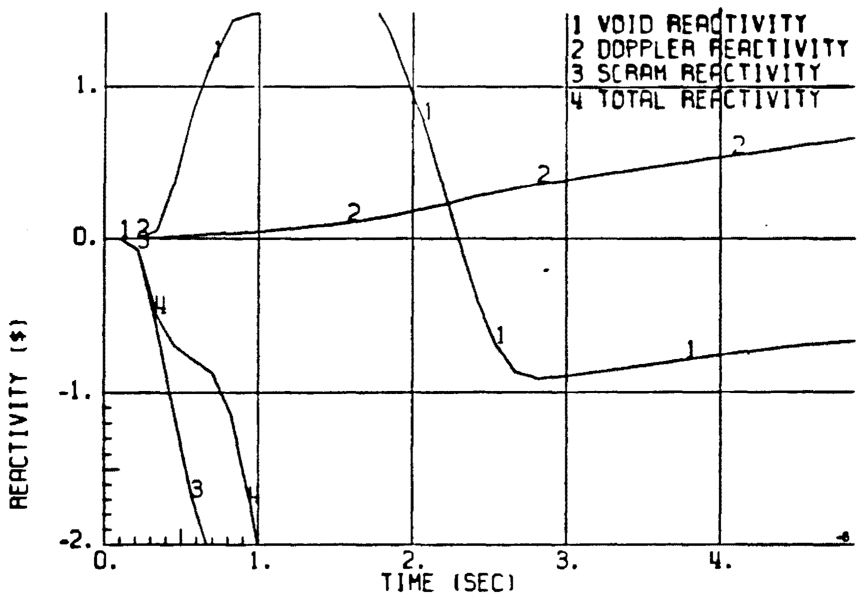




[HISTORICAL INFORMATION]

SYSTEM ENERGY RESOURCES, INC. GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	GENERATOR LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER, 109.6% FLOW, 373 °F TFW EOC1-2K MWD/T FIGURE 15D.13-5 SHEET 1 OF 2





[HISTORICAL INFORMATION]

SYSTEM ENERGY RESOURCES, INC. GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	GENERATOR LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER, 109.6% FLOW, 373 °F TFW EOC1-2K MWD/T FIGURE 15D.13-5 SHEET 2 OF 2

Figure 15D.13-6
(Sheets 1 of 2 thru 2 of 2)

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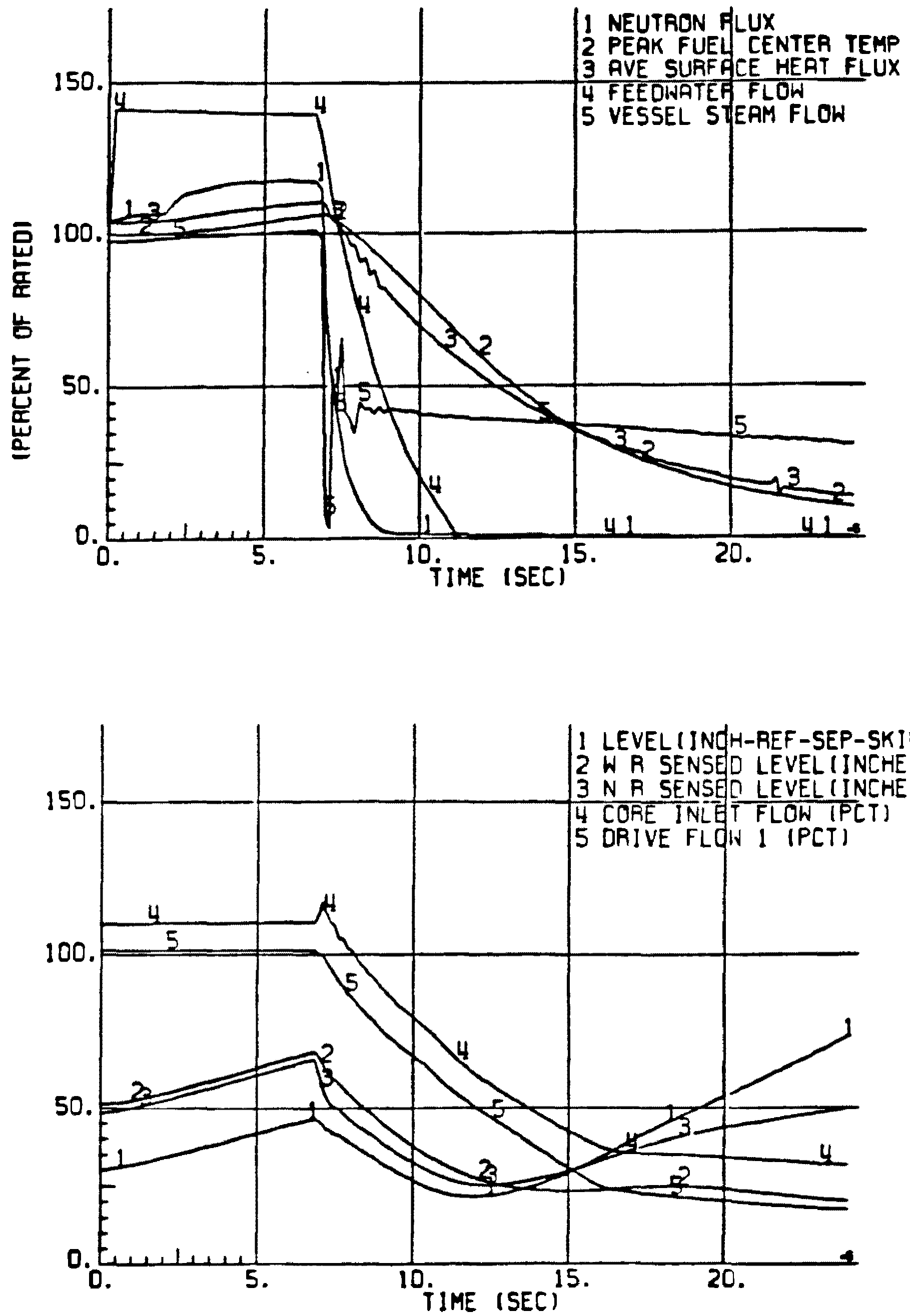
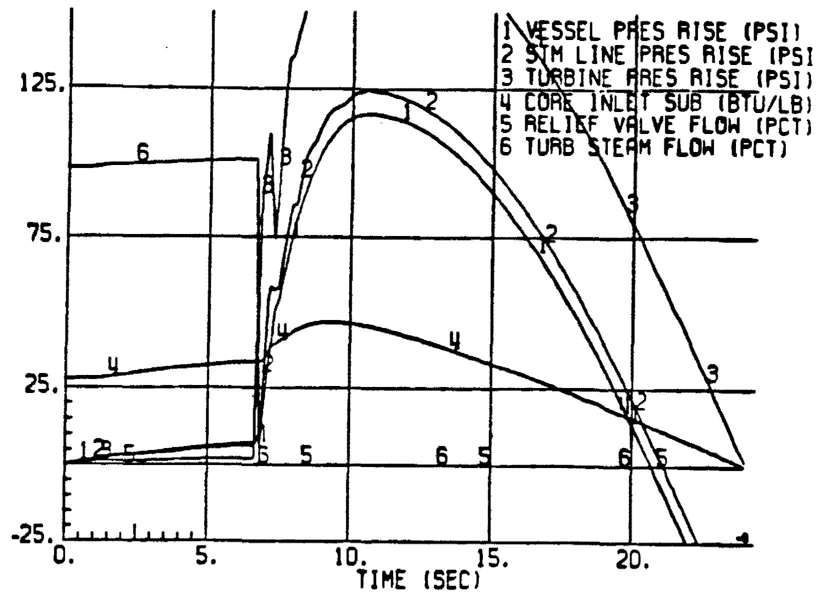
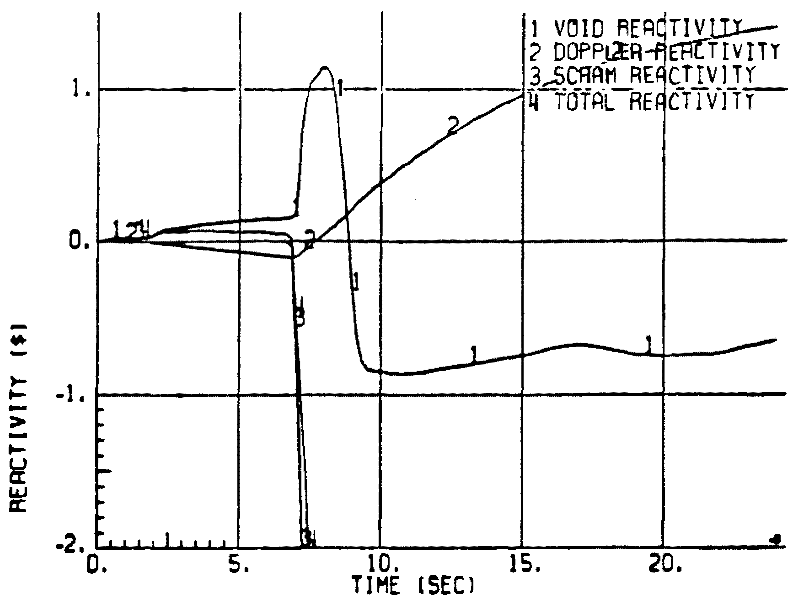


[HISTORICAL INFORMATION]

SYSTEM ENERGY RESOURCES, INC. GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE 104.2% POWER, 109.6% FLOW, 373 °F TFW EOC1-2K MWD/T FIGURE 15D.13-7 SHEET 1 OF 2





[HISTORICAL INFORMATION]

SYSTEM ENERGY RESOURCES, INC. GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	FEEDWATER CONTROLLER FAILURE 104.2% POWER, 109.6% FLOW, 373 °F TFW EOC1-2K MWD/T FIGURE 15D.13-7 SHEET 2 OF 2

Figure 15D.13-8
(Sheets 1 of 2 thru 2 of 2)

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15D.14 ELIMINATION OF THE APRM TRIP SETDOWN REQUIREMENT

This section addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. The references to "current cycle" in this section refer to the initial cycle. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [The GGNS Technical Specifications require that the flow biased APRM trips be lowered (setdown) when the core maximum total peaking factor exceeds the design total peaking factor. The "APRM setdown" requirement originated from the obsolete Hench-Levy Minimum Critical Heat Flux Ratio (MCHFR) thermal limit criterion.

The change to GETAB/GEXL and the move to secondary reliance on flux scram for licensing transient evaluations (for transients terminated by anticipatory or direct scram) has provided more effective and operationally acceptable alternatives to the setdown requirement. The GGNS MEOD evaluation uses transient analyses to define thermal limits initial conditions (operating limits) which conservatively assure that all licensing criteria are satisfied without setdown of the APRM scram and flow biased rod block trips.

The objective of this evaluation is to justify removal of the peaking factor setdown requirement. Those licensing areas which might be affected by the elimination of the setdown requirement are:

a. fuel thermal-mechanical integrity, and

b. loss-of-coolant accident.

The following criteria assure satisfaction of the applicable licensing requirements and were applied to demonstrate the acceptability of elimination of the setdown requirement.

a. MCPR safety limit shall not be violated as a result of any abnormal operating transient,

b. All fuel thermal mechanical performance shall remain within the design and licensing bases, and

c. peak cladding temperature and maximum cladding oxidation fraction following a LOCA shall remain within the limits defined by the applicable regulations.

The safety evaluations therefore include abnormal operational transients and LOCA analysis.]

15D.14.1 Transient Evaluation

[HISTORICAL INFORMATION] [A large data base was used to study the trend of transient severity without the average power range monitor (APRM) core peaking factor setdown. This data base was established by analyzing limiting transients over a range of power and flow conditions and was used to develop plant operating limits (MCPR and MAPLHGR) which will assure that margins to fuel integrity limits are equal to or larger than those in existence at the present time.

Results from the above transient analyses were used to establish the MAPLHGR versus power and flow and to verify or establish the MCPR versus power and flow limits. A variety of GGNS specific Feedwater Controller Failure and Load Rejection With Bypass Failure Transients with and without Feedwater Heater(s) Out of Service (FWHOS) and Final Feedwater Temperature Reduction (FFWTR) together with a bounding BWR/6 analysis and a large data base of operating plants results were used to assure that suitable conservatism exists for operation in the MEOD with FFWTR and FWHOS mode of operations. Results of the bounding BWR/6 analyses are tabulated in Table 15D.14-1. Results of the GGNS specific analyses are tabulated in Table 15D.14-2.]

15D.14.2 Loss of Coolant Accident

[HISTORICAL INFORMATION] [The impact of the elimination of the APRM setdown requirement on LOCA is examined in the maximum extended operating domain. It is found that the current MAPLHGR limits are adequate to protect against a Loss of Coolant Accident even without APRM setdown to assure that peak cladding temperature remain below the initial cycle LOCA results in Chapter 6. This is because the initial cycle LOCA analysis documented in Chapter 6 is performed without taking credit for the APRM setdown.]

15D.14.3 Plant Operating Limit

[HISTORICAL INFORMATION] [Power and flow dependent limits on local peak power and MCPR are imposed such that fuel design and safety criteria are satisfied without the peaking factor setdown.]

15D.14.3.1 Flow Dependent MCPR Limit

[HISTORICAL INFORMATION] [The current flow dependent MCPR limits remain unchanged because the design basis flow runout event is a slow flow/power increase event which is not terminated by scram.]

15D.14.3.2 Power Dependent MCPR Limit

[HISTORICAL INFORMATION] [The current power dependent MCPR limits are modified based on results of GGNS specific analysis and prior bounding BWR/6 analysis as well as a large operating plant data base to include:

a. A new set of limits including a new limit format for core power below 40% (power level where reactor scram on turbine control valve fast closure are bypassed) which consists of both high and low core flow dependent power dependent MCPR Limit due to the significant sensitivity to initial core flow below this bypass power level. This set of limits also apply to Feedwater Heater(s) Out of Service (FWHOS) and Final Feedwater Temperature Reduction (FFWTR) operation.

b. A new MCPR multiplier limit (K_p) is established for core power above 40% to replace the absolute power dependent $MCPR_p$ limit to eliminate the complication of having several $MCPR_p$ curves for different modes of operation. A generic K_p curve is to be applied to different rated operating limits above 40% power.

These new power dependent MCPR limits and multiplier (K_p) limits are shown in Figure 15D.14-1.]

15D.14.3.3 Flow Dependent MAPLHGR Limit

[HISTORICAL INFORMATION] [The flow dependent MAPLHGR limits were determined using the three-dimensional BWR simulator (Reference 3) to analyze the slow flow runout transients. These factors are derived such that the peak transient MAPLHGR during these events is not increased above the fuel design basis values. The flow dependent MAPLHGR factor ($MAPFAC_F$) limit is shown in Figure 15D.14-2. The actual flow dependent MAPLHGR limits are equal to this $MAPFAC_F$ multiplied by the rated MAPLHGR limit.]

15D.14.3.4 Power Dependent MAPLHGR Limit

[HISTORICAL INFORMATION] [In the absence of the APRM scram setdown requirement, special limits are substituted to assure adherence to the fuel thermal mechanical design bases. Power dependent limits are generated using the same data base as the power dependent MCPR Limits.

As previously discussed under $MCPR_p$ (K_p) limit, a significant sensitivity to initial core flow exists below 40% core power. Therefore, a set of both high and low core flow limits is provided. These limits are derived to assure that the peak transient MAPLHGR is not increased above the fuel design basis transient values. Appropriate MAPLHGR(p) limits are selected based on GGNS specific and bounding BWR/6 transient analyses and trends observed in the operating plant data base. The new power dependent MAPLHGR factor ($MAPFAC_p$) limit is presented in Figure 15D.14-3. The actual power dependent MAPLHGR limits are equal to this $MAPFAC_p$ multiplied by the rated MAPLHGR limit.

For single loop operation (SLO), the most restrictive of the SLO MAPLHGR factor and this MAPLHGR factor will define the limiting condition of operation.]

15D.14.3.5 Governing Overall Limit

[HISTORICAL INFORMATION] [At any given power/flow state, all four limits must be determined. The most limiting MCPR and the most limiting MAPLHGR (maximum of $MCPR_p$ (K_p multiplied by rated MCPR limit) and $MCPR_f$ and minimum of $MAPLGHR_p$ and $MAPLHGR_f$) will be governing limit. The rated operating limit MCPR value for the different modes of operation are presented in Table 15D.14-3.]

TABLE 15D.14-1: BOUNDING BWR/6 TRANSIENT ANALYSIS RESULTS FOR ELIMINATION OF APRM TRIP SETDOWN (INITIAL CYCLE) [HISTORICAL INFORMATION]

Case #	Transient[a]	Power (%)	Flow (%)	Scram[b]	ΔCPR[c]	$MCPR_p$[d]
1	FWCF - W/RPT[e]	100	111.7	L8	.15	1.21
2	FWCF - W/RPT[e]	53.5	116	L8	.39	1.45
3a	FWCF - W/RPT[e]	40	92	L8	.44	1.50
3b	FWCF - W/RPT[e]	40	92	L8	.47	1.53
3c	LRNBP - N/RPT	40	92	PR	.99	2.05
4a	FWCF - N/RPT[e]	25	64	L8	.19	1.25
4b	LRNBP - N/RPT	25	64	PR	1.04	2.10
5a	FWCF - N/RPT[e]	40	50	L8	.13	1.19
5b	LRNBP - N/RPT	40	50	PR	.71	1.77
6a	FWCF - N/RPT[e]	25	50	L8	.16	1.22
6b	LRNBP - N/RPT	25	50	PR	.92	1.98

FOOTNOTES:

a W/RPT = With Recirculation Pump Trip
N/RPT = No Recirculation Pump Trip

b L8 = High Water Level 8 Scram
PR = Pressure Scram

c ODYN Option A adder included

d based on SLMCPR = 1.06 $MCPR_p$ = SLMCPR + ΔCPR

e with feedwater temperature reduction of 170°F, i.e., TFW = 250°F

TABLE 15D.14-2: GRAND GULF TRANSIENT ANALYSIS RESULTS FOR ELIMINATION OF APRM TRIP SETDOWN (INITIAL CYCLE) [HISTORICAL INFORMATION]

Case #	Transient[a]	Power (%)	Flow (%)	Scram[b]	ΔCPR[c]	$MCPR_p$[d]
1	FWCF - W/RPT[e]	100	108.6	L8	.13	1.19
2	FWCF - W/RPT[e]	85	110	L8	.17	1.23
3	FWCF - W/RPT[e]	70	112	L8	.23	1.29
4	FWCF - W/RPT[e]	46	114	L8	.38	1.44
5a	FWCF - W/RPT[e]	40	105	L8	.39	1.45
5b	FWCF - W/RPT[e]	40	105	L8	.40	1.46
5c	LRNBP - N/RPT	40	105	PR	.89	1.95
6	LRNBP - N/RPT	25	72	PR	1.01	2.07
7	LRNBP - N/RPT	40	50	PR	.62	1.68
8	LRNBP - N/RPT	25	50	PR	.83	1.89

FOOTNOTES:

a W/RPT = With Recirculation Pump Trip
N/RPT = No Recirculation Pump Trip

b L8 = High Water Level 8 Scram
PR = Pressure Scram

c ODYN Option A adder included

d based on SLMCPR = 1.06 $MCPR_p$ = SLMCPR + ΔCPR

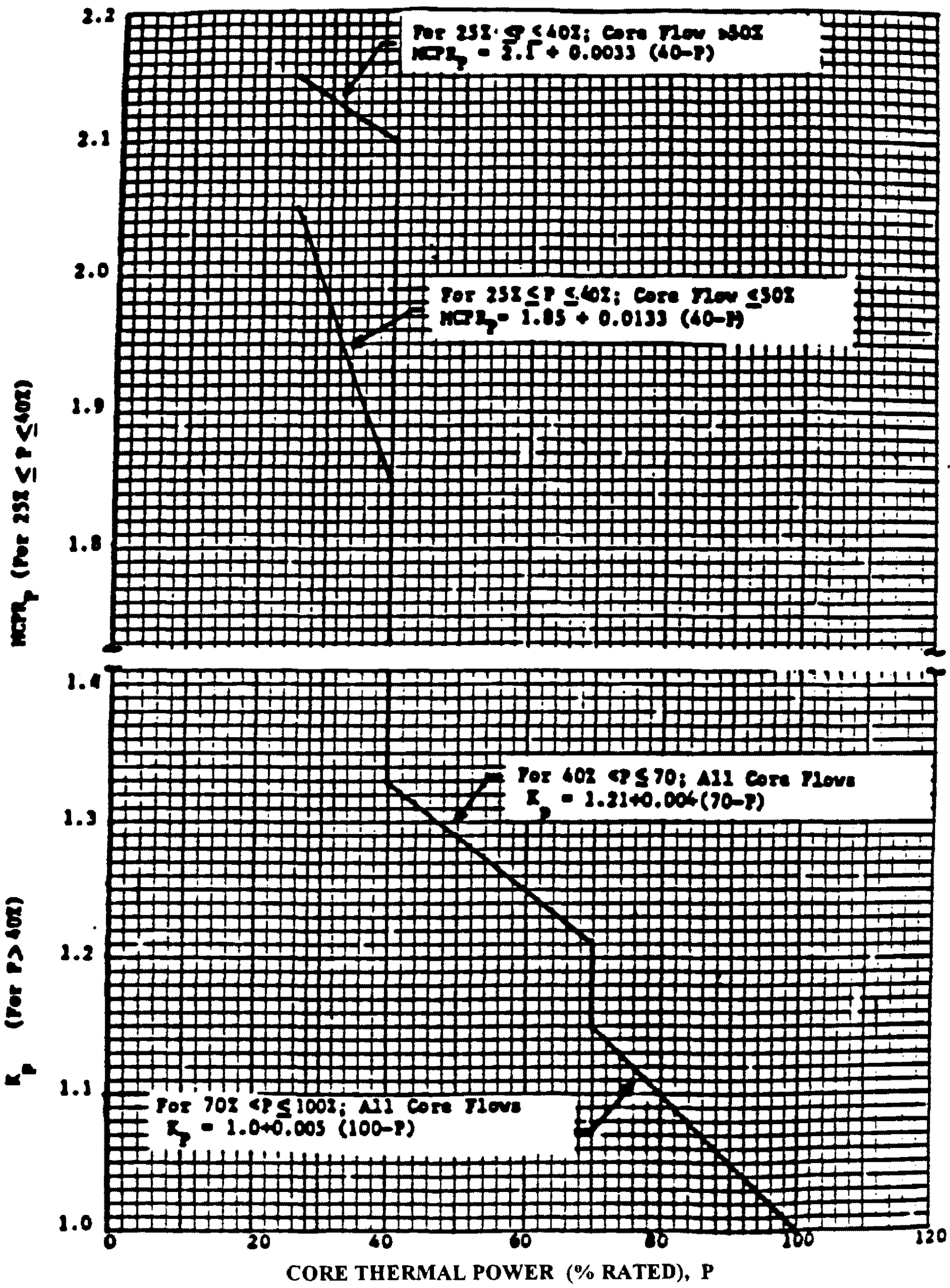
e with feedwater temperature reduction of 170°F, i.e., TFW = 250°F

TABLE 15D.14-3: RATED OPERATING LIMIT MCPR VALUES (INITIAL CYCLE) [HISTORICAL INFORMATION]

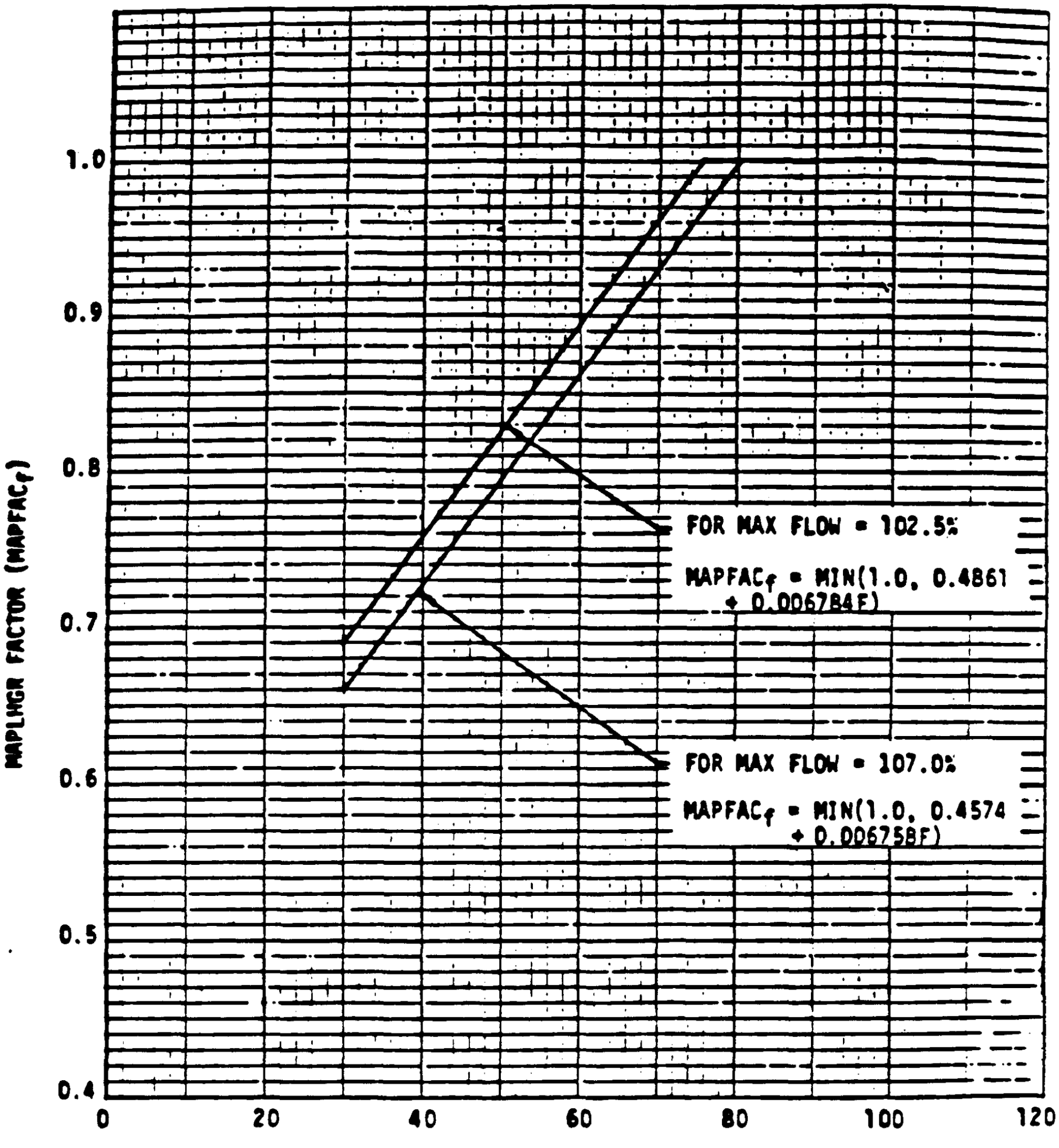
Mode of Rated* Operation	OLMCPR
Current FSAR P/F Map (Fig. 15D.3-2)	1.18
FWHOS (Appendix 15B)	1.18 (420°F to 370°F, up to 100% flow)
SLO (Appendix 15C)	1.18
MEOD (Appendix 15D)	1.18
FWHOS in MEOD (Appendix 15D)	1.18(420°F to 370°F, up to 105% flow)

* These values are for cycle 1 only. These values are to be applied with the K_p curve for off-rated conditions above 40% power.
$MCPR_p(p) = K_p(p) * OLMCPR$ rated.

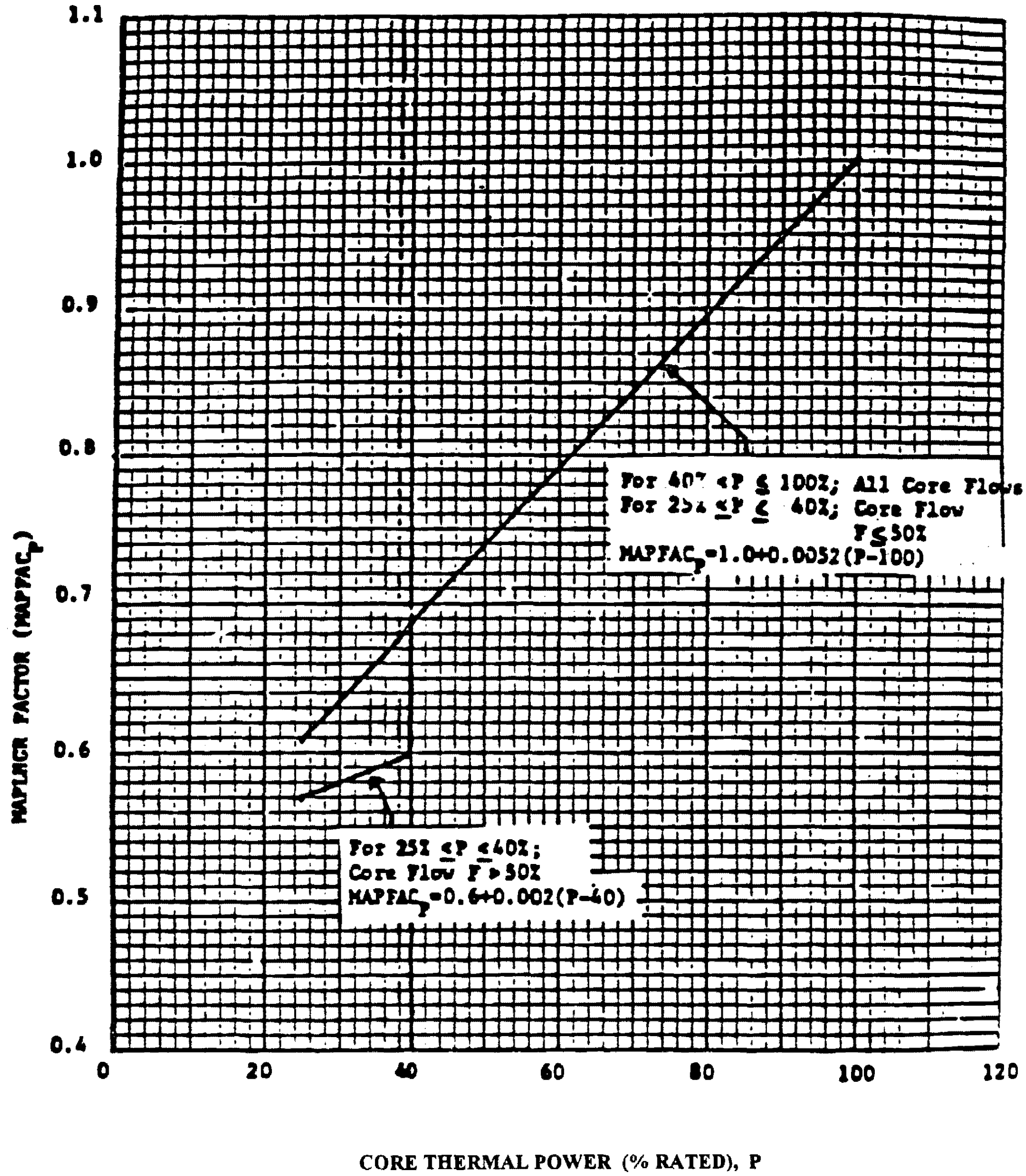
** All evaluations and results are limited to GE6 fuel used in operating strategies with the target Haling end of cycle exposure distribution. Various modes of spectral shift operation in which the cycle average void distribution significantly exceeds that obtained with a Haling strategy can violate the validity of these limits.



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	POWER DEPENDENT MCPR LIMITS (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15D.14-1



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	POWER DEPENDENT MAPLHGR LIMIT (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15D.14-2



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	POWER DEPENDENT MAPLHGR LIMITS (INITIAL CYCLE) [HISTORICAL INFORMATION] FIGURE 15D.14-3

15D.15 REFERENCES

1. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors" NEDO-24154 Oct. 1978.

2. R. B. Linford "Analytical Methods of Plant Transients Evaluations for the General Electric Boiling Water Reactor" NEDO-10802 April 1973.

3. "Three Dimensional BWR Core Simulator" NEDO-20953-A, January 1977.

4. Letter, J. S. Charnley (GE) to F. J. Miraglia (NRC), "Loss of Feedwater Heating Analysis", July 5, 1983 (MFN-125-83).

5. NEDE-22277-P-1, "Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria", October 1984.

6. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTARII)", latest edition.

7. "BWR Core Thermal Hydraulic Stability" Service Information Letter, SIL 380 Revision 1, February 1984.

8. "Grand Gulf-1 Reactor Internals Vibration Measurements Summary Report" NEDE 31148P February 1986.

9. General Electric Company Report "GGNS Maximum Extended Operating Domain Analysis", March 1986.

10. NEDC-33612P, Rev. 0, “Safety Analysis Report for Grand Gulf Nuclear Station Maximum Extended Load Line Limit Analysis Plus,” September 2013.

ADDENDUM 1 TO THE APPENDIX 15D GGNS MAXIMUM EXTENDED OPERATING DOMAIN ANALYSIS

The updated analysis of the FWHOS is presented in Appendix 15B.

INTRODUCTION

This addendum addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [This addendum presents the Grand Gulf Nuclear Station (GGNS) plant specific analysis results of the 100°F loss of feedwater heating (LFWH) transient for Maximum Extended Operating Domain (MEOD) analysis and Feedwater Heater(s) Out-of-Service (FWHOS) operation in MEOD. The plant specific analysis results presented in this Addendum replace the generic-statistical-evaluation based LFWH analysis results which are presented in Section 15D.4.1 for MEOD and in Section 15D.13.2 for MEOD/FWHOS of the GGNS MEOD analysis report (Reference 1).]

ANALYSIS BASES

[HISTORICAL INFORMATION] [The GGNS plant specific analysis for the 100°F loss of feedwater heating event was performed using the General Electric three-dimensional BWR Core Simulator (Reference 2) documented in the NRC approved GESTAR Amendment (Reference 3). The analysis was performed for the P8X8R fueled GGNS Cycle 1 core. All the cases analyzed were initiated from the GGNS cycle 1 exposure accounting cases based on actual plant data. The following cases were run at the most limiting exposure point in cycle:

1. 104.2% power/100% core flow with an initial FW temperature corresponding to rated FW temperature of 420°F (1 case).

2. 104.2% power/75% core flow and 104.2% power/108% core flow with an initial FW temperature corresponding to rated FW temperature of 420°F (2 cases).

3. 104.2% power/100% core flow with an initial FW temperature corresponding to rated FW temperature of 370°F (1 case).]

ANALYSIS RESULTS

[HISTORICAL INFORMATION] [The critical power ratio (CPR) results are summarized in Table 1. This analysis shows the following results:

1. MEOD

   (a) The LFWH transient impact on CPR is less severe than the Feedwater Controller Failure (FWCF) transient case. The worst LFWH ΔCPR is 0.07 at rated and increased core flow conditions (ICF) compared to the FWCF ΔCPR of 0.09 (Table 15D.4-5 of Reference 1). The MCPR is 1.11, which is above the safety limit MCPR of 1.06.

   (b) ΔCPR at 75% core flow is slightly larger than the high core flow case. However, a larger CPR margin exists at this low core flow because of the higher OLMCPR required by the flow dependent MCPR limit. The MCPR is 1.19, which is well above the safety limit MCPR of 1.06.

2. FWHOS in MEOD

   (a) The 100°F LFWH event has less effect with colder feedwater than with the normal feedwater temperature (Case 2 vs. 4 in Table 1). The results confirmed that the LFWH event is less severe when initiated from lower initial feedwater temperature than from the rated feedwater temperature. This trend is not affected by initial core flow. Therefore, the LFWH analysis for FWHOS in MEOD is adequately bounded by the 420°F ΔCPR results.]

CONCLUSIONS

[HISTORICAL INFORMATION] [Based on the results of this analysis it is concluded that no change in current Technical Specification MCPR limits is required for operation in MEOD or FWHOS operation in MEOD for the range of rated feedwater temperature from 420°F to 370°F. This conclusion is the same as that given in Reference 1.

The plant specific results are bounded by the generic statistical bounding value in Reference 4.]

REFERENCES

1. GGNS Maximum Extended Operating Domain Analysis, March 1986, Appendix 15D.

2. J. A. Wooley, "Three-Dimensional BWR Core Simulator," NEDO-20953-A, January 1977.

3. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-US, Revision 7, August 1985. (GE Proprietary).

4. J. S. Charnley to the USNRC, "Loss of Feedwater Heating Analysis", July 5, 1983.

TABLE 1: SUMMARY OF THE 100°F LFWH TRANSIENT CPR RESULTS
104.2% POWER (INITIAL CYCLE)
[HISTORICAL INFORMATION]

Case	Core Flow (% NBR)	ICPR	ΔCPR	MCPR	Rated FW Temperature (°F)
1	108.0	1.18	0.07	1.11	420
2	100.0	1.18	0.07	1.11	420
3	75.0	1.27	0.08	1.19	420
4	100.0	1.18	0.06	1.12	370

ADDENDUM 2 TO THE APPENDIX 15D GGNS MAXIMUM EXTENDED OPERATING DOMAIN ANALYSIS

This addendum addresses analyses performed by the NSSS vendor (GEH) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

[HISTORICAL INFORMATION] [This addendum provides supplemental information which was requested by the NRC for clarification of some contents of the GGNS Maximum Extended Operating Domain Analysis Report, March 1986.

1. Section 15D.4.1 Abnormal Operating Transients

   In page 15D.4-1 it was stated that the results of the Grand Gulf unique evaluation show that the ΔCPR results for all the cases analyzed are bounded by the bounding BWR/6 Standard Plant analysis presented in Table 15D.4-1. However, a comparison of the ΔCPR results in Table 15D.4-1 and Table 15D.4-5 shows that this statement is correct except one case; the 104.2/75 FWCF ΔCPR 0.084 in Table 15D.4-1 does not bound the ΔCPR 0.09 of the same transient in Table 15D.4-5.

   The BWR/6 bounding ΔCPR results in Table 15D.4-1 are the values calculated per ICPR = 1.06 + ΔCPR for each transient. Therefore, the ICPR values are not the same as those given in Table 15D.4-5. For the case in question, the ICPR is 1.096 compared to 1.27 for the Grand Gulf unique case. The bounding BWR/6 ΔCPR for ICPR of 1.27 is 0.09 (>0.084) which is equal to the Grand Gulf unique value of 0.09.

   Therefore, the Grand Gulf unique ΔCPR results for all the cases analyzed are either equal to or bounded by the bounding BWR/6 Standard Plant analysis.

2. Section 15D.6 Loss of Coolant Accident Analysis

   The bounding BWR/6 Loss of Coolant Accident (LOCA) analysis was performed in the Maximum Extended Operating Domain (MEOD) boundary and the results were reviewed for GGNS. It was concluded that the initial cycle MAPLHGR limits presented in Chapter 6 are adequate to cover the entire MEOD. The results of the study are described below.

A generic BWR/6 LOCA analysis was performed along the entire MEOD boundary as defined in Figure 15D.3-1. The analysis was conducted to define MAPLHGR restrictions (multipliers) versus core flow (if any) required to cover operation in the MEOD region. This analysis was performed by comparing the values of key parameters which affect the peak cladding temperature (PCT) along the MEOD boundary to the equivalent values in the GGNS FSAR. Based on this evaluation, it was concluded that operation in the MEOD region results in no more than 5°F PCT increase over the initial cycle GGNS FSAR 6.3 ECCS analysis.

Operation at core flows greater than rated tends to reduce the calculated PCT slightly due to the higher core flow during the period when the recirculation pumps are coasting down. It is the extended load line in the MEOD region which is a concern with regard to ECCS performance and PCT response. The higher rod line will permit a higher power (higher initial stored energy in the fuel) at a given flow. This increases the chance of losing nucleate boiling at the highest power axial node prior to the time of jet pump uncovery. This phenomena is called early boiling transition (BT), and could affect the calculated PCT.

The two major parameters that affect PCT in the design basis LOCA calculation which are sensitive to the higher core power and lower core flow are the time of BT at the high power axial node of the limiting fuel assembly and the calculated reflooding time. Early BT results in a less efficient removal of the initial stored energy from the fuel, which tends to increase the calculated PCT. The lower initial power at lower core flow tends to decrease the calculated PCT. This occurs because the lower power results in lower core spray vaporization, leading to less counter current flow limitation (CCFL) at the upper tie plate in the fuel bundles and an earlier core reflooding time.

The variation of the bundle inlet flow during a LOCA event is determined by a number of parameters, the most important being the break size, the water inventory in the reactor at the start of the event, and the steady state core power and flow conditions. The first two of these are accounted for in the standard FSAR LOCA analysis. The effect of variations in the third was accounted for by performing analyses at the power flow points along the MEOD boundary of Figure 15D.3-1. The assumed initial minimum critical power ratio (MCPR) is also an important parameter in determining whether or not early BT will occur at the high power axial node. Credit was taken for the initial cycle requirement on MCPR versus core flow with an additional 2 percent conservatism added to satisfy 10CFR50 Appendix K. The required MEOD flow dependent MCPR limit was reviewed to show that there is no significant impact on the results.

RESULTS

The generic ECCS analysis for the BWR/6-218 standard plant showed no early BT prior to jet pump uncovery at the high power node for any of the analysis points on the MEOD boundary. Thus, no MAPLHGR reductions are required for operation in the MEOD region from ECCS considerations. The BWR/6-218 was selected as the bounding plant based on lower initial and minimum hot bundle mass flux as shown in Table 1. The lower initial and minimum hot bundle mass flux for a BWR/6-218 makes it the most sensitive BWR/6 to lower initial core flows and thus bounding in terms of any PCT increase which might result from operation in the MEOD. The most limiting point in the analysis was found to be at 102 percent power/85 percent core flow. As stated above, the mass flux in the hot bundle is a key parameter in determining whether or not early BT will occur. Table 2 shows the effect of the reduced core flow on the time of BT. As can be seen from Table 2, early BT will not occur at the high power axial node along the MEOD boundary. Tables 1 and 2 also show that the GGNS response was very similar to that for the BWR/6-218 plant. Thus, the results of the generic study are applicable to GGNS and no MAPLHGR reductions are required from ECCS considerations. Figure 1 shows the normalized core flow versus time plot for the GGNS DBA recirculation suction line break at 100 percent and 85 percent core flow. Figure 2 shows the calculated MCPR versus time for the BWR/6-218 plant at 100, 85, and 75 percent core flow to demonstrate that the most severe response is at 85 percent core flow.

The slightly earlier high power node BT times shown in Table 2 at 75, 80, and 110 percent core flow occur after jet pump uncovery for the BWR/6-218 plant and are estimated to result in a PCT increase of about 2°F. This estimate is based on generic sensitivity studies which show a PCT increase of about 20°F for a one second change in dryout time when dryout occurs near ten seconds. The changes in dryout time are of similar magnitude for

GGNS, with a conservative estimate of less than 5°F PCT increase over the maximum initial cycle PCT reported in section 6.3 of the FSAR.

The ECCS analysis in MEOD described above concluded that the initial cycle GGNS FSAR Chapter 6 results are impacted by less than 5°F PCT and therefore meet the 10CFR50.46 limits.

3. Section 15D.4.3 Flow Runout Transient

This section addresses analyses performed by the NSSS vendor (GE) for the initial fuel cycle. Discussions applicable to the current fuel cycle are provided in the appropriate sections in the main body of the UFSAR. See the discussion in subsection 15.0 for additional information on the relationship between the analyses performed by the NSSS vendor and the reload fuel vendor.

In this section, it was stated that this event was reanalyzed as part of the MEOD program to include the highest rod line for the ELLR, up to 102.5% maximum flow and the ICFR, up to 107% maximum core flow. The following supplements this statement.

The $MCPR_f$ curves for 102.5% and 107% maximum core flow were generated following the same basic procedure and approach as used for a BWR/4 MG set plant and therefore, the curves would be applied in a similar manner. The only difference is that a scoop tube limits the maximum core flow in the MG set plant, whereas in a flow control valve plant like GGNS the electric output of the Flow Controller in the recirculation flow control system limits the valve position and maximum core flow. The flow limit is set in such a manner that core flow does not exceed the maximum flow at the rated power.

The $MCPR_f$ calculation is based on a rod line with the limiting slope which bounds possible variations of the slope under any Xenon condition, equilibrium or non-equilibrium. The approved BWR Core Thermalhydraulic Analysis Code was used for MCPR calculation. The $MCPR_f$ for 102.5% maximum core flow was calculated as follows. First, the MCPR of the peak power bundle in the core was put on the safety limit MCPR of 1.06 at 104.2% power (105% steam flow)/102.5% core flow. This was done by adjusting the power of the peak power bundle. Next, the peak power bundle MCPR was calculated along the limiting rod line to determine the $MCPR_f$ as function of core flow.A similar procedure was used for the 107% maximum core flow $MCPR_f$ calculation. The only difference was that the peak power bundle MCPR was put on 1.06 at 107% core flow and core power corresponding to 107% core flow on the same rod line.

The slow flow runout analyses have been performed for the steepest rod line in the ELLR region and compared to results of the analysis for the 105% rod line. The evaluation has shown that the $MCPR_f$ curves based on the 105% rod line bound the highest rod line case (ELLR case) because the maximum core flow attainable on the ELLR rod line is less for a given maximum recirculation FCV position due to higher two-phase pressure drop.]

TABLE 1: EFFECT OF CORE FLOW ON HOT BUNDLE INLET MASS FLUX (INITIAL CYCLE) [HISTORICAL INFORMATION]

	Initial Core Flow (% Rated)	Initial Hot Bundle Mass Flux (1bm/hr-ft2)	Minimum Hot Bundle Mass Flux* (1bm/hr-ft2)
BWR/6-218 Std Plant			
	110	1.177 E6	0.463 E6
	100	1.050 E6	0.411 E6
	85	0.890 E6	0.363 E6
	80	0.835 E6	0.341 E6
	75	0.782 E6	0.319 E6
	70	0.731 E6	0.298 E6
	65	0.681 E6	0.275 E6
	60	0.632 E6	0.248 E6
	55	0.583 E6	0.221 E6
	45	0.487 E6	0.157 E6
GGNS			
	100	1.083 E6	0.424 E6
	85	0.910 E6	0.368 E6

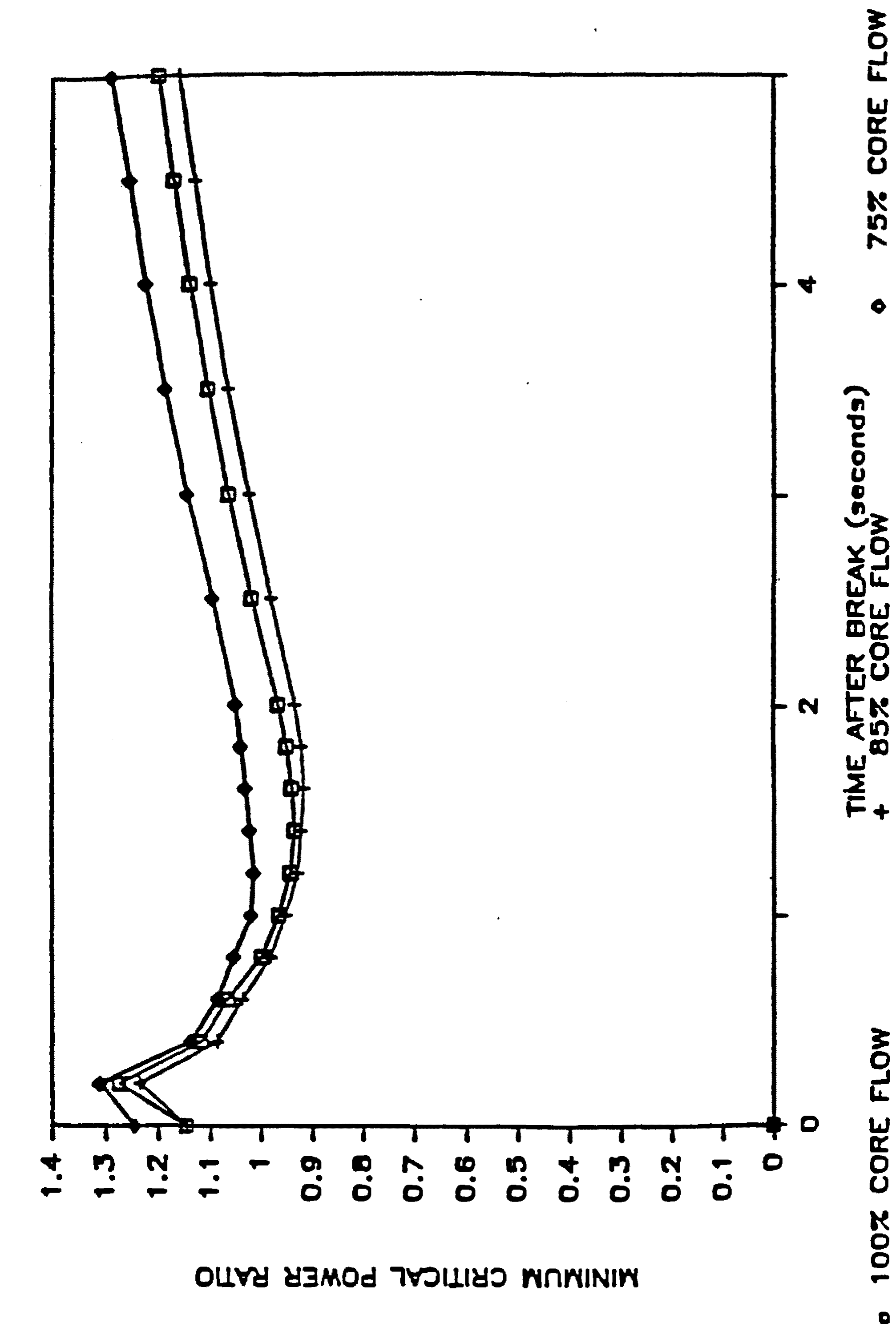
* For t < 1.0 seconds

TABLE 2: EFFECT OF CORE FLOW ON BOILING TRANSITION (BT) IN THE HOT CHANNEL FOR DBA RECIRCULATION SUCTION LINE BREAK (INITIAL CYCLE) [HISTORICAL INFORMATION]

	Initial Core Flow (% Rated)	Initial MCPR	Time of BT for High Power Node (Sec) **	Time of BT for Upper Node (Sec)	Elevation of Upper Node (Ft from TAF)
BWR/6-218 Std Plant					
	110	1.147	9.88*	0.98	2.344
	100	1.147	9.96	1.18	4.427
	85	1.147	10.18	1.02	4.427
	80	1.206	9.88*	0.94	2.344
	75	1.245	9.92*	9.92	0.
	70	1.279	10.16	10.16	0.
	65	1.319	10.22	10.22	0.
	60	1.343	10.12	10.12	0.
	55	1.377	10.62	10.62	0.
GGNS					
	100	1.147	9.60	1.32	4.427
	85	1.147	9.54*	1.00	4.427

* Slightly earlier than base case, but PCT impact is less than 5°F.

**BT occurs after jet pump uncovery, which is approximately 8 seconds after LOCA.



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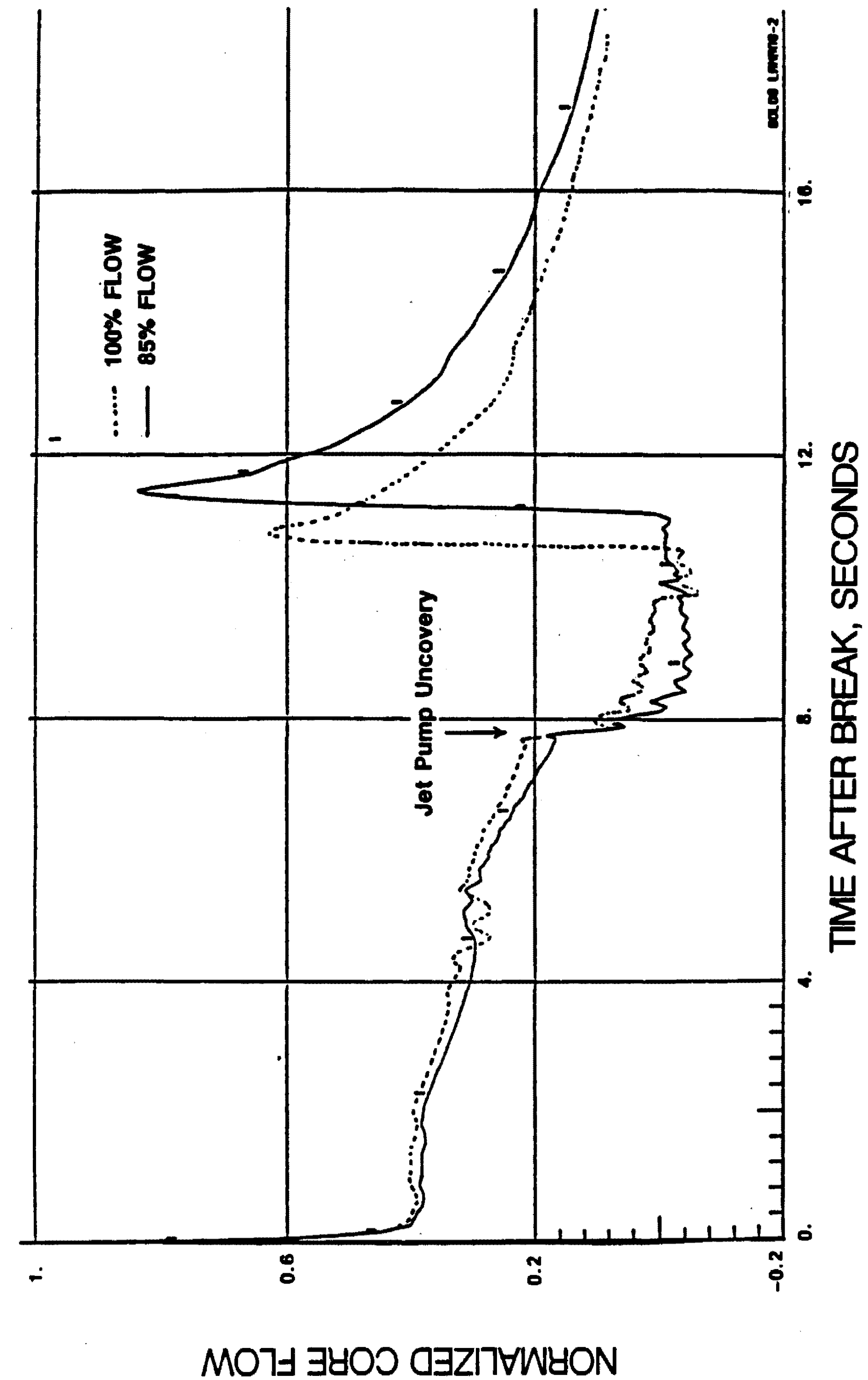
FLOW EFFECT ON MCPR
DBA SUCTION BREAK
(INITIAL CYCLE) [HISTORICAL INFORMATION]
FIGURE 1



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NORMALIZED CORE FLOW
VERSUS TIME
(INITIAL CYCLE) [HISTORICAL INFORMATION]
FIGURE 2