TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

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LIST OF TABLES (Continued)

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LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

15.0 ACCIDENT ANALYSES

15.0 GENERAL

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

These analyses were originally performed to evaluate plant operation within the standard power-flow operating map <Figure 4.4-2>. Subsequently, analyses were performed which extended the analyzed operating region and allowed operation with certain equipment out-of-service. These modes of operation are discussed in general in <Section 15.0.5>. Specific discussions and analyses are presented in the following Chapter 15 Appendices:

- <Appendix 15D> Partial Feedwater Heating Operation Analysis
- <Appendix 15E> Maximum Extended Operating Domain Analysis
- <Appendix 15F> Recirculation System Single-Loop Operation Analysis

For power uprate to 3,758 MWt core thermal power, some of the baseline transient analyses are re-evaluated. Specifically, per Appendix E of (Reference 10), analyses are performed for the limiting transient events, which includes all events that establish the core thermal operating limits and the events that show bounding conformance to the other transient protection criteria (e.g., ASME overpressure limits).

Transient events that are re-analyzed at the uprated power are labeled "at 3,758 MWt core power." The transient events that are not re-analyzed are preserved without any updates or labels. The key input conditions for the 3,758 MWt core power cases are listed in <Table 15.0-1> along with the values for the original baseline analyses.

For partial arc operation, a limiting cases evaluation is used similar to power uprate. For the limiting events, full arc operation is assumed in the analyses which is more limiting than partial arc operation. Non-limiting events are not re-analyzed for partial arc and are preserved without any updates.

Certain limiting safety analyses are reperformed each operating cycle to determine and/or verify safety margins. A discussion of the reload safety analysis process is presented in <Section 15.0.6>. The methods, input conditions, and results for the current cycle are presented in <Appendix 15B> of this Chapter.

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.

This chapter addresses two types of operating conditions addressed by the Code of Federal Regulations. It compares the radiation releases from anticipated operational transients to the <10 CFR 20> limits on the "anticipated average radiation levels." The consequences of very unlikely events (faulted events) are compared to the <10 CFR 100> or <10 CFR 50.67> (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <10 CFR 50.67> limits. The analyses described in this chapter show that the consequences for these two types of events are less severe than the corresponding 10 CFR limits.

Unless otherwise identified, it is assumed that all equipment (safety grade or nonsafety grade) is available to mitigate the transients described and analyzed in this chapter. However, only safety grade equipment is assumed to be used to mitigate accidents and safely shut down the reactor.

15.0.2 ANALYTICAL CATEGORIES

Transient and accident events contained in this chapter are discussed in individual categories as required by <Regulatory Guide 1.70>. The results of the analyses of these events are summarized in <Table 15.0-2a> and <Table 15.0-2b> for events in the main text of <Chapter 15>. <Appendix 15D>, <Appendix 15E> and <Appendix 15F> present summary tables for partial feedwater heating, ME0D and single loop operation respectively. <Appendix 15B>, Reload Safety Analysis presents these results for the events analyzed for each reload. Each event is assigned to one of the following categories:

a. Decrease in Reactor Coolant Temperature:

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

b. Increase in Reactor Pressure:

Excessive reactor pressure increases could result in rupture of the reactor coolant pressure boundary (RCPB). Increasing pressure also collapses moderator voids thereby increasing core reactivity and power level which could result in fuel cladding failure from overheating.

c. Decrease in Reactor Coolant System Flow Rate:

A reduction in the core coolant flow rate could cause overheating of the fuel cladding if 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 due to increased core flow disturbances. Increased core flow reduces the void content of the moderator, thereby 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 component containing radioactivity is postulated in this category.

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 situation is postulated.

In order to address all of the credible transient events in these eight categories, the initial operating licenses for BWR plants are based on the analysis of a spectrum of approximately 20 to 25 USAR events, assignable to one of the above categories. In this manner, the most severe transient events relative to LHGR, CPR, and reactor coolant system pressure are identified. A review of these transient results was used to determine which transients have the potential for being limiting. From this General Electric has established that the limiting transients will always be within a set of transients identified in <Section 15.0.6>.

15.0.3 EVENT EVALUATION

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

Conditions which lead to the initiating events analyzed in this chapter are described within the categories designated above. The frequency of occurrence of each event was determined based upon available operating data at the time of analysis as described below:

a. Incidents of moderate frequency - these are incidents that may occur once during a calendar year to once every 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) above in the Nuclear Systems Operational Analyses in <Appendix 15A>.

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 100>.

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 100> or <10 CFR 50.67> (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <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 5 rem whole body, 30 rem inhalation and 75 rem skin (5 rem TEDE for the alternative source term LOCA and Fuel Handling Accident analyses).

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

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.

15.0.3.2.1 Single Failure and Operator Error Criteria

15.0.3.2.1.1 General

This section discusses application of "single failure" and "single operator error" criteria to the analyses of the postulated events in this chapter. Single active component failure (SACF) criteria are applied to design basis accident categories only.

Transient evaluations are judged against a criteria 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 utilize safety systems which can accommodate SACF criteria). Even under these postulated events, the plant damage allowances or limits are much the same as those for normal operation.

15.0.3.2.1.2 Initiating Event Analysis

Initiating event analysis consists of the following:

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

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

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

d. any single electrical component failure, or

e. any single operator error.

Operator error is defined as an active deviation from written operating procedures or nuclear plant standard operating practices. 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.

In addition, actions subsequent to the initial operator error which have an effect on the designed operation of the plant, but are not necessarily directly related to the operator error, may be included.

Examples of single operator errors are as follows:

a. An increase in power above the established flow control power limits by control rod withdrawal out of 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 or Single Operator Failure Analysis

Single active component failure or single operator failure analysis is as follows:

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

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

15.0.3.3 Core and System Performance

15.0.3.3.1 Introduction

<Section 4.4>, "Thermal and Hydraulic Design," describes the various fuel failure mechanisms. Avoidance of unacceptable results <Section 4.4.1.4> for incidents of moderate frequency is verified statistically with consideration given to data 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 (Reference 1). 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 for the initial and reload cores. 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 MCPR during significant abnormal events is calculated using a transient core heat transfer analysis computer program. The computer program is based on a multinode, single channel thermal-hydraulic model which requires 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 model include a physical description of the bundle, and channel inlet flow and enthalpy, pressure and power generation as functions of time.

A detailed description of the analytical model may be found in Appendix C of (Reference 3). The initial condition assumed for all full power transient MCPR calculations is that the fuel bundle is operating at or above the MCPR Operating Limit for the initial and subsequent reload cores. Maintaining MCPR greater than the MCPR Safety Limit for the initial and subsequent reload cores 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."

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 System."

15.0.3.3.2 Input Parameters and Initial Conditions for Analyzed Events

In general, the events analyzed within this chapter have values for input parameters and initial conditions as specified in <Table 15.0-1>. The input parameters and initial conditions for the analyses supporting PNPP operation in various operating modes and/or with equipment out-of-service are presented in tables contained in the following Chapter 15 appendices:

- <Appendix 15D> Partial Feedwater Heating Operation Analysis
- <Appendix 15E> Maximum Extended Operating Domain Analysis
- <Appendix 15F> Recirculation System Single-Loop Operation Analysis

Certain limiting or potentially limiting safety analyses are reperformed each operating cycle to determine and/or verify safety margins. The methods, input parameters and initial conditions for each transient reanalyzed as part of the reload are presented in <Appendix 15B> of this Chapter. Analyses which assume data inputs different than these values are designated accordingly in the appropriate event discussion.

15.0.3.3.3 Initial Power/Flow Operating Constraints

The analytical basis for most of the initial core transient safety analyses is the thermal power at rated core flow (100 percent) corresponding to 105 percent nuclear boiler rated (NBR) 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. 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 core flow line A-H, and the left bound is 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 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- 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- 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 (Reference 1) 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- 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 limits.

Certain localized events are evaluated at other than the above mentioned conditions. These conditions are discussed pertinent to the appropriate event. Reactor operation up to the APRM rod block line, which is above the power levels corresponding to the design flow control line except at low drive flows, is assumed for ECCS analysis.

15.0.3.3.4 Results

The results of analytical evaluations are provided for each event. In addition critical parameters are shown in <Table 15.0-2a> and <Table 15.0-2b>. From the data in these tables and the other similar tables an evaluation of the limiting event for that particular category and parameter can be made.

Similar tables are provided for each of the extended operating domain/modes of operation <Appendix 15D>, <Appendix 15E> and <Appendix 15F>; partial feedwater heating, MEOD and single loop operation respectively). <Appendix 15B>, Reload Safety Analysis contains a summary table for all reanalyzed reload events. In <Table 15.0-3> a summary of applicable accidents is provided. This table compares the GE calculated amount of failed fuel to that used in worst case radiological calculations.

15.0.3.4 <u>Barrier Performance</u>

This section 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.

15.0.3.4.1 Reactor Coolant Pressure Boundary Performance

The significant areas of interest for internal pressure damage are the high pressure portions of the reactor coolant pressure boundary (the reactor vessel and the high pressure pipelines attached to the reactor vessel). The overpressure below which no damage can occur is defined as the pressure increase over design pressure allowed by the applicable ASME Boiler and Pressure Vessel Code Section III for the reactor vessel and the high pressure nuclear system piping. Because this ASME Code permits pressure transients up to 10 percent over design pressure for upset events, (Reference 4) the design pressure portion of the reactor coolant pressure boundary meets the design requirement if peak nuclear system pressure remains below 1,375 psig (110 percent x 1,250 psig). Comparing the events considered in this section with those used in the mechanical design of equipment reveals that either the accidents are the same or that the accident in this section results in less severe stresses than those assumed for mechanical design.

The Low-Low Set (LLS) 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 LLS design on reactor coolant pressure is demonstrated, in <Chapter 5>, on the MSIV closure event. This is considered bounding for all other pressurization events and, therefore, is not simulated in the analysis presented in this chapter.

A sensitivity study was also performed to support higher analytical limits for relief valve setpoints. The study shows an increase of 20 psi in the relief valve setpoint causes less than 20 psi increase in reactor peak pressures. However, these reactor peak pressures are still well below the ASME code limit (1,375 psig). Also, the increase of 20 psi in the relief setpoints does not have any effect on the peak surface heat flux, since all safety/relief valves open after the occurrence of MCPR during transients. Therefore, the analytical limits for relief valve setpoints in the Technical Specifications were 20 psi higher than those listed in <Table 15.0-1>.

The analytical limits used for the relief valve setpoints for the current reload analysis are listed on <Table 15B.15.0-1> for the pressurization transients and on <Table 15B.5.2-1> for the overpressurization transients. The analytical values are the basis for the deviation of the Technical Specification value.

15.0.3.5 Radiological Consequences

In this chapter, the consequences of radioactivity release during the three types of events: incidents of moderate frequency (anticipated operational transients), infrequent incidents (abnormal operational transients) and 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 typically considered:

a. The first is based on conservative assumptions for the purposes of worst case bounding of event consequences to determine the adequacy of the plant design to meet <10 CFR 100> or <10 CFR 50.67> (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <10 CFR 50.67>) guidelines. This analysis is referred to as the "design basis analysis."

b. The second is based on realistic assumptions to reflect expected radiological consequences. This analysis is referred to as the "realistic analysis." The "realistic analysis" is not performed for the LOCA analysis, or the Fuel Handling Accident.

Results for both are shown to be within NRC guidelines.

Doses resulting from the events in <Chapter 15> are determined either manually or by computer code. Doses associated with Offgas System Failure <Section 15.7.1.1> are evaluated using GASPAR II <NUREG/CR-4653> (Reference 8). Time dependent releases are evaluated with the TACT computer code (Reference 2)(Reference 6). Instantaneous or "puff" type releases are evaluated by methods based on those presented in <Regulatory Guide 1.3>, and <Regulatory Guide 1.183>. The General Electric NEDO-31400 analysis (Reference 7) also is utilized in determining doses associated with a Control Rod Drop Accident <Section 15.4.9>. Dose conversion factors and breathing rates are presented in <Table 15.0-4>.

15.0.4 NUCLEAR SAFETY OPERATIONAL ANALYSIS (NSOA) RELATIONSHIP

<Appendix 15A> is a comprehensive, total plant, system-level, qualitative failure modes and effects analysis, relative to all the <Chapter 15> events considered, the protective sequences utilized to

accommodate the events and their effects, and the systems involved in protective actions. Interdependency of analysis and cross-referral of protective actions are integral to this chapter and the appendices.

Contained in <Appendix 15A> is a summary table which classifies events by frequency only (i.e., not just within a given category such as decrease in core coolant temperature).

15.0.5 EXTENDED OPERATING DOMAINS AND MODES OF OPERATION

The Perry Nuclear Power Plant (PNPP) has been analyzed for the modes of operation described in the appendices listed below:

- <Appendix 15D> Partial Feedwater Heating Operation Analysis
- <Appendix 15E> Maximum Extended Operating Domain Analysis
- <Appendix 15F> Recirculation System Single-Loop Operation Analysis

Certain limiting safety analyses are reperformed each operating cycle to determine and/or verify safety margins. The reload methodology is discussed in general is in <Section 15.0.6> and the current cycle results are presented in <Appendix 15B> of this chapter.

A brief summary of the operating domains and modes of operation is provided below. <Table 15.0-5> provides a summary listing of those transients/accidents analyzed under the above appendices.

15.0.5.1 Partial Feedwater Heating (PFH) Operation

<Appendix 15D> presents the results of a safety and impact evaluation for the operation of Unit 1 of the Perry Nuclear Power Plant (PNPP) with partial feedwater heating at steady-state conditions during the operating cycle and beyond the end-of-cycle conditions. This evaluation was performed at 3,758 MWt for core power and is applicable for subsequent reload cycles. The results of this evaluation justify PNPP

operation at 100% thermal power under steady-state conditions with a rated feedwater temperature ranging from 425.5°F to 325.5°F, and operation under beyond end-of-cycle conditions with a rated feedwater temperature ranging from 425.5°F to 255.5°F.

Operation with partial feedwater heating (PFH) occurs in the event that (1) certain stage(s) or string(s) or individual heater(s) becomes inoperable, or (2) by intentionally valving out the extraction steam to the feedwater heaters at the end of an operating cycle.

A discussion of potential modifications to the Technical Specification MCPR limits necessary to implement partial feedwater heating is provided in <Section 15.0.5.2>.

15.0.5.2 Maximum Extended Operating Domain (MEOD) Operation

<Appendix 15E> presents the results of a safety and impact evaluation for operation of Unit 1 of the Perry Nuclear Power Plant (PNPP) in an expanded operating envelope called the Maximum Extended Operating Domain. This permits improved power ascension capability to full power as well as provide additional flow range at rated power including an increased flow region to compensate for reactivity reduction due to exposure during an operating cycle.

The Maximum Extended Operating Domain (MEOD) is shown in <Figure 15E.2-1>. The extended load line region (ELLR) boundary is limited by 81% core flow at 100% power and the MEOD Boundary Line as defined in USAR 15E.2. The Increased Core Flow Region (ICFR) is bounded by the 105% core flow line.

The MEOD appendix also justifies Partial Feedwater Heating (PFH) operation as described in <Appendix 15D> for rated feedwater temperature ranging from 425.5°F to 325.5°F during and beyond the operating cycle in the MEOD (ELLR and ICFR), and for a feedwater temperature ranging from 325.5°F to 255.5°F for beyond end-of-cycle conditions in the ICFR.

Modifications to the Technical Specification MCPR limits may be required each cycle to define the operating limit MCPR for each temperature regime. A summary of the rated operating limit MCPR values for various modes of operation is presented in <Appendix 15B>, Reload Safety Analysis for the current reload cycle.

15.0.5.3 Recirculation System Single-Loop Operation

<Appendix 15F> justifies that Unit 1 of the Perry Nuclear Power Plant can safely operate with a single recirculation loop out-of-service at up to approximately 67% of rated thermal power. This evaluation is applicable to initial and reload cycle operation with normal feedwater heating and within the standard operating domain shown in <Figure 4.4-2> of the USAR.

Single-loop operation (SLO) is desirable in the event a recirculation pump problem or other component maintenance renders one loop inoperative. To justify single-loop operation, accidents and abnormal operational transients associated with power operation, as presented in <Chapter 6> and <Chapter 15>, were reviewed with one recirculation pump in operation. Increased uncertainties in the total core flow and traversing in-core probe (TIP) readings result in a small increase in the fuel cladding integrity safety limit MCPR.

SLO can also result in changes to plant response during a LOCA. These changes are accommodated by the application of reduction factors to the two-loop operation MAPLHGRs if required. MAPLHGR reduction factors are evaluated on a plant-by-plant and fuel type dependent basis. In each subsequent reload, reduction factors are checked for validity and, if new fuel types are added, new reduction factors may be needed in order to maintain the validity of the SLO analysis. MAPLHGR reduction factors for the current reload cycle (as needed) are provided in the Reload Safety Analysis, <Appendix 15B>.

15.0.6 RELOAD SAFETY ANALYSIS

For each cycle, analyses are performed to ensure that with the new fuel and core arrangement, operation will occur within the specified operating limits, margin is demonstrated for anticipated operational occurrences (transients), and unacceptable consequences will not result for design basis events. If the results of these analyses demonstrate that the above criteria may not be met, adjustments are made to the operating limits to ensure the plant will continue to meet its' licensing safety analysis basis. For reload analysis purposes, a subset of the original FSAR events is reanalyzed to confirm that the plant will continue to meet the requirements of the safety analysis.

15.0.6.1 Determination of the Limiting Transients

In order to address all of the credible transient events in the eight categories, the initial operating licenses for BWR plants are based on the analysis of a spectrum of approximately 20 to 25 USAR events, assignable to one of the above categories. In this manner, the most severe transient events relative to LHGR, CPR, and reactor coolant system pressure are identified. To determine the limiting transient events to be analyzed for each reload, a generic approach as documented and described in GESTAR (Reference 5) was followed.

This generic approach involved examining the relative dependency of critical power ratio (CPR) on various thermal-hydraulic parameters. Sensitivity studies were performed generically to determine the effect of changes in bundle power, flow, subcooling, R-factors, and pressure on CPR for different fuel designs. It was found that the CPR is most dependent on the R-factor and bundle power. A slight sensitivity to pressure and flow changes and relative independence to changes in inlet subcooling was also shown. The R-factor is a function of bundle geometry and local power distribution and is assumed to be constant throughout the transient. Therefore, transients which would be limiting in CPR would primarily involve significant changes in power.

A review of these transient results (e.g., pressure, power, flow, was used to determine which transients have the potential for being limiting. From this, General Electric has established that the limiting transients will always be within a certain set of transients, identified below. These are the transients which involve significant effects on power, heat flux and reactor vessel pressure peaks.

Based on results of this sensitivity study, it was concluded that the anticipated operational occurrences most likely to limit operation because of MCPR considerations for a BWR are:

1. Limiting Decrease in Reactor Coolant Temperature Event:
   Loss of Feedwater Heating (Manual Control) - <Section 15.1.1>

2. Limiting Temperature Decrease/Pressurization Event:
   Feedwater Controller Failure (Maximum Demand) - <Section 15.1.2>

3. Limiting Pressurization Events:
   Pressure Regulator Failure Downscale – <Section 15.2.1>
   [BWR/6 only]
   Generator Load Rejection Without Bypass - <Section 15.2.2>
   or
   Turbine Trip Without Bypass – <Section 15.2.3>

4. Reactivity and Power Distribution Anomalies:
   Rod Withdrawl Error at Power - <Section 15.4.2>

General Electric in the reload topical report (Reference 5) established and the NRC concurred in their Safety Evaluation Report (Reference 5) that most of the events analyzed as part of the USAR need not be reanalyzed or reassessed for plant-specific reload core licensing applications. Therefore, only the above limiting or potentially limiting transients are reanalyzed each cycle. A discussion of the reload process, input conditions, transients, and results is presented in <Appendix 15B>, Reload Safety Analysis. If Exposure-Dependent MCPR Limits are used, the pressurization transients are analyzed for each exposure interval.

<Table 15.0-5> provides a summary listing of those transients/accidents typically evaluated as part of the reload safety analysis.

15.0.6.2 MCPR Operating Limit Calculational Procedure

A plant and cycle unique MCPR operating limit is established to provide adequate assurance that the fuel cladding integrity or MCPR safety limit for that plant is not exceeded for any moderate frequency, transient. This operating requirement is obtained by addition of the maximum DCPR value for the most limiting transient (including any imposed adjustment factors) from conditions postulated to occur at the plant to the MCPR safety limit.

15.0.7 REFERENCES FOR SECTION 15.0

1. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," November 1973 (NEDO-10959 and NEDE-10958).

2. U.S. Nuclear Regulatory Commission Computer Code Tact IIIS, Computer Code for Calculating Radiological Consequences of Time Varying Radioactive Releases, February 1975, Accident Analysis Branch, personal communication.

3. “General Electric Company Model for Loss-of-Coolant Analysis in Accordance with <10 CFR 50, Appendix K>,” December 1975 (NEDO-20566).

4. ASME Boiler and Pressure Vessel Code, Section III, Class 1, “Nuclear Power Plant Components,” Article NB-7000, “Protection Against Overpressure.”

5. General Electric Company “General Electric Standard Application for Reactor Fuel” including the “United States Supplement,” NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision).

6. U.S. Nuclear Regulatory Commission Computer Code TACT 5, Computer Code for Calculating Radiological Consequences of Time Varying Radioactive Releases, <NUREG/CR-5106>, June 1988.

7. General Electric Company “Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor” NEDO-31400A, Oct. 1992.

8. U.S. NRC Computer Code GASPAR II, Computer Code to Perform Environmental Dose Analyses for Release of Radioactive Effluents. <NUREG/CR-4653>, March 1987.

9. (Deleted)

10. GE Nuclear Energy, “Generic Guidelines for General Electric Boiling Water Reactor Power Uprate,” Licensing Topical Report NEDC-31897P-A, Class III (Proprietary), May 1992.

11. Federal Guidance Report 11, "Licensing Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", 2nd Printing, 1989.

12. CCC-652 Oak Ridge National Laboratory RSICC Computer Code Collection MACCS2, V.1.12 Code Package, 1997.

13. Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil", 1993.

TABLE 15.0-1

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS[8]

		Baseline	3,758 MWt
1.	Thermal Power Level, MWt		
	Warranted Value	3,579	-
	Analysis Value	3,729	3,758
2.	Steam Flow, lbs per hr		
	Warranted Value	15.40×10^6	-
	Analysis Value (nominal)[1][3][4]	16.17×10^6	16.30×10^6
3.	Core Flow, lbs per hr	104×10^6	109.2×10^6 [9]
4.	Feedwater Flow Rate, lb per sec		
	Warranted Value (NBR)	4,269	-
	Analysis Value (nominal)[1]	4,483	4,528
5.	Feedwater Temperature, °F	425	425.5
6.	Vessel Dome Pressure, psig	1,045	1,025
7.	Vessel Core Pressure, psig	1,056	1,041
8.	Turbine Bypass Capacity, % NBR	35[7]	23.63
9.	Core Coolant Inlet Enthalpy,		
	Btu per lb[4]	529.9	-
	Btu per lb[3]	528.9	528.5[3]
10.	Turbine Inlet Pressure, psig	960	967
11.	Fuel Lattice	P8x8R	GE10 GE11 GE12
12.	Core Average Gap Conductance,		
	Btu/sec-ft2-°F[4]	0.1546	-
	Btu/sec-ft2-°F[3]	0.1892	.4257[3]
13.	Core Leakage Flow, %[4]	12.9	-
	%[3]	11.0	15.4[3]
14.	Required MCPR Operating Limit	1.18	1.29
15.	MCPR Safety Limit for Incidents of Moderate Frequency	1.06	1.09

TABLE 15.0-1 (Continued)

		Baseline	3,758 MWt
16.	Doppler Coefficient (-)¢/°F		
	Analysis Data(4)(5)	0.132	.145(3)(10)
17.	Void Coefficient (-)¢/% Rated Voids		
	Analysis Data for Power Increase Events(4)(5)	14.0	11.12
	Analysis Data for Power Decrease Events(4)(5)	4.0	-
18.	Core Average Rated Void Fraction, %(4)(5)	42.54	44.0
19.	Scram Reactivity, $ΔK		
	Analysis Data(4)(5)	<Figure 15.0-2>	Same
20.	Control Rod Drive Speed, Position versus time	<Figure 15.0-3>	Same
21.	Nuclear Characteristics used in ODYN Simulations	EOEC(6)	EOC
22.	Jet Pump Ratio, M	2.257	2.257
23.	Safety/Relief Valve Capacity, % NBR		
	@ 1,210 psig(4)	111.4	-
	@ 1,210 psig(3)	110.8	98.0(11)
	Manufacturer	Dikker	Dikker
	Quantity Installed	19	19
24.	Relief Function Delay, seconds	0.4	0.4
25.	Relief Function Response		
	a. Time Constant, seconds	0.1	-
	b. Stroke Time, seconds	-	0.15
26.	Safety Function Delay, seconds	0.0	0.0
27.	Safety Function Response		
	a. Time Constant, seconds	0.2	-
	b. Stroke Time, seconds	-	0.30

TABLE 15.0-1 (Continued)

		Baseline	3,758 MWt
28.	Setpoints for Safety/Relief Valves		
	Safety Function, psig	1,175, 1,185, 1,195, 1,205, 1,215,	1,200, 1,216, 1,226 1,153, 1,143, 1,133
	Relief Function, psig	1,125, 1,135, 1,145, 1,155	
29.	Number of Valve Groupings Simulated		
	Safety Function, No.	5	3
	Relief Function, No.	4	3
30.	Safety/Relief Valve Reclosure Setpoint both Modes (% of setpoint)		
	Maximum Safety Limit (used in analysis)	98	97
	Minimum Operational Limit	89	-
31.	High Flux Trip, % NBR		
	Analysis setpoint (122 x 1.042), % NBR	127.2	122.0
32.	High Pressure Scram Setpoint, psig	1,095	1,095
33.	Vessel Level Trips, Feet Above Separator Skirt Bottom		
	Level 8 - (L8), feet	5.89	5.94
	Level 4 - (L4), feet	4.04	3.87
	Level 3 - (L3), feet	2.165	2.08
	Level 2 - (L2), feet	(-) 1.739	-2.39
34.	APRM Simulated Thermal Power Trip Scram, % NBR		
	Setpoint % NBR	118.8	115.0
	Time Constant, sec	7	6.6
35.	RPT Delay, seconds	0.14	0.14
36.	RPT Inertia Time Constant for Analysis(2),		
	Maximum - sec	5	6
	Minimum - sec	3	-
37.	Total Steamline Volume, ft^3	3,850	4,681(12)

TABLE 15.0-1 (Continued)

	Baseline	3,758 MWt
38. Set Pressure of Anticipated Transient Pump Trip - psig (nominal)	1,135	1,113

NOTES:

(1) Actual analysis value is within ±0.2%.
(2) The inertia time constant is defined by the expression:

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

where

t = inertia time constant (sec).

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

n = rated pump speed (rps).

g = gravitational constant (ft/sec^2).

T_o = pump shaft torque (lb-ft).

(3) Used only for ODYN.

(4) Used only for REDY.

(5) For transients simulated on the ODYN computer model, this input is calculated by ODYN and shown in the plot for each simulated transient.

(6) EOEC - End of equilibrium cycle.

(7) See <Table 15.0-2a> and <Section 10.2.1>.

(8) These input parameters and initial conditions for transients pertain to those transients discussed within the main text of Chapter 15. The Chapter 15 Appendices for the current cycle reload safety analysis <Appendix 15B> and those to support extended operating modes and operation with equipment out-of-service <Appendix 15D>, <Appendix 15E> and <Appendix 15F> provide input parameters and initial conditions for their specific operating regimes.

[9] All but ELL transients were run at ICF conditions. ICF core flow is 105% of rated which is 1.05 x 104E6 = 109.2 x 10E6

[10] ICF conditions – 100%P / 105%F

[11] Capacity at OPL-3 reference pressure of 1080 psig.

[12] Total steamline volume from vessel to TSV not including bypass leg.

TABLE 15.0-2a

SUMMARY OF EVENTS

Section No.	Figure No.	Description	Maximum Neutron Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Bottom Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	ΔCPR[4] -	Frequency Category[1]	Duration of Blowdown: No. of Valves Initially Actuated[5]	Duration of Blowdown: Duration of Blowdown (sec)
15.1		DECREASE IN REACTOR COOLANT TEMPERATURE									
15.1.1	15.1-1	Loss of Feedwater Heating, AFC	111.5	1,045	1,087	1,034	105.8	See Note[2]	a	0	0
15.1.1	15.1-2	Loss of Feedwater Heating, MFC [9]	124.2	1,060	1,102	1,047	113.7	0.12[7][8]	a	0	0
15.1.2	15.1-3	Feedwater Control Failure, Max Demand [10]	124.3	1,163	1,193	1,159	105	0.10[5][7][8]	a	19	5
15.1.3	15.1-4	Pressure Regulator Failure -Open, 130% Flow	104.2	1,138	1,161	1,136	100	[2]	a	10	5
15.2		INCREASE IN REACTOR PRESSURE									
15.2.1	15.2-1	Pressure Regulator Downscale Failure [11]	156.8	1,187	1,221	1,181	102.6	0.09[8]	a	19	7
15.2.2	15.2-2	Generator Load Rejection, Bypass-On	128.2	1,160	1,189	1,157	100	0.05	a	19	5
15.2.2	15.2-3	Generator Load Rejection, Bypass-Off[12]	198.7	1,203	1,233	1,202	102.7	0.08[7][8]	b	19	7
15.2.3	15.2-4	Turbine Trip, Bypass-On	114.5	1,158	1,188	1,155	100	0.05	a	19	5

TABLE 15.0-2a (Continued)

Section No.	Figure No.	Description	Maximum Neutron Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Bottom Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	ΔCPR[4] -	Frequency Category[1]	Duration of Blowdown: No. of Valves Initially Actuated[5]	Duration of Blowdown: Duration of Blowdown (sec)
15.2.3	15.2-5	Turbine Trip, Bypass-Off [13]	179.4	1,202	1,231	1,201	101.3	0.05	b	19	7
15.2.4	15.2-6	Main Steam Line Isolation, Position Scram [14]	105.3	1,177	1,207	1,174	100	See Note[2]	a	19	5
15.2.5	15.2-7	Loss of Condenser Vacuum at 2 inches per sec	120.0	1,157	1,186	1,153	100	See Note[2]	a	19	5
15.2.6	15.2-8	Loss of Auxiliary Power Transformer	104.2	1,175	1,190	1,173	100	See Note[2]	a	1	5
15.2.6	15.2-9	Loss of All Grid Connections	111.0	1,159	1,184	1,156	100	See Note[2]	a	19	7
15.2.7	15.2-10	Loss of All Feed-water Flow	104.2	1,045	1,086	1,034	100	See Note[2]	a	0	0
15.3		DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE									
15.3.1	15.3-1	Trip of One Recirculation Pump Motor	104.3	1,046	1,087	1,035	100	See Note[2]	a	0	0
15.3.1	15.3-2	Trip of Both Recirculation Pump Motors	104.2	1,141	1,155	1,139	100	See Note[2]	a	10	5
15.3.2	15.3-3	Fast Closure of One Recirc. Valve - 60%/sec	104.2	1,049	1,087	1,037	100	See Note[2]	a	10	5

TABLE 15.0-2a (Continued)

Section No.	Figure No.	Description	Maximum Neutron Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Bottom Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	ΔCPR[4] -	Frequency Category[1]	Duration of Blowdown: No. of Valves Initially Actuated[5]	Duration of Blowdown: Duration of Blowdown (sec)
15.3.2	15.3-4	Fast Closure of Two Main Recirc. Valves 11%/sec	104.2	1,142	1,151	1,139	100	See Note[2]	a	10	5
15.3.3	15.3-5	Seizure of One Recirculation Pump	104.2	1,139	1,153	1,137	100	See Note[2]	c	10	5
15.4		REACTIVITY AND POWER DISTRIBUTION ANOMALIES									
15.4.4	15.4-1	Startup of Idle Recirculation Loop	100.0	988	1,002	983	148.7	See Note[3]	a	0	0
15.4.5	15.4-2	Fast Opening of One Recirculation Valve	215.0	978	993	974	135	See Note[3]	a	0	0
15.4.5	15.4-3	Fast Opening of Two Recirc. Valves - 11%/sec	149.0	974	990	971	123.4	See Note[3]	a	0	0
15.5		INCREASE IN REACTOR COOLANT INVENTORY									
15.5.1	15.5-1	Inadvertent HPCS Pump Start	104.2	1,045	1,087	1,034	100	See Note[2]	a	0	0

NOTES:

(1) a = incidents of moderate freq; b = infrequent incidents; c = limiting faults
(2) No significant change in CPR.
(3) Not start from full power.
(4) Option A ΔCPR adjustment factor is included as specified in the NRC staff safety evaluation for the General Electric Topical Report - Qualification of the One-Dimensional Core Transient Model for BWR, NEDO-24154 and NEDE-24154-P is applicable.
(5) Expected number of SRV actuations based on analytical prediction.
(6) Analysis has been performed to conclude that turbine bypass capacity as low as 25% NBR does not affect the bounding ΔCPR results. See Reference <Appendix 15D.11-6>.

TABLE 15.0-2a (Continued)

SUMMARY OF EVENTS

NOTES:

[7] This transient was performed as part of the Partial Feedwater Heating, Maximum Extended Operating Domain and/or Single Loop Operation Analyses. For the initial conditions, required operating states and results of these analyses refer to <Appendix 15D>, <Appendix 15E>, and <Appendix 15F> respectively.

[8] This transient is reperformed as part of the current cycle reload safety analysis. <Appendix 15B> presents the results of the Reload Safety Analysis. These results supersede previous transient analyses <Chapter 15> and Note 7 performed at the same power, flow, feedwater temperature, and cycle exposure conditions.

[9] The loss of Feedwater Heater event is re-analyzed at 3,758 MWt core power. The event is documented in <Section 15.1.1> and <Appendix 15D>.

[10] The Feedwater Controller Failure event is re-analyzed at 3,758 MWt core power with feedwater temperature reduction. The re-analysis is discussed in <Appendix 15D>.

[11] The Pressure Regulator Downscale Failure event is re-analyzed at 3,758 MWt core power with both normal and reduced feedwater temperature. The event with normal feedwater temperature is documented in <Table 15.0-2b>. The event with reduced feedwater temperature is bounded by the event with normal feedwater temperature.

[12] The Generator Load Rejection event is re-analyzed at 3,758 MWt core power with both normal and reduced feedwater temperature. The event with normal feedwater temperature is documented in <Table 15.0-2b>. The event with reduced feedwater temperature is documented in <Appendix 15D>.

[13] The Turbine Trip event is re-analyzed at 3,758 MWt core power with normal feedwater temperature. The event is documented in <Table 15.0-2b>.

[14] The Main Steamline Isolation Valve Closure – Flux Scram event is re-analyzed at 3,758 MWt core power with normal feedwater temperature but at 102% power. The event is documented in <Table 15.0-2b>.

TABLE 15.0-2b

SUMMARY OF EVENTS ANALYZED AT POWER UPRATE CONDITIONS

Analysis ID	Analysis Name	Transient Output File Names (.*)	ODYN PID	SUB EVENTS		Power %	Flow %	Steam Flow %
CYC8A01	ICF/HBB	006EC_E00000_T02_ODYNV09_LRNBP	0BC2F	2RVOS		100.0	105.0	100.0
CYC8A01	ICF/HBB	006EC_E00000_T03_ODYNV09_TTNBP	04255	2RVOS		100.0	105.0	100.0
CYC8A01	ICF/HBB	006EC_E00000_T04_ODYNV09_PRFDS	01D5A	2RVOS		100.0	105.0	100.0
CYC8A01	ICF/HBB	006EC_E00000_T05_ODYNV09_MSIVF	02FE8	OPP	6SVOS	102.0	105.0	102.3

TRANSIENT NAME	ODYN PID	ANALYSIS ID	EXPOSURE Mwd/st	PEAK FLUX (N) % ref	PEAK FLUX (Q/A) % init	MAX INTGR QFUEL PU	MAX NET REACT $	DCPR G1246I	DCPR G1136	DCPR G1036
LRNBP	0BC2F	CYC8A01	E00000	291.74	109.00	0.32	0.67	0.1837	0.1382	0.0766
TTNBP	04255	CYC8A01	E00000	250.43	106.52	0.26	0.63	0.1706	0.1260	0.0622
PRFDS	01D5A	CYC8A01	E00000	139.11	104.39	0.00	0.25	0.0771	0.0560	0.0424

TRANSIENT	ODYN	ANALYSIS	EXPOSURE	G1246I		G1136		G1036	
NAME	PID	ID	Mwd/st	DCPRB	DCPRA	DCPRB	DCPRA	DCPRB	DCPRA
LRNBP	0BC2F	CYC8A01	E00000		0.1991		0.1524		0.0828
TTNBP	04255	CYC8A01	E00000		0.1848		0.1400		0.0682
PRFDS	01D5A	CYC8A01	E00000		0.0894		0.0681		0.0513

TABLE 15.0-2b (Continued)

TRANSIENT NAME	ODYN PID	ANALYSIS ID	EXPOSURE Mwd/st	PEAK FLUX Q/A % init	PEAK DOME PRESSURE RATE psi/sec	PEAK PRESSURE DOME psig	PEAK PRESSURE P(V) psig	PEAK PRESSURE P(SL) psig	MIN DELTA P(UCL) psi	MIN DELTA P(SSV) psi	MIN DELTA P(ECL) psi
LRNBP	0BC2F	CYC8A01	E00000	109.00	232.3	1199.4	1228.7	1191.6	146.3		271.3
TTNBP	04255	CYC8A01	E00000	106.52	236.9	1198.7	1228.0	1190.7	147.0		272.0
PRFDS	01D5A	CYC8A01	E00000	104.39	100.4	1186.5	1219.3	1180.3	155.7		280.7
MSIVF	02FE8	CYC8A01	E00000	120.10	222.9	1267.7	1295.3	1264.3	79.7		204.7

TABLE 15.0-3

SUMMARY OF ACCIDENTS

Section	Title	Failed Fuel Pins: GE Calculated Value	Failed Fuel Pins: NRC Worst Case Assumption
<Section 15.3.3>	Seizure of One Recirculation Pump	None	
<Section 15.3.4>	Recirculation Pump Shaft Break	None	
<Section 15.4.9>	Control Rod Drop Accident	<1200	770
<Section 15.6.2>	Instrument Line Break	None	None
<Section 15.6.4>	Steam System Pipe Break Outside Containment	None	None
<Section 15.6.5>	LOCA Within RCPB	None	100%
<Section 15.6.6>	Feedwater Line Break	None	None
<Section 15.7.1.1>	Main Condenser Offgas Treatment System Failure	N/A	N/A
<Section 15.7.3>	Liquid Radwaste Tank Failure	N/A	N/A
<Section 15.7.5>	Spent Fuel Cask Drop Accident	None	None
<Section 15.7.6>	Fuel Handling Accident Inside Containment (GE12 and GE14 fuel/GNF2 fuel) w/triangular mast	151/150	
<Section 15.8>	ATWS	SPECIAL EVENT	

TABLE 15.0-4

DOSE CONVERSION FACTORS[1]

Isotope	Thyroid (rem/Ci)	Whole Body 0.25xMeV/dis
I-131	1.49E+6	8.72E-2
I-132	5.35E+4	5.13E-1
I-133	3.97E+5	1.55E-1
I-134	2.54E+4	5.32E-1
I-135	1.24E+5	4.21E-1
Kr-83m		5.02E-6
Kr-85		3.72E-2
Kr-85m		5.25E-4
Kr-87		1.87E-1
Kr-88		4.64E-1
Kr-89		5.25E-1
Xe-131m		2.92E-3
Xe-133m		8.00E-3
Xe-133		9.33E-3
Xe-135m		9.92E-2
Xe-135		5.72E-2
Xe-137		4.53E-2
Xe-138		2.81E-1

Breathing Rates

Time Period (hr)	Breathing Rate m^3/sec	
	MSLB	FHA, LOCA, CRDA
0-8	3.5E-4[2]	3.5E-4[2]
8-24	1.75E-4	1.8E-4
24-720	2.32E-4	2.3E-4

NOTES:

(1) The following dose conversion factors (DCF's) are used in the alternative source term analyses;

FHA, CRDA, MSLB - EPA Federal Guidance Report 11 - 1989 (Reference 11) and EPA Federal Guidance Report 12-1993 (Reference 13).

LOCA - CEDE: EPA Federal Guidance Report 11 – 1989 (Reference 11)
DDE/EDE: MACCS2 computer code (Reference 12), which used Federal Guidance Report 12 – 1993 (Reference 13).

(2) This breathing rate was used for the duration of the Control Room radiological consequence analyses.

TABLE 15.0-5

SUMMARY OF LIMITING TRANSIENT ANALYSES FOR
EXTENDED OPERATING MODES (MEOD, PFH AND SLO)
OR REPERFORMED FOR EACH RELOAD ANALYSIS

Transient/Accident Analysis	USAR Section	Also Analyzed in these USAR Chapter 15 Appendices			
		MEOD 15E	PFH 15D	SLO 15F	RELOAD 15B
Thermal-Hydraulic Stability Analysis	<Section 4.4.4.6>	YES	YES	YES	YES
Overpressurization Protection (MSIV Closure-Flux Scram)	<Section 5.2.2>	YES	NO	NO	YES
ECCS Performance Evaluation (LOCA)	<Section 6.3.3> <Section 15.6.5>	YES	YES	YES	YES
Loss of Feedwater Heating - 100°F (LFWH)	<Section 15.1.1>	YES	YES	NO	YES
Feedwater Controller Failure-Maximum Demand (FWCF)	<Section 15.1.2>	YES	YES	YES	YES
Pressure Regulator Failure (Downscale)	<Section 15.2.1>	YES	NO	NO	YES
Generated Load Rejection with Bypass Failure (LRNBP)	<Section 15.2.2>	YES	YES	YES	YES
Rod Withdrawal Error at Power (RWE)	<Section 15.4.2>	YES	YES	YES	YES
Anticipated Transients Without Scram (ATWS)	<Section 15.8>	YES	YES	YES	NO

15.1 DECREASE IN REACTOR COOLANT TEMPERATURE

15.1.1 LOSS OF FEEDWATER HEATING

This transient was performed as part of the initial cycle analyses supporting PNPP operation in various operating modes and/or with equipment out-of-service, results of which are presented in the following Chapter 15 appendices:

- <Appendix 15D> Partial Feedwater Heating Operation Analysis
- <Appendix 15E> Maximum Extended Operating Domain Analysis

Certain limiting safety analyses are reperformed each operating cycle to determine and/or verify safety margins. The methods, input conditions, and results for the current cycle for the loss of feedwater heating event are presented in <Appendix 15B> of this chapter.

The initial cycle analyses used a point model methodology and the results concluded that, for this event, the automatic control is less severe than in manual control <Figure 15.1-1> and <Figure 15.1-2>. The documentation of this original analyses is preserved in this section.

The re-analysis at 3,758 MWt core power used a 3-dimensional, coupled nuclear-thermal hydraulics core simulator computer model. The methodology and results of this re-analysis are documented in <Appendix 15D>.

15.1.1.1 Identification of Causes and Frequency Classification

15.1.1.1.1 Identification of Causes

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

a. Steam extraction line to heater is closed,

b. Steam is bypassed around heater.

The first case produces a gradual cooling of the feedwater. In the second case, the steam bypasses the heater and no heating of that feedwater occurs. In either case the reactor vessel receives cooler feedwater. The maximum number of feedwater heaters 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 result in a reduction of up to 100°F in core inlet temperature. This increases core power due to the negative void reactivity coefficient. The event can occur with the reactor in either the automatic or manual control mode. In automatic control, some compensation of core power is realized by modulation of core flow, so the event is less severe than in manual control.

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 at full power.

15.1.1.2 Sequence of Events and Systems Operation

15.1.1.2.1 Sequence of Events

<Table 15.1-1> and <Table 15.1-2> list the sequence of events for this transient and its effect on various parameters is shown in <Figure 15.1-1> and <Figure 15.1-2>.

15.1.1.2.1.1 Identification of Operator Actions

In the automatic flux/flow control mode, the reactor settles out at a lower recirculation flow with no change in steam output. An average power range monitor (APRM) neutron flux or thermal power alarm will alert the operator that he should insert control rods to get back down to the rated flow control line, or that he should reduce flow if in the manual mode. Operating procedures describe turbine generator operation with feedwater heaters out-of-service. If reactor scram occurs, as it does in manual flow control mode, the operator must monitor the reactor water level and pressure controls.

15.1.1.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems. This event results in a slow prolonged power increase without flux spikes.

The flow biased thermal power monitor (TPM) is the primary protection system trip in mitigating the consequences of this event.

The TPM conservatively estimates thermal power by passing the APRM signal through a six second time constant (as compared to the actual fuel time constant of seven to ten seconds). A scram is initiated when thermal power exceeds the flow-biased function shown in <Figure 15.1-5>. For a slow transient this limit will be reached before the APRM scram because of its 6 to 8 percent lower setpoint. As can be seen from <Figure 15.1-5>, the flow-biased trip setpoint is equal to 114 percent NBR maximum. Since the transients in this chapter are analyzed at 104.2 percent NBR power, the scram setpoints are 1.042 times higher than shown in <Figure 15.1-5>, or 114% x 1.042 = 118.8% NBR, as shown on <Table 15.0-1>.

The TPM is a safety grade system and is designed to be single failure proof. Surveillance testing of the TPM is included in the technical specifications.

Required operation of Engineered Safety Features (ESF) is not expected for either of these transients.

15.1.1.2.3 The Effect of Single Failures and Operator Errors

These two events generally lead to an increase in reactor power level. The TPM mentioned in <Section 15.1.1.2.2> is the mitigating system and is designed to be single failure proof. See <Appendix 15A> for additional discussion of this subject.

15.1.1.3 Core and System Performance

15.1.1.3.1 Mathematical Model

The predicted dynamic behavior has been determined using a computer model of a generic direct-cycle BWR. This model is described in detail in (Reference 1). This computer model has been verified through extensive comparisons with actual BWR test data. Some of the significant features of the nonlinear 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 multinode core steady-state calculations. A second-order void dynamic model with the void boiling sweep time calculated as a function of core flow and void conditions is also 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.

15.1.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in <Table 15.0-1>. The loss of feedwater heating transient was analyzed for the end of equilibrium cycle. This is the most limiting core condition with highest void coefficient and lowest scram reactivity.

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 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.

15.1.1.3.3 Results

In the automatic flux/flow control mode, the recirculation flow control system responds to the power increase by reducing core flow so that steam flow from the reactor vessel to the turbine remains essentially constant. 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 80 percent of rated flow. The MCPR reached in the automatic control mode is greater than for the more limiting manual flow control mode. This method of control is not currently in use.

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>.

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 the average fuel temperature increases 120°F. The increased core inlet subcooling aids core thermal margins and minimum CPR is 1.07. 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>.

If the reactor scrams, water level drops to the low level trip point (L2). This initiates recirculation pump trip as shown in <Table 15.1-2>.

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

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

As noted above and shown in <Figure 15.1-1> and <Figure 15.1-2>, 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.

15.1.1.5 Radiological Consequences

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

15.1.2 FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND

This transient was performed as part of initial cycle analyses supporting PNPP operation in various operating modes and/or with equipment out-of-service, results of which are presented in the following Chapter 15 appendices:

- <Appendix 15D> Partial Feedwater Heating Operation Analysis
- <Appendix 15E> Maximum Extended Operating Domain Analysis
- <Appendix 15F> Recirculation System Single-Loop Operation Analysis

Certain limiting safety analyses are reperformed each operating cycle to determine and/or verify safety margins. The methods, input conditions, and results for the current cycle for the feedwater controller failure-maximum demand event are presented in <Appendix 15B> of this chapter.

This event was re-analyzed at 3,758 MWt core power conditions shown in <Table 15.0-1>, but with feedwater temperature of 255.5°F The results of the new analysis are reported in <Appendix 15D>.

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

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

15.1.2.2.1.1 Identification of Operator Actions

The operator should:

a. Observe that feedwater pump trip has terminated the failure event.

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

c. Identify causes of the failure and report all key plant parameters during the event.

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. Important system operational actions for this event are high level scram and tripping of the main turbine and feedwater pumps, recirculation pump trip (RPT) 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 model of a generic direct-cycle BWR (Reference 2). This computer model has been improved and verified through extensive comparison of its predicted results with actual BWR test data.

This nonlinear computer 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 eight pressure nodes incorporating mass and momentum balances which will predict any wave phenomena present in the steam line during pressurization transients.

d. The core average axial water density and pressure distribution is calculated using a single channel to represent the heat active flow and a single channel to represent bypass flow. A model, representing liquid and vapor mass and energy conservation, and mixture momentum conservation, describes 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, pressure, and load demand are represented together with their dominant nonlinear characteristics.

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

g. 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) and used in analysis for other transients.

15.1.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in <Table 15.0-1>.

End of equilibrium 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 percent NBR feedwater flow occurs at a system design pressure of 1,065 psig.

15.1.2.3.3 Results

The simulated feedwater controller transient is shown in <Figure 15.1-3>. The high water level turbine trip and feedwater pump trip are initiated at approximately 12 seconds. Scram occurs almost simultaneously, and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. MCPR remains above safety limit. The turbine bypass system opens to limit peak pressure in the steam line near the safety/relief valves to 1,159 psig and the pressure at the bottom of the vessel to about 1,193 psig.

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

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

15.1.2.3.4 Consideration of Uncertainties

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

15.1.2.4 Barrier Performance

As noted above the consequences of this 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 <Section 15.2.4.5>. Therefore, the radiological exposures noted in <Section 15.2.4.5> cover the consequences of this event.

15.1.3 PRESSURE REGULATOR FAILURE - OPEN

This transient was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed transients.

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 and through the turbine bypass valves resulting from a pressure regulator malfunction is limited by a maximum flow limiter imposed at the turbine controls. This limiter is set to limit maximum steam flow to approximately 130 percent NB rated.

If both the controlling pressure regulator and the backup regulator fail to the open position, the turbine admission valves and the turbine bypass valves can be 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 Identification of Operator Actions

When regulator trouble is preceded by spurious or erratic behavior of the controlling device, it may be possible for the operator to transfer to the backup controller in time to prevent the full transient. However, if such efforts are not successful and the event occurs, no operator actions are required to maintain adequate fuel thermal margin. If the depressurization is not stopped by automatic isolation (e.g., if the mode switch is not in RUN), then the operator shall manually control the cooldown so as not to exceed the approved cooldown rate. If this is not possible, the operator shall then terminate the cooldown by manually shutting the MSIVs. Longer term operator safety actions may be required after reactor isolation has occurred. Suppression pool cooling may be needed to maintain suppression pool temperature within its required limits if significant safety relief flow occurs after the isolation.

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 otherwise noted.

Initiation of HPCS and RCIC system functions will occur when the vessel water level reaches the L2 setpoint. 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 depressurization. The course of the event can follow more than one sequence:

a. For the original analysis, the sequence given in <Table 15.1-4> and <Figure 15.1-4> represents the most probable course of the event (especially when initial power is below rated). The initial part of the depressurization is abrupt enough to cause a rapid reactor water level swell and initiation of high water (level 8) reactor scram and trip of the main turbine and feedwater pumps. Tripping of the main turbine will initiate another scram signal and will initiate a signal to open the turbine bypass valves (at this point the valves will go full open and be held full open due to the pressure regulator failure). The turbine trip will cause a temporary pressure increase which will be quickly mitigated by the open turbine bypass valves and SRV actuations.

   All of the high level instrumentation is designed to be single failure proof, but these high level functions are not vital to protection in this event since the level increase is momentary. Termination of the event occurs when the depressurization has been isolated. The <Table 15.1-4> sequence shows the MSIV isolation to automatically occur (with no operator action) when the steam line pressure falls to its initiation setpoint, terminating the potentially rapid depressurization and cooldown. Instruments for sensing low turbine inlet pressure are also designed to be single failure proof for initiation of MSIV closure.

However, since there is a period of time between the high water level trips and the low steam line pressure isolation (about 28 seconds in <Table 15.1-4>), there is a possibility that an operator could react to the high level scram and make a reactor mode switch transfer from RUN to SHUTDOWN. Such rapid action would deactivate the automatic MSIV isolation on low pressure since it is only active in the RUN mode. If this should occur the operator shall control the cooldown rate so as not to exceed 100°F during any one-hour period. If this is not possible the operator should terminate the cooldown by manually shutting the MSIVs.

b. It is possible that the course of the event may not reach the high water level trips discussed above (Reference 3). Then the depressurization proceeds more simply to the low pressure MSIV isolation. Scram is directly associated with the valve closure, and no operator mode switch interaction is anticipated before the MSIV closure.

Reactor scram sensing, whether originating from high reactor water level or the limit switches on the main steam line isolation valves, is designed to be single failure proof.

See <Appendix 15A>, Event 23, 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 <Section 15.1.1.3.1> is used to simulate this event.

15.1.3.3.2 Input Parameters and Initial Conditions

This transient is simulated by setting the regulator output to a high value, which causes the turbine admission valves and the turbine bypass valves to open. A regulator failure with 130 percent steam flow was simulated as a worst case.

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 setpoint for main steamline isolation initiation. This is within the specification limits of the valve and represents a conservative assumption.

The primary sequence shown in <Table 15.1-4> includes high reactor water level (level 8) scram and trip of the main turbine and feedwater pumps shortly after the rapid depressurization begins (at about 3 seconds into the event). The upper analytical limits for the high water level setpoints are used. Isolation follows significantly later (after about 28 seconds) when continued flow through the turbine bypass valves re-establishes depressurization.

If the high level trips are not reached, then a reactor scram would be initiated when the main steamline isolation valves reach the 10 percent closed position (closure initiated when pressure reaches the setpoint for the main steam line low pressure isolation). 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 15.0-1>. Perry did not model the scenario where reactor scram occurs at the steam line low pressure isolation setpoint.

15.1.3.3.3 Results

<Figure 15.1-4> shows graphically that the high water level trip and isolation valve closure stop vessel depressurization and produce a normal shutdown of the isolated reactor.

The rapid depressurization results in formation of steam voids in the reactor coolant and causes a rapid decrease in reactor power almost immediately. The steam void formation also produces a water level increase that is shown to reach the high water level trip setpoint (level 8) about 3 seconds into the event. This produces a reactor scram and trip of the main turbine and feedwater pumps. The turbine valve closure causes a brief pressurization, lifting safety/relief valves, but they reclose quickly as the depressurization continues due to the pressure control failure that holds the turbine bypass valves wide open. The main steam line isolation valves automatically close at approximately 28 seconds when pressure at the turbine decreases below 783 psig. 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 transient results in only momentary depressurization of the nuclear system, cooldown stresses of the components remain within design allowances for upset category events. The safety/relief valves need only to relieve the pressure increase caused by the turbine trip (initiated after power has decreased due to the initial depressurization), and provide long term decay heat removal following isolation. The reactor coolant pressure boundary is not threatened by high internal pressure.

15.1.3.3.4 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. For example, the turbine valves will open to the valves-wide-open state admitting slightly more than the rated steam flow, and with the limiter in this analysis set to fail at 130 percent it is expected that something less than 23 percent flow would be bypassed. This is not a limiting factor on this plant.

Depressurization at slower rates may also be terminated by other protective actions, e.g., the low pressure MSIV isolation.

15.1.3.4 Barrier Performance

Barrier performance analyses were not required since the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which fuel, pressure vessel or containment are designed. Peak pressure in the bottom of the vessel reaches 1,162 psig which is below the ASME code limit of 1,375 psig for the reactor coolant pressure boundary. Minimum vessel dome pressure of 790 psig occurs at about 30 seconds.

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 <Section 15.2.4.5>. Therefore, the radiological exposures noted in <Section 15.2.4.5> cover the consequences of this event.

15.1.4 INADVERTENT SAFETY/RELIEF VALVE OPENING

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. It is therefore simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve design is provided in <Chapter 5>.

This transient is similar to the incident of a safety/relief valve sticking open. This is the only operational transient that requires operator action to attempt to reclose the valve or shut down the plant when suppression pool temperature exceeds the technical specification limit.

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 basis, 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 this event.

15.1.4.2.1.1 Identification of Operator Actions

Control room alarms from the safety/relief valve open/close monitor, or from the suppression pool temperature monitor, will provide the operator pertinent information for his action. The plant operator must reclose the valve as soon as possible and check that reactor and T-G output return to normal. If the valve cannot be closed, plant shutdown should be initiated. The elapsed time the operator has depends on the temperature of the suppression pool water at the onset of the event. However, the operator is required to scram the reactor when the suppression pool temperature reaches the technical specification limit.

15.1.4.2.2 Systems Operation

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

15.1.4.2.3 The Effect of Single Failures and Operator Errors

Failure of additional components (e.g., pressure regulator, feedwater flow controller) is discussed elsewhere in <Chapter 15>. In addition, a detailed discussion of such effects is given in <Appendix 15A>.

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

15.1.4.3.1 Mathematical Model

The reactor model briefly described in <Section 15.1.1.3.1> was previously used to simulate this event in earlier FSARs. This model is discussed in detail in (Reference 2). 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 102 percent of rated core power conditions when a safety/relief valve is inadvertently opened. Manual recirculation flow control is assumed. Flow through the relief valve at normal plant operating conditions stated above is approximately 7 percent of rated steam flow.

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

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

15.1.6.1 Identification of Causes and Frequency Classification

15.1.6.1.1 Identification of Causes

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

In startup or cooldown operation, if the reactor were critical or near critical, a very slow increase in reactor power could result. A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for the RHR heat exchangers. The resulting temperature decrease would cause a slow 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 RHR heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. Scram will occur before any thermal limits are reached if the operator does not take action. The sequence of events for this event is shown in <Table 15.1-6>.

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 RHR shutdown cooling system.

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 <Appendix 15A>.

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 FOR SECTION 15.1

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

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

3. Calculation FM-085, "Support for SC 05-03/License Amendment Request 14-007"

TABLE 15.1-1

SEQUENCE OF EVENTS FOR <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 AFC system automatically reduces core flow to maintain initial steam flow.
100	Reactor variables settle into new steady-state.

TABLE 15.1-2

SEQUENCE OF EVENTS FOR <FIGURE 15.1-2>

Time-sec	Event
0	Initiate a 100°F temperature reduction into the feedwater system.
5	Initial effect of unheated feedwater starts to raise core power level and steam flow.
7	Turbine control valves start to open to regulate pressure.
36	APRM initiates reactor scram on high thermal power.
53	Wide Range (WR) sensed water level reaches Level 2 (L2) setpoint.
53	Recirculation pump trip initiated due to Level 2 trip.
>80 (est.)	HPCS/RCIC flow enters vessel (not simulated).
>90 (est.)	Reactor variables settle into limit cycle.

TABLE 15.1-3

SEQUENCE OF EVENTS FOR <FIGURE 15.1-3>

Time-sec	Event
0	Initiate simulated failure of 130% upper limit at system design pressure of 1,065 psig on feedwater flow.
11.8	L8 vessel level setpoint initiates reactor scram and trips main turbine and feedwater pumps.
11.9	Recirculation pump trip (RPT) actuated by stop valve position switches.
11.9	Main turbine bypass valves opened due to turbine trip.
13.2	Safety/relief valves open due to high pressure.
18.2	Safety/relief valves close.
>20 (est.)	Water level dropped to low water level setpoint (L2).
>50 (est.)	RCIC and HPCS flow into vessel (not simulated).

TABLE 15.1-4

SEQUENCE OF EVENTS FOR <FIGURE 15.1-4>

Time-sec	Event
0	Simulate maximum limit on steam flow to main turbine.
2.1	Turbine control valves wide open.
2.28	Vessel water level (L8) trip initiates reactor scram and main turbine and feedwater pump turbine trips.
2.28	Turbine trip initiates turbine bypass valve operation to full flow.
2.29	Main turbine stop valves reach 90% open position and initiate recirculation pump trip (RPT) to low speed.
2.38	Turbine stop valves closed. Turbine bypass valves opening to full flow.
2.4	Recirculation pump motor circuit breakers open causing decrease in core flow to low speed operation.
5.2	Group 1 pressure relief valves actuated.
9.0	Group 1 pressure relief valves close (turbine bypass valves still open).
25	Vessel water level reaches Level 2 (L2) setpoint. HPCS and RCIC logic initiated. Complete trip of recirculation pumps.
28	Main steam line isolation on low turbine inlet pressure (783 psig).
33	MSIVs closed. Bypass valves remain open, exhausting steam in steamlines downstream of MSIVs.
55 (est.)	RCIC and HPCS systems flow into vessel (not simulated).

TABLE 15.1-5

SEQUENCE OF EVENTS FOR INADVERTENT SAFETY/RELIEF VALVE OPENING

Time-sec	Event
0	Initiate opening of 1 safety/relief valve.
0.5 (est.)	Relief flow reaches full flow.
15 (est.)	System establishes new steady-state operation.

TABLE 15.1-6

SEQUENCE OF EVENTS FOR INADVERTENT RHR SHUTDOWN COOLING OPERATION

Approximate Elapsed Time	Event
0	Reactor at states B or D <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.

15.2 INCREASE IN REACTOR PRESSURE

15.2.1 PRESSURE REGULATOR FAILURE - CLOSED

This transient was performed as part of initial cycle analyses supporting PNPP operation in various operating modes and/or with equipment out-of-service, results of which are presented in <Appendix 15E> - Maximum Extended Operating Domain Analysis.

Certain limiting safety analyses are reperformed each operating cycle to determine and/or verify safety margins. The methods, input conditions, and results for the current cycle for the pressure regulator failure-closed event are presented in <Appendix 15B> of this chapter.

This event was re-analyzed at 3,758 MWt core power conditions shown in <Table 15.0-1>. The results of the new analysis are reported in this section.

15.2.1.1 Identification of Causes and Frequency Classification

15.2.1.1.1 Identification of Causes

Two identical pressure regulators are provided to maintain primary system pressure control. They independently sense pressure just upstream of the main turbine stop valves and compare it to two separate setpoints to create proportional error signals that produce each regulator output. One regulator is selected to be the controlling regulator for the main turbine control valves.

It is assumed for purposes of this transient analysis that a single failure occurs which erroneously causes the controlling regulator to close the main turbine control valves and thereby increase reactor pressure. If this occurs, the backup regulator is ready to take control.

It is also assumed, for purposes of this transient analysis, that a single failure occurs which causes a downscale failure of the pressure regulator demand to zero. Should this occur, it could cause full closure of turbine control valves as well as an inhibit of steam bypass flow and thereby increase reactor power and pressure. When this occurs, reactor scram will be initiated when the high neutron flux scram setpoint is reached.

15.2.1.1.2 Frequency Classification

a. One Pressure Regulator Failure - Closed

This event is treated as a moderate frequency event.

b. Pressure Regulator Downscale Failure

This event is treated as a moderate frequency event.

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

15.2.1.2.1 Sequence of Events

a. One Pressure Regulator Failure - Closed

A failure of the primary or controlling pressure regulator in the closed mode as discussed in <Section 15.2.1.1.1> will cause the turbine control valves to close momentarily. The pressure will increase, because the reactor is still generating initial steam flow. The backup regulator will reopen the valves and re-establish steady-state operation.

b. Pressure Regulator Downscale Failure

<Table 15.2-1> lists the sequence of events for <Figure 15.2-1>.

15.2.1.2.2 Identification of Operator Actions

a. One Pressure Regulator Failure - Closed

The operator should verify that the backup regulator assumes proper control. However, these actions are not required to terminate the event as discussed in <Section 15.2.1.2.4.a>.

b. Pressure Regulator Downscale Failure

The operator should:

1. Monitor that all rods are in.

2. Monitor reactor water level and pressure.

3. Observe that the safety/relief valves open at their setpoint.

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

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

15.2.1.2.3 Systems Operation

a. One Pressure Regulator Failure - Closed

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

b. Pressure Regulator 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. High system pressure is limited by safety/relief valve operation.

15.2.1.2.4 The Effect of Single Failures and Operator Errors

a. One Pressure Regulator Failure - Closed

The nature of the first assumed failure produces a slight pressure increase in the reactor until the backup regulator gains control, since no other action is significant in restoring normal operation. If the backup regulator fails at this time, the control valves will start to close causing reactor pressure to increase, and a flux scram trip would be initiated to shut down the reactor. This event is similar to a pressure regulator failure closed. Detailed discussions on this subject can be found in <Appendix 15A>.

b. Pressure Regulator 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 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 nonlinear, dynamic model (ODYN) described briefly in <Section 15.1.2.3.1> is used to simulate this 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 15.0-1>.

15.2.1.3.3 Results

a. One Pressure Regulator Failure - Closed

   Pressure at the turbine inlet increases quickly (less than approximately 2 seconds) due to the sharp closing action of the turbine control valves which reopen when the backup regulator gains control. This pressure disturbance in the vessel is not expected to exceed flux or pressure scram trip setpoints.

b. Pressure Regulator Downscale Failure

   A pressure regulation downscale failure is simulated at 100 percent NB rated steam flow condition in <Figure 15.2-1>.

   Neutron flux increases rapidly because of the void reduction caused by the pressure increase. When the sensed neutron flux reaches the high neutron flux scram setpoint, a reactor scram is initiated. The neutron flux increase is limited to 139 percent NB rated by the reactor scram. Peak fuel surface heat flux does not exceed

104.4 percent of its initial value. MCPR for this transient is still above the safety MCPR limit. Therefore, the design basis is satisfied.

15.2.1.3.4 Consideration of Uncertainties

All systems utilized for protection in this event were assumed to have the most conservative allowable characteristic (e.g., relief setpoints, scram stroke time and worth 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 Regulator 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 and containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.1.4.2 Pressure Regulator Downscale Failure

Peak pressure at the safety/relief valves reaches 1,180 psig. The peak nuclear system pressure reaches 1,219 psig at the bottom of the vessel, well below the reactor coolant system transient pressure limit of 1,375 psig.

15.2.1.5 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

<Section 15.2.4.5>. Therefore, the radiological exposures noted in <Section 15.2.4.5> cover the consequences of this event.

15.2.2 GENERATOR LOAD REJECTION

The generator load rejection with bypass failure transient was performed as part of initial cycle analyses supporting PNPP operation in various operating modes and/or with equipment out-of-service, results of which are presented in the following Chapter 15 appendices:

- <Appendix 15D> Partial Feedwater Heating Operation Analysis
- <Appendix 15E> Maximum Extended Operating Domain Analysis
- <Appendix 15F> Recirculation System Single-Loop Operation Analysis

This event was re-analyzed at 3,758 MWt core power conditions shown in <Table 15.0-1>. The results of the new analysis are reported in this section.

Certain limiting safety analyses are reperformed each operating cycle to determine and/or verify safety margins. The methods, input conditions, and results for the current cycle for the generator load rejection with bypass failure event are presented in <Appendix 15B> of this chapter.

15.2.2.1 Identification of Causes and Frequency Classification

15.2.2.1.1 Identification of Causes

Fast closure of the turbine control valves (TCV) is initiated whenever electrical grid disturbances occur which result 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 turbine generator rotor. Closure of the main turbine control valves initiates a scram signal and will cause a sudden reduction in steam flow which results in an increase in system pressure.

15.2.2.1.2 Frequency Classification

a. Generator Load Rejection

This event is categorized as an incident of moderate frequency.

b. Generator Load Rejection with Bypass Failure

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

Frequency:	0.0036/plant year
Mean time between events (MTBE):	278 years

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.

However, PNPP is committed to compare the consequences of this transient with the allowable MCPR of a moderate frequency event.

15.2.2.2 Sequence of Events and Systems Operation

15.2.2.2.1 Sequence of Events

a. Generator Load Rejection - Turbine Control Valve Fast Closure

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

b. Generator Load Rejection with Failure of Bypass

A loss of generator electrical load at high power with bypass failure produces the sequence of events listed in <Table 15.2-3>.

15.2.2.2.1.1 Identification of Operator Actions

The operator should:

a. Verify proper bypass valve performance.

b. Observe that the feedwater/level controls have maintained the reactor water level at a satisfactory value.

c. Observe that the pressure regulator is controlling reactor pressure at the desired value.

d. Record peak power and pressure.

e. Verify relief valve operation.

15.2.2.2.2 Systems Operation

a. Generator Load Rejection with Bypass

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 unless stated otherwise.

Turbine control valve (TCV) fast closure initiates a scram trip signal for power levels greater than 38 percent NB rated. In addition, recirculation pump trip (RPT) is initiated. Both of these trip signals satisfy single failure criteria 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 setpoints is assumed to function normally during the time period analyzed.

b. Generator Load Rejection with Failure of Bypass

The sequence of events for this failure is the same as above except that failure of the main turbine bypass valves is assumed for the entire transient.

15.2.2.2.3 The Effect of Single Failures and Operator Errors

Mitigation of pressure increase is accomplished by the reactor protection system functions. Turbine control valve trip scram and RPT are designed to satisfy the single failure criteria. 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 computer model described in <Section 15.1.2.3.1> was used to simulate this event.

15.2.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in <Table 15.0-1>.

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

The closure characteristics of the turbine control valves are assumed such that the valves operate in the full arc (FA) mode and have a full stroke closure time, from fully open to fully closed, of 0.15 seconds. This is consistent with the design specification limit <Section 10.2>. Full arc operation is more limiting than partial arc.

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. However, overspeed effects on recirculation pumps are included in the analysis.

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.

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 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 limiting fuel thermal margin and overpressure effects have occurred.

15.2.2.3.3 Results

a. Generator Load Rejection with Bypass

<Figure 15.2-2> shows the results of the generator trip from 105 percent NB rated power. Peak neutron flux rises 24 percent above initial conditions.

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

b. Generator Load Rejection with Failure of Bypass

For the case of bypass failure, peak neutron flux reaches about 292 percent of rated, average surface heat flux reaches 109 percent of its initial value <Figure 15.2-3>.

MCPR stays above 1.10 for this event.

15.2.2.3.4 Consideration of Uncertainties

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

All systems utilized for protection in this event were assumed to have the most conservative allowable response (e.g., relief setpoints, scram stroke time and worth 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

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

15.2.2.4.2 Generator Load Rejection with Failure of Bypass

Peak pressure at the safety/relief valves reaches 1,192 psig. The peak reactor coolant system pressure reaches 1,229 psig at the bottom of the vessel, well below the reactor coolant system transient pressure limit of 1,375 psig.

15.2.2.5 Radiological Consequences

While the consequences of the events identified previously 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 <Section 15.2.4.5>. Therefore, the radiological exposures noted in <Section 15.2.4.5> cover the consequences of this event.

15.2.3 TURBINE TRIP

The turbine trip with failure of the bypass event was re-analyzed at 3,758 MWt core power conditions shown in <Table 15.0-1>. The results of the new analysis are reported in this section.

This transient was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed transients.

15.2.3.1 Identification of Causes and Frequency Classification

15.2.3.1.1 Identification of Causes

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

15.2.3.1.2 Frequency Classification

15.2.3.1.2.1 Turbine Trip

This 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 with Failure of the Bypass

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

Frequency:	0.0064/plant year
MTBE:	156 years

Frequency Basis: As discussed in <Section 15.2.2.1.2.b>, the failure rate of the bypass is 0.0048. Combining this with the turbine trip frequency of 1.33 events/plant year yields the frequency of 0.0064/plant year.

However, PNPP is committed to compare the consequences of this transient with the allowable MCPR of a moderate frequency event.

15.2.3.2 Sequence of Events and Systems Operation

15.2.3.2.1 Sequence of Events

a. Turbine Trip

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

b. Turbine Trip with Failure of the Bypass

Turbine trip at high power with bypass failure produces the sequence of events listed in <Table 15.2-5>.

15.2.3.2.1.1 Identification of Operator Actions

The operator should:

a. Verify auto transfer of buses supplied by generator. If automatic transfer does not occur, manual transfer must be made.

b. Monitor and maintain reactor water level at required level.

c. Depending on conditions, initiate normal operating procedures for cooldown, or maintain pressure for restart purposes.

d. Put the mode switch in the shutdown position before the reactor pressure decays to <850 psig.

e. Secure the RCIC operation if reactor water level can be maintained above Level 2 without RCIC in operation.

f. Prevent reactor vessel water level from dropping to MSIV isolation signal (Level 1).

g. Monitor control rod drive positions and insert both the IRMs and SRMs.

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

15.2.3.2.2 Systems Operation

a. Turbine Trip

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

Turbine stop valve closure initiates a reactor scram trip via position signals to the protection system. Credit is taken for successful operation of the reactor protection system.

Turbine stop valve closure initiates recirculation pump trip (RPT) thereby reducing core flow.

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

Below 38 percent NB rated power level, main stop valve scram trip and recirculation pump trip inhibit signals derived from the first stage pressure of the turbine, are activated. This is because below the 38 percent power level, the neutron flux scram or reactor pressure scram functions alone can provide adequate core protection, should the turbine bypass valves fail to open. This power level is not related to installed bypass capacity; however, the installed bypass capacity is sufficient to accommodate a turbine trip without the necessity of shutting down the reactor.

b. Turbine Trip with Failure of the Bypass

This sequence of events is the same as (a) above except that failure of the main turbine bypass system is assumed.

15.2.3.2.3 The Effect of Single Failures and Operator Errors

a. Turbine Trips at Power Levels Greater Than 38 Percent NBR

Mitigation of pressure increase is accomplished by the reactor protection system functions. Main stop valve closure scram trip and RPT are designed to satisfy single failure criteria.

b. Turbine Trips at Power Levels Less Than 38 Percent NBR

This sequence is the same as in (a) above, with the exception that reactor pump trip and stop valve closure scram trip are normally bypassed at these power levels. Protection is still available from other reactor protection channels such as, high flux and high pressure to scram the reactor, should a single failure occur.

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

15.2.3.3.1 Mathematical Model

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

15.2.3.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in <Table 15.0-1>.

Turbine stop valves full stroke closure time is 0.1 second. This is consistent with the design specification limit given in <Section 10.2>.

A reactor scram is initiated by position switches on the stop valves when the valves are less than 90 percent open. (This stop valve scram trip signal is automatically bypassed when the reactor is below 40 percent NB rated power level.)

Reduction in core recirculation flow is initiated by position switches on the main stop valves, which actuate trip circuitry which trips the recirculation pumps. (This trip signal is also bypassed below 40% NB rated power level.)

15.2.3.3.3 Results

a. Turbine Trip

A turbine trip with the bypass system operating normally is simulated at 105 percent NB rated steam flow conditions in <Figure 15.2-4>.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited to 114.5 percent of rated by the stop valve scram and the RPT system. Peak fuel surface heat flux does not exceed its initial value.

b. Turbine Trip with Failure of Bypass

A turbine trip with failure of the bypass system is simulated at 100 percent NB rated steam flow conditions in <Figure 15.2-5>.

Peak neutron flux reaches 250 percent of its rated value, and average surface heat flux reaches 106.5 percent of initial value. Therefore, this transient is less severe than the generator load rejection with failure of bypass transient as described in <Section 15.2.2.3.3.b>.

c. Turbine Trip with Bypass Failure, Low Power

This transient is less severe than a similar one at high power. Below 38 percent of rated power, the turbine stop valve closure and turbine control valve closure scrams, and the end of cycle recirculation pump trip, are automatically bypassed. The scram which terminates the transient is initiated by high neutron flux or high vessel pressure. The bypass valves are assumed to fail; therefore, system pressure will increase until the safety/relief setpoints are reached. At this time, because of the relatively low power of this transient event, relatively few relief valves will open to limit reactor pressure. Peak pressures are not expected to greatly exceed the safety/relief valve setpoints and will be significantly below the reactor coolant system transient pressure limit of 1,375 psig. Peak surface heat flux and peak fuel center temperature remain at relatively low values and MCPR remains well above the GETAB safety limit <Section 15.0.3.3.3>.

15.2.3.3.4 Consideration 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-rods-out end-of-equilibrium cycle conditions is assumed.

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

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

15.2.3.4 Barrier Performance

15.2.3.4.1 Turbine Trip

Peak pressure in the bottom of the vessel reaches 1,188 psig, which is below the ASME code limit of 1,375 psig for the reactor coolant pressure boundary. Vessel dome pressure does not exceed 1,158 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 with Failure of the Bypass

The safety/relief valves open and close sequentially as the stored energy is dissipated and the pressure falls below the setpoints of the valves. Peak nuclear system pressure reaches 1,228 psig at the vessel bottom; therefore, the overpressure transient is clearly below the reactor coolant pressure boundary transient pressure limit of 1,375 psig. Peak dome pressure does not exceed 1,199 psig.

15.2.3.4.2.1 Turbine Trip with Failure of Bypass at Low Power

Qualitative discussion is provided in <Section 15.2.3.3.3.c>.

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

<Section 15.2.4.5>. Therefore, the radiological exposures noted in <Section 15.2.4.5> cover the consequences of this event.

15.2.4 MSIV CLOSURE

This transient was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed transients. A similar event, MSIV closure with a flux scram is analyzed for overpressure protection purposes only, as part of the reload safety analysis discussed in <Appendix 15B>.

The results of the original MSIV closure event analysis are preserved in this section. The MSIV closure with a flux scram event was re-analyzed at 3,758 MWt core power conditions shown in <Table 15.0-1>. The event is described in detail in <Appendix 15B>, and the results are reported in <Table 15.0-2b> and <Figure 15.2-6b>.

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 pressure, high steam flow, low water level, or manual action.

15.2.4.1.2 Frequency Classification

a. 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 or low condenser vacuum, 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 four 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.

b. 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 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 <u>Sequence of Events and Systems Operation</u>

15.2.4.2.1 Sequence of Events

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

15.2.4.2.1.1 Identification of Operator Actions

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

a. Observe that all rods have inserted.

b. Observe that the relief valves have opened for reactor pressure control.

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

d. (Deleted)

e. When the reactor vessel level has recovered to a satisfactory level, secure HPCS.

f. When the reactor pressure has decayed sufficiently for RHR operation, put it into service per procedure.

g. Do not reset and open MSIVs unless conditions warrant and be sure the pressure regulator setpoint is above vessel pressure.

h. Survey maintenance requirements and complete the scram report.

15.2.4.2.2 Systems Operation

a. Closure of All Main Steam Isolation Valves

MSIV closures initiate a reactor scram trip via position signals to the reactor protection system. Credit is taken for successful operation of the reactor protection system.

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

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

b. Closure of One Main Steam Isolation Valve

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 closure 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 a reactor scram via MSIV position switches and the reactor protection system. Relief valves also operate to limit system pressure. These functions are designed to single failure criteria.

Failure of a single relief valve to open is not expected to have any significant effect. Such a failure is expected to result in less than a 5 psi increase in the maximum vessel pressure rise. The peak pressure will still remain considerably below 1,375 psig. The design basis and performance of the pressure relief system is discussed in <Section 5.0>.

15.2.4.3 Core and System Performance

15.2.4.3.1 Mathematical Model

The computer model described in <Section 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 15.0-1>.

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

a. Closure of All Main Steam Isolation Valves

   <Figure 15.2-6a> 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 NB rated steam flow. Peak neutron flux and fuel surface heat flux show no increase.

   Water level decreases sufficiently to cause a recirculation system trip and initiation of the HPCS and RCIC systems at some time greater than 10 seconds. However, there is a delay up to 30 seconds before the water supply enters the vessel. Nevertheless, there is no change in the thermal margins.

b. 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 approximately 75 to 80 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 rated power conditions, the steam flow disturbance raises vessel pressure and reactor power enough to initiate a high neutron flux scram. Since this transient is considerably milder than closure of all MSIV's at full power, 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 SRV setpoints.

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 <Section 15.2.4.3.3.a>.

15.2.4.3.4 Consideration 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 end-of-equilibrium cycle conditions is assumed.

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

d. Setpoints of the safety/relief valves are assumed to be 1 to 2 percent higher than the valve's nominal setpoint.

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 2.7 seconds after the start of isolation. The valves close sequentially as the stored energy is dissipated but continue to discharge intermittently due to decay heat. Peak pressure at the vessel bottom reaches 1,207 psig, clearly below the pressure limits of the reactor coolant pressure boundary. Peak pressure in the main steam line is 1,174 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 General Observations

The radiological impact of many transients and accidents involves the consequences which do not lead to fuel rod damage as a direct result of the event itself. Additionally, many events do not lead to the depressurization of the primary system but only the venting of sensible heat and energy via fluids at coolant loop activity through relief valves to the suppression pool. In the case of previously defective fuel rods, a depressurization transient will result in considerably more fission product carryover to the suppression pool than will hot standby transients. The time duration of the transient varies from several minutes to greater than four hours, further increasing the variation in activity release.

The above observations lead to the conclusion that radiological events can involve a broad spectrum of results. For example:

a. Where appropriate operator action (seconds) results in quick return (minutes) to planned operation, little radiological impact results.

b. Where major RCPB equipment failure requires immediate plant shutdown and its attendant depressurization under controlled shutdown (4 hours), the radiological impact is greater.

In order to envelope the potential radiological impact of MSIV closure, a worst case like major equipment failure (b) is described below. However, it should be noted that most transients involve appropriate operator action and the analysis conservatively over-predicts the actual radiological impact by a factor greater than 100.

15.2.4.5.2 Depressurization - Shutdown Evaluation

15.2.4.5.2.1 Fission Product Release from Fuel

While no fuel rods are damaged as a consequence of this event, fission product activity associated with normal coolant activity levels as well as that released from previously defective rods will be released to the suppression pool as a consequence of SRV actuation and vessel depressurization. The release of activity from previously defective rods is based in part upon measurements obtained from operating BWR plants (Reference 1).

Those transients identified previously which cause SRV actuation will result in various vessel depressurization and steam blowdown rates. The transient evaluated in this section is that which maximizes the radiological consequences for all transients of this nature. This transient is the closure of all main steam line isolation valves. The specific models and assumptions used in the evaluation are described in (Reference 2). The activity released to the environs is presented in <Table 15.2-7> which was used in evaluating the radiological dose consequences in this section.

15.2.4.5.2.2 Fission Product Release to Environment

Since this event does not result in the immediate need to purge the containment, it is assumed that purging of the containment through the containment vessel and purge system occurs under average annual meteorological conditions and commences 8 hours after initiation of the event.

15.2.4.5.2.3 Offsite Dose

As noted above, purging of the containment is assumed to occur under average annual meteorological conditions. To simplify the radiological calculation, it is assumed the radiological dose commitment at the site boundary is proportional to the average annual X/Q value, which is 7.1E-6 sec/m^3. The breathing rate is assumed to be 347 cc/sec and the dose recipient is located at one position for the entire release period. The radiological doses for this event are 0.65 mrem whole body and 0.037 mrem thyroid.

15.2.4.5.2.4 Onsite Dose

The onsite radiological consequences of this event are presented in <Section 12.2.2>.

15.2.5 LOSS OF CONDENSER VACUUM

This transient was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed transients.

15.2.5.1 Identification of Causes and Frequency Classification

15.2.5.1.1 Identification of Causes

Various system malfunctions which can cause a loss of condenser vacuum due to some single equipment failure are designated in <Table 15.2-9>.

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 15.2-10> lists the sequence of events for <Figure 15.2-7>.

15.2.5.2.1.1 Identification of Operator Actions

The operator should:

a. Verify auto transfer of buses supplied by generator to incoming power; if automatic transfer has not occurred, manual transfer must be made.

b. Monitor and maintain reactor water level at required level.

c. Depending on conditions, initiate normal operating procedures for cooldown, or maintain pressure for restart purposes.

d. Put the mode switch in the “shutdown” position before the reactor pressure decays to <850 psig.

e. If auto initiation occurred due to low water level, and RCIC is no longer required, secure RCIC operation.

f. Monitor control rod drive positions and insert both the IRMs and SRMs.

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

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 associated with loss of main turbine condenser vacuum are designated in <Table 15.2-11>.

15.2.5.2.3 The Effect of Single Failures and Operator Errors

This event does not lead to an increase in reactor power level.

Failure of the integrity of the condenser gas treatment system is considered to be an accident situation and is described in <Section 15.7.1>.

Single failures will not effect the vacuum monitoring and turbine trip devices which are redundant. The protective sequences of the anticipated operational transient are shown in <Appendix 15A> to be single failure proof.

15.2.5.3 Core and System Performance

15.2.5.3.1 Mathematical Model

The computer model described in <Section 15.1.1.3.1> 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 15.0-1> unless otherwise noted.

Turbine stop valves full stroke closure time is 0.1 seconds.

A reactor scram is initiated by position switches on the stop valves when the valves are less than 90 percent open. This stop valve scram trip signal is automatically bypassed when the reactor is below 40 percent NB rated power level.

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

15.2.5.3.3 Results

Under this hypothetical 2 inches Hg per second vacuum decay condition, the turbine bypass valve and main steam isolation valve closure would follow main turbine and feedwater turbine trips about 5 seconds after they initiate the transient. This transient, therefore, is similar to a normal turbine trip with bypass. The effect of main steam 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 NB rated steam flow conditions. Peak neutron flux reaches 120 percent of NB rated power while average fuel surface heat flux shows no increase. Safety/relief valves open to limit the pressure rise, then sequentially reclose as the stored energy is dissipated.

15.2.5.3.4 Consideration of Uncertainties

The reduction or loss of vacuum in the main turbine condenser will sequentially trip the main and feedwater turbines and close the 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 directly dependent upon the rate at which the vacuum pressure is lost. Normal loss of vacuum due to loss of cooling water pumps or steam jet air ejector problem produces a very slow rate of loss of vacuum (minutes, not seconds) <Table 15.2-9>. If corrective actions by the reactor operators are not successful, then simultaneous trips of the main and feedwater turbines, and ultimately complete isolation by closing the bypass valves (opened with the main turbine trip) and the MSIVs, will occur.

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

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

a. Slowest allowable control rod scram motion is assumed.

b. Scram worth shape for all-rods-out condition is assumed.

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

d. Setpoints of the safety/relief valves are assumed to be 1 to 2 percent higher than the valve’s nominal setpoint.

15.2.5.4 Barrier Performance

Peak nuclear system pressure is 1,182 psig at the vessel bottom, below the reactor coolant system transient pressure limit of 1,375 psig. Vessel dome pressure does not exceed 1,154 psig. A comparison of these values to those for turbine trip with bypass failure at high power shows the similarities between these two transients. The prime differences are the loss of feedwater and main steam line isolation, and the resulting low water level trips.

15.2.5.5 Radiological Consequences

While the consequences of loss of condenser vacuum events 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 <Section 15.2.4.5>. Therefore, the radiological exposures noted in <Section 15.2.4.5> cover the consequences of this event.

15.2.6 LOSS OF AC POWER

This transient was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed transients.

15.2.6.1 Identification of Causes and Frequency Classification

15.2.6.1.1 Identification of Causes

a. Loss of Auxiliary Power Transformer

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

b. 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

a. Loss of Auxiliary Power Transformer

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

b. 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

a. Loss of Auxiliary Power Transformer

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

b. Loss of All Grid Connections

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

15.2.6.2.1.1 Identification of Operator Actions

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

a. Following the scram, verify all rods in.

b. Check that emergency diesel generators start and carry the vital loads.

c. Check that relays on the reactor protection system (RPS) drop out.

d. Check that both RCIC and HPCS start when reactor vessel level drops to the initiation point after the relief valves open.

e. Break vacuum before loss of sealing steam occurs.

f. Check turbine generator auxiliaries during coastdown. Verify that the turbine dc oil pump is operating satisfactorily to prevent turbine bearing damage.

g. When both the reactor pressure and level are under control, secure both HPCS and RCIC as necessary after it has been verified that initiation is not due to a LOCA.

h. Continue cooldown per the normal procedure.

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

15.2.6.2.2 Systems Operation

a. Loss of Auxiliary Power 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 auxiliary power. Estimates of the responses of the various reactor systems (assuming loss of the auxiliary transformer) provide the following simulation sequence:

1. Recirculation pumps are tripped at a reference time (t=0), with normal coastdown times.

2. Within 8 seconds, the loss of main condenser circulating water pumps causes condenser vacuum to drop to the main turbine and feedwater turbine trip setting, causing stop valve closure and scram when the stop valves are less than 90 percent open, assuming 0.5 in. Hg/sec vacuum decay rate. However, scram, main turbine and feedwater turbine tripping may occur earlier

than this time, if water level reaches the high water level (L8) setpoint before 8 seconds.

3. At approximately 28 seconds, the loss of condenser vacuum is expected to reach the bypass valves closure setpoint and main steam line isolation setpoint.

Following main steam line isolation the reactor pressure is expected to increase until the safety/relief valve setpoints are reached. The valves subsequently operate in a cyclic manner to discharge the decay heat to the suppression pool.

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.

b. Loss of All Grid Connections

Same as <Section 15.2.6.2.2.a> 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 scram and initiates recirculation pump trip (already tripped at t=0).

15.2.6.2.3 The Effect of Single Failures and Operator Errors

Loss of the auxiliary power transformer in general leads to a reduction in power level due to rapid pump coastdown with pressurization effects due to turbine trip occurring after the reactor scram has occurred. Failures of the reactor protection systems have been considered and satisfy single failure criteria; no change in analyzed consequences is expected. See <Appendix 15A> for details on single failure analysis.

15.2.6.3 Core and System Performance

15.2.6.3.1 Mathematical Model

The computer model described in <Section 15.1.1.3.1> was used to simulate this event.

15.2.6.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in <Table 15.0-1> and under the assumed systems constraints described in <Section 15.2.6.2.2>, for both loss of auxiliary power transformer and all grid connections.

15.2.6.3.3 Results

a. Loss of Auxiliary Power Transformer

   <Figure 15.2-8> shows the simulated transient. The initial portion of the transient is similar to the recirculation pump trip transient. At 2 seconds, scram and main steam isolation valve closure occur.

   Sensed level drops to the RCIC and HPCS initiation setpoint at approximately 20 seconds after loss of auxiliary power. Reactor pressure is controlled by use of the relief valves.

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

b. Loss of All Grid Connections

Loss of all grid connections is a more general form of loss of auxiliary power. It essentially takes on the characteristic response of the standard full load rejection discussed in <Section 15.2.2>. <Figure 15.2-9> shows the simulated event. Peak neutron flux reaches 111 percent of NB rated power while fuel surface heat flux shows no increase.

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.

15.2.6.4 Barrier Performance

15.2.6.4.1 Loss of Auxiliary Power 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 setpoints. The pressure in the vessel bottom is limited to a maximum value of 1,182 psig, well below the vessel pressure limit of 1,375 psig.

15.2.6.5 Radiological Consequences

While the consequences of the events identified previously 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 <Section 15.2.4.5>; therefore, the radiological exposures noted in <Section 15.2.4.5> cover the consequences of this event.

15.2.7 LOSS OF FEEDWATER FLOW

This transient was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed transients. The plant type generic analyses documented in (Reference 6) concluded that "All BWR/4, 5, and 6 plants will maintain adequate water level for loss of feedwater flow transients for uprated power operation."

15.2.7.1 Identification of Causes and Frequency Classification

15.2.7.1.1 Identification of Causes

A loss of feedwater flow could occur from pump failures, feedwater controller failures, operator errors, or reactor system variables such as high vessel water level (L8) trip signal.

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

<Table 15.2-14> lists the sequence of events for <Figure 15.2-10>.

15.2.7.2.1.1 Identification of Operator Actions

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

a. Verify all rods in, following the scram.

b. Verify that the recirculation pumps trip on reactor low low level.

c. Verify HPCS and RCIC initiation.

d. Secure HPCS when reactor level and pressure are under control.

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

f. Monitor turbine coastdown and turbine generator auxiliaries; break vacuum as necessary.

g. Complete scram report and survey maintenance requirements.

15.2.7.2.2 Systems Operation

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

Containment isolation, when it occurs, would also initiate a main steam 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.

15.2.7.2.3 The Effect of Single Failures and Operator Errors

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 criteria; therefore, any additional failure in these shutdown methods would not aggravate or change the simulated transient. See <Appendix 15A> for details.

15.2.7.3 Core and System Performance

15.2.7.3.1 Mathematical Model

The computer model described in <Section 15.1.1.3.1> was used to simulate this event.

15.2.7.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in <Table 15.0-1>.

15.2.7.3.3 Results

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 setpoint 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.

15.2.7.3.4 Consideration of Uncertainties

End-of-cycle scram characteristics are assumed.

This transient is most severe from high power conditions, because the rate of level decrease is greatest and the amount of stored 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 these systems have no significant effect on the results of this transient.

15.2.7.4 Barrier Performance

Peak pressure in the bottom of the vessel reaches 1,087 psig, which is below the ASME Code limit of 1,375 psig for the RCPB. Vessel dome pressure does not exceed 1,045 psig. The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.7.5 Radiological Consequences

The consequences of this event do not result in any fuel failure. Therefore, no analysis of the radiological consequences is required.

15.2.8 FEEDWATER LINE BREAK

Refer to <Section 15.6.6>.

15.2.9 FAILURE OF RHR SHUTDOWN COOLING

This transient was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed transients.

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

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

a. Loss of all offsite ac power.

b. Utilization of safety shutdown equipment only.

c. No operator involvement until 10 minutes after accident initiation.

These assumptions certainly would change the initial incident (malfunction of RHR suction valve) from a moderate frequency incident to a design basis accident. However, the event is evaluated as a moderate frequency event.

15.2.9.1 Identification of Causes and Frequency Classification

15.2.9.1.1 Identification of Causes

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

15.2.9.1.2 Frequency Classification

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

a. No RHR valves have failed in the shutdown cooling mode in BWR total operating experience. Note, the PNPP Heat Exchanger Bypass Valve 1E12F0048A has two documented failures.

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

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

15.2.9.2.1 Sequence of Events

The sequence of events for this event is shown in <Table 15.2-15>.

15.2.9.2.1.1 Identification of Operator Actions

For the early part of the transient, the operator actions are identical to those described in <Section 15.2.6> (loss of offsite power event with isolation/scram). The operator should do the following:

a. At 13 minutes into the transient, initiate suppression pool cooling (for purposes of this analysis, it is assumed that only one RHR heat exchanger is available).

b. All of the feedwater inventory in the feedwater piping after the No. 5 heater is assumed to be injected into the vessel.

c. Initiate RPV depressurization by manual actuation of ADS valves.

d. After the RPV is depressurized to approximately 100 psig, the operator should attempt to open the RHR shutdown cooling suction valve that lost power (these attempts are assumed unsuccessful).

e. At 100 psig RPV pressure, the operator establishes a closed cooling path as described in the notes for <Figure 15.2-11>.

15.2.9.2.2 System Operation

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

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. This event can occur only during the low pressure portion of a normal reactor shutdown and cooldown, when the RHR system is operating in the shutdown cooling mode. During this time 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 assumed for operator action is an estimate of how long it would take the operator to initiate the necessary actions; it is not a time by which he must initiate action.

15.2.9.3.2 Mathematical Model

In evaluating this event, the important parameters to consider are reactor depressurization rate and suppression pool temperature. Models used for this evaluation are described in (Reference 3) and (Reference 4).

15.2.9.3.3 Input Parameters and Initial Conditions

<Table 15.2-16> shows the input parameters and initial conditions used in evaluation of this event.

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 re-established using other, normal shutdown cooling equipment. In cases where both of the RHR shutdown cooling suction valves cannot be opened, alternate paths are available to accomplish the shutdown cooling function <Figure 15.2-12>. An evaluation has been performed assuming the worst single failure that could disable a RHR shutdown cooling valve.

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 (Reference 5) and <Figure 15.2-11>.

The alternate shutdown systems are capable of performing the function of transferring heat from the reactor to the environment using only safety grade systems. Even if it is additionally postulated that all of the 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 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 (14.7 psia and 200°F) conditions.

15.2.9.3.4.1 Full Power to Approximately 100 psig

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

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

These systems are designed to routinely perform 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 reaching the 100 psig level.

15.2.9.3.4.2 Approximately 100 psig to Cold Shutdown

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

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

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

c. There is no offsite power available.

In the event that the RHR shutdown suction line is not available because of single failure, the first action to be taken will be to maintain the 100 psig level while personnel gain access and effect repairs. For example, if a single electrical failure caused the suction valve to fail in the closed position, a hand wheel is provided on the valve to allow manual operation. Nevertheless, if for some reason the normal shutdown cooling supply cannot be restored, 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. Safety system 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 emergency diesel power)

RHR Loop B (Division 2 emergency diesel power)

RHR Loop C (Division 2 emergency diesel power)

HPCS (Division 3 emergency diesel power)

RCIC (DC Division 1)

LPCS (Division 1 emergency diesel power)

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

Case A. Division 1 fails, Division 2 and Division 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 the cooling loops described in Activity Cl of <Figure 15.2-11>.

Case B. Division 2 fails, Division 1 and Division 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 emergency diesel power, the safety function is accomplished by establishing the cooling loops described in Activity C2 of <Figure 15.2-11>. Simplified RHR flow diagrams are shown in <Figure 15.2-14 (1)>, <Figure 15.2-14 (2)>, <Figure 15.2-15 (1)>, and <Figure 15.2-15 (2)>.

Using the above assumptions the suppression pool temperature is shown in <Figure 15.2-17> and <Figure 15.2-17a>.

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. Note that the containment structural analysis includes margin for two thermal cycles over 150°F but less than 185°F. 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 <Section 15.2.4.5>. Therefore, the radiological exposures noted in <Section 15.2.4.5> cover the consequences of this event.

15.2.10 LOSS OF INSTRUMENT AIR

This transient was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed transients.

15.2.10.1 Event Evaluation

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

15.2.10.2 Analysis of Effects and Consequences

Loss of the instrument air system will result in shutdown of the reactor due to closing of the main steam isolation valves. Failure of instrument air will not interfere with safe shutdown of the reactor since all equipment using instrument air is designed to fail to a position that is consistent with safe shutdown of the plant.

Air operated equipment that must be available for use in the event of an instrument air system failure, is provided with backup accumulators to provide the required air supply.

15.2.10.3 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.

15.2.10.4 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 FOR 15.2

1. Brutschy F. G., et al, "Behavior of Iodine in Reactor Water During Plant Shutdown and Startup," (NEDO-10585).

2. Tschaeche, A. N., "Mark III Suppression Pool Source Terms," 22A6215 (July 1978).

3. Fukushima, T. Y., "HEX01 User Manual," July 1976 (NEDE-23014).

4. Bilanin, W. I., Bodily, R. J., and Cruz, G. A., "The General Electric Mark III Pressure Suppression Containment System Analytical Model (Supplement 1)," September 1975 (NEDO-20533, Supplement 1).

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

6. GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991.

TABLE 15.2-1

SEQUENCE OF EVENTS FOR <FIGURE 15.2-1>
PRESSURE REGULATOR DOWNSCALE FAILURE

Time-sec	Event
0	Simulate zero steam flow demand to main turbine and bypass valves.
0	Turbine control valves start to close.
.98	Neutron flux reaches high flux scram setpoint and initiates a reactor scram.
1.68	Recirculation pump drive motors are tripped due to high dome pressure.
2.77	Safety/relief valves open due to high pressure.
NA	Vessel water level (L8) trip initiates main turbine and feedwater turbine trips.
3.76	Main turbine stop valves closed.
5.0	Safety/relief valves close.
7.20	Group 1 safety/relief valves open again to relieve decay heat.
>9 (est.)	Group 1 safety/relief valves close.

TABLE 15.2-2

SEQUENCE OF EVENTS FOR <FIGURE 15.2-2>
GENERATOR LOAD REJECTION

Time-sec	Event
(-)0.015 (approx.)	Turbine generator detection of loss of electrical load.
0	Turbine generator power load unbalance (PLU) devices trip to initiate turbine control valve fast closure.
0	Fast control valve closure (FCV) initiates scram trip.
0	Fast control valve closure (FCV) initiates a recirculation pump trip (RPT).
0.07	Turbine control valves closed.
0.1	Turbine bypass valves start to open.
1.5	Safety/relief valves open due to high pressure.
4.0	Vessel water level (L8) trip initiates trip of the feedwater turbines.
6.9	Safety/relief valves close.

TABLE 15.2-3

SEQUENCE OF EVENTS FOR <FIGURE 15.2-3>
GENERATOR LOAD REJECTION, TRIP SCRAM, BYPASS-OFF

Time-sec	Event
(-)0.015 (approx.)	Turbine generator detection of loss of electrical load.
0	Turbine generator power load unbalance (PLU) devices trip to initiate turbine control valve fast closure.
0	Turbine bypass valves fail to operate.
0	Fast control valve closure (FCV) initiates scram trip.
0	Fast control valve closure (FCV) initiates a recirculation pump trip (RPT).
0.001	Turbine control valves closed.
1.49	Safety/relief valves open due to high pressure.
NA	Vessel water level (L8) trip initiates trip of the feedwater turbines.
4.5	Safety/relief valves close.
>6	Group 1 safety/relief valves open again to relieve decay heat.
>6 (est.)	Group 1 safety/relief valves close again.

TABLE 15.2-4

SEQUENCE OF EVENTS FOR <FIGURE 15.2-4>
TURBINE TRIP, TRIP SCRAM, BYPASS AND RPT-ON

Time-sec	Event
0	Turbine trip initiates closure of main stop valves.
0	Turbine trip initiates bypass operation.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip.
0.01	Main turbine stop valves reach 90% open position and initiate a recirculation pump trip (RPT).
0.1	Turbine stop valves close.
0.1	Turbine bypass valves start to open.
1.6	Safety/relief valves open due to high pressure.
4.0	Vessel water level (L8) trip initiates trip of the feedwater turbines.
6.9	Safety/relief valves close.

TABLE 15.2-5

SEQUENCE OF EVENTS FOR <FIGURE 15.2-5>
TURBINE TRIP, TRIP SCRAM, BYPASS-OFF, RPT-ON

Time-sec	Event
0	Turbine trip initiates closure of main stop valves.
0	Turbine bypass valves fail to operate.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip.
0.01	Main turbine stop valves reach 90% open position and initiate a recirculation pump (RPT) trip.
0.1	Turbine stop valves close.
1.55	Safety/relief valves open due to high pressure.
NA	Vessel water level (L8) trip initiates trip of the feedwater turbines.
4.5	Safety/relief valves close.
>6	Group 1 safety/relief valves open again to relieve decay heat.
>6 (est.)	Group 1 safety/relief valves close again.

TABLE 15.2-6

SEQUENCE OF EVENTS FOR <FIGURE 15.2-6a>
THREE SECOND CLOSURE OF ALL MAIN STEAM LINE
ISOLATION VALVES WITH POSITION SWITCH SCRAM TRIP

Time-sec	Event
0	Initiate closure of all main steam line isolation valves (MSIV).
0.3	MSIVs reach 90% open.
0.3	MSIV position trip scram initiated.
1.9	Recirculation pump drive motors are tripped due to low water Level 3 (L3) trip.
2.7	Safety/relief valves open due to high pressure.
8.1	Safety/relief valves close.
9.1	Group 1 safety/relief valves open again to relieve decay heat.
>10 (est.)	Group 1 safety/relief valves close again.
>10 (est.)	Vessel water level reaches L2 setpoint.
>40 (est.)	HPCS and RCIC flow into vessel (not included in simulation).

TABLE 15.2-7

CLOSURE OF ALL MAIN STEAM ISOLATION VALVES
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

I-131	7.3E-3
132	1.9E-2
133	6.5E-3
134	1.8E-4
135	3.4E-3
Kr-83m	1.4E+1
85	5.4E-1
85m	7.7E+1
87	5.7E+0
88	1.0E+2
Xe-131m	3.7E+0
133m	3.0E+1
133	1.4E+3
135m	2.3E-2
135	4.8E+2

<TABLE 15.2-8>

DELETED

TABLE 15.2-9

TYPICAL RATES OF DECAY FOR CONDENSER VACUUM

	Cause	Estimated Vacuum Decay Rate
a.	Failure or Isolation of Steam Jet Air Ejectors	<1 inch Hg/minute
b.	Loss of Sealing Steam to Shaft Gland Seals	~1 to 2 inches Hg/minute
c.	Opening of Vacuum Breaker Valves	~2 to 12 inches Hg/minute
d.	Loss of One or More Circulating Water Pumps	~4 to 24 inches Hg/minute

TABLE 15.2-10

SEQUENCE OF EVENTS FOR <FIGURE 15.2-7>

Time-sec	Event
-3.0 (est.)	Initiate simulated loss of condenser vacuum at 2 inches of Hg per second.
0.0 (est.)	Low condenser vacuum main turbine trip actuated.
0.0 (est.)	Low condenser vacuum feedwater trip actuated.
0.01	Main turbine trip initiates recirculation pump trip (RPT) and scram.
1.8	Safety/relief valves open due to high pressure.
5.0	Low condenser vacuum initiates main steam isolation valve closure.
5.0	Low condenser vacuum initiates bypass valve closure.
6.9	Safety/relief valves close.
8.0	Group 1 safety/relief valves open again to relieve decay heat.
13.0	Group 1 safety/relief valves close again.
13.7	Water level reaches Level 2 setpoint and initiates HPCS and RCIC.
15.7	Group 1 safety/relief valves open again to relieve decay heat.
22.5	Group 1 safety/relief valves close again.
27.5	Group 1 safety/relief valves open again to relieve decay heat.
32.8	Group 1 safety/relief valves close again.
41.7	Group 1 safety/relief valves open again to relieve decay heat.

TABLE 15.2-10 (Continued)

Time-sec	Event
43.7 (est.)	HPCS and RCIC flow enters vessel (not in simulation).
46.7	Group 1 safety/relief valves close again.

TABLE 15.2-11

TRIP SIGNALS ASSOCIATED WITH LOSS OF CONDENSER VACUUM

Vacuum (inches of Hg absolute)	Protective Action Initiated
4 to 5	Alarm
7 to 10	Main turbine trip and feedwater turbine trip (stop valve closures).
20 to 23	Main steam isolation valve (MSIV) closure and bypass valve closure.

TABLE 15.2-12

SEQUENCE OF EVENTS FOR <FIGURE 15.2-8>

Time-sec	Event
0	Loss of auxiliary power transformer occurs.
0	Recirculation system pump motors are tripped.
0	Hotwell and condensate booster pumps are tripped.
0	Condenser circulating water pumps tripped.
2.0	Main steam isolation valves close due to loss of power causing a reactor scram.
4.0	Feedwater pump turbines are tripped.
5.1	Safety/relief valves open due to high pressure.
10.1	Safety/relief valves close.
20.1	Vessel water level reaches Level 2 setpoint.
50.1 (est.)	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.2-13

SEQUENCE OF EVENTS FOR <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 power load unbalance (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.
0.1	Turbine bypass valves open.
1.7	Safety/relief valves open due to high pressure.
2.0	Main steam isolation valves close due to loss of power.
4.0	Feedwater pumps trip due to MSIV closure.
18.6	Safety/relief valves close.
20.4	Vessel water level reaches Level 2 setpoint.
28.0	Closure of turbine bypass valves is initiated via low condenser vacuum.
50.4	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.2-14

SEQUENCE OF EVENTS FOR <FIGURE 15.2-10>

Time-sec	Event
0	Trip of all feedwater pumps initiated.
5	Feedwater flow decays to zero.
7.0	Vessel water level (L3) trip initiates scram trip and recirculation flow runback.
14.9	Vessel water level reaches Level 2.
15.1 (est.)	Recirculation pumps trip due to Level 2 trip.
44.9 (est.)	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.2-15

SEQUENCE OF EVENTS FOR FAILURE OF RHR SHUTDOWN COOLING
(at 3729 MWt)

Approximate Elapsed Time	Event
0	Reactor is operating at 104.2% rated power when loss of offsite power occurs initiating plant shutdown.
0	Concurrently loss of division power (i.e., loss of one diesel generator) occurs.
10 min	Suppression pool cooling initiated to prevent overheating from SRV actuation[(1)].
10 min	Controlled depressurization initiated (100°F/hr) using selected safety/relief valves
128 min	Blowdown to approximately 100 psig completed.
158 min	Personnel are sent to open RHR shutdown cooling suction valve; this fails.
163 min	ADS valves are opened to complete blowdown to suppression pool, and RHR pump discharge is redirected from pool to vessel via LPCI line. Alternate shutdown cooling path has now been established.

NOTE:

[(1)] See <Table 15.2-12> for earlier detailed sequence of events for loss of ac power transient.

Revision 12
January, 2003

TABLE 15.2-15a

SEQUENCE OF EVENTS FOR FAILURE OF RHR SHUTDOWN COOLING
(at 3833 MWt)

Approximate Elapsed Time	Event
0	Reactor is operating at 102% rated power when loss of offsite power occurs initiating plant shutdown.
0	Concurrently loss of division power (i.e., loss of one diesel generator) occurs.
13 min	Suppression pool cooling initiated to prevent overheating from SRV actuation[1].
15 min	Controlled depressurization initiated (100°F/hr) using selected safety/relief valves.
135 min	Blowdown to approximately 100 psig completed.
165 min	Personnel are sent to open RHR shutdown cooling suction valve; this fails.
170 min	ADS valves are opened to complete blowdown to suppression pool, and RHR pump discharge is redirected from pool to vessel via LPCI line. Alternate shutdown cooling path has now been established.

NOTE:

[1] See <Table 15.2-12> for earlier detailed sequence of events for loss of ac power transient.

TABLE 15.2-16

INPUT PARAMETERS FOR EVALUATION OF FAILURE
OF RHR SHUTDOWN COOLING
(at 3729 MWt)

Initial power corresponding to 104.2% rated power	
Initial Suppression pool mass, lbm	6,301,000[1]
RHR, KHX value, Btu/sec/°F	440
Initial vessel conditions	
Pressure, psia	1,060
Temperature, °F	552
Initial primary fluid inventory, lbm	564,080
Initial pool temperature, °F	95
Emergency service water temperature, °F	85
Vessel heat capacity, Btu/lbm/°F	0.125
HPCS water level, ft	
on	40.22
off	48
HPCS flow rate, lbm/sec	834
LPCI flow rate per loop, lbm/sec	987
LPCS flow rate, lbm/sec	834

NOTE:

[1] A new strainer design, which has the strainer resting on the floor of the suppression pool, replaces the individual strainers for the ECCS and RCIC system pumps in response to <NRC Bulletin 96-03>. The new strainer displaces ~426ft^3 of suppression pool water. Analysis has shown that the displacement of the water has a negligible effect on the existing analyses.

TABLE 15.2-16a

INPUT PARAMETERS FOR EVALUATION OF FAILURE
OF RHR SHUTDOWN COOLING
(at 3833 MWt)

Initial power corresponding to 102% rated power	
Initial Suppression pool mass, lbm	6,386,000[(1)]
RHR, KHX value, Btu/sec/°F	440
Initial vessel conditions	
Pressure, psia	1,060
Temperature, °F	552
Initial primary fluid inventory, lbm	564,080
Initial pool temperature, °F	95
Emergency service water temperature, °F	85
Vessel heat capacity, Btu/lbm/°F	0.125
HPCS water level, ft	
on	40.45
off	48.78
HPCS flow rate, lbm/sec	973
LPCI flow rate per loop, lbm/sec	1,098
LPCS flow rate, lbm/sec	973

NOTE:

(1) A new strainer design, which has the strainer resting on the floor of the suppression pool, replaces the individual strainers for the ECCS and RCIC system pumps in response to <NRC Bulletin 96-03>. The new strainer displaces ~426ft^3 of suppression pool water. Analysis has shown that the displacement of the water has a negligible effect on the existing analyses.

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

15.3.1 RECIRCULATION PUMP TRIP

This transient was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed transients.

15.3.1.1 Identification of Causes and Frequency Classification

15.3.1.1.1 Identification of Causes

Recirculation pump trip (RPT) is provided by design to mitigate transient effects on core and RCPB design margins. RPT will occur in response to:

a. Reactor vessel water level L2 setpoint trip.

b. Failure to scram high pressure setpoint trip.

c. Motor branch circuit over-current protection.

d. Motor overload protection.

e. Suction or discharge valves not fully open.

f. Auto transfer sequence incomplete (40 sec).

Unintended RPT will occur in response to:

a. Operator error.

b. Loss of electrical power to the pumps.

c. Equipment or sensor failures and malfunctions which initiate the above design trips.

15.3.1.1.2 Frequency Classification

Trip of one or both recirculation pumps is categorized as an event of moderate frequency.

15.3.1.2 Sequence of Events and Systems Operation

15.3.1.2.1 Sequence of Events

a. Trip of One Recirculation Pump

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

b. Trip of Two Recirculation Pumps

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

c. Identification of Operator Actions

   1. Trip of One Recirculation Pump

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

2. Trip of Two Recirculation Pumps

The operator should ascertain that the reactor scrams with the turbine trip resulting from reactor water level swell. The operator should regain control of reactor water level through RCIC operation, or by restart of a feedwater pump, monitoring reactor water level and pressure after shutdown. When both reactor pressure and level are under control, the operator should secure both HPCS and RCIC as necessary. The operator should also determine the cause of the trip prior to returning the system to normal.

15.3.1.2.2 Systems Operation

a. 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.

b. 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 (L8) instrumentation to trip the turbine. Reactor shutdown relies on scram trips from the turbine stop valves. High system pressure is limited by SRV operation.

15.3.1.2.3 The Effect of Single Failures and Operator Errors

a. Trip of One Recirculation Pump

Since no corrective action is required per <Section 15.3.1.2.2.a>, 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.

b. Trip of Two Recirculation Pumps

<Table 15.3-2> lists the vessel level (L8) scram as the first response to initiate corrective action in this transient. This scram trip signal is 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 <Section 15.1.1.3.1> is used to simulate this event.

15.3.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in <Table 15.0-1>.

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

15.3.1.3.3 Results

a. 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 5.6 seconds. However, the ratio of diffuser mass flow to pump mass flow in the active jet pumps increases considerably and produces approximately 130 percent of normal diffuser flow and 54 percent of rated core flow. MCPR remains above the safety limit and the fuel thermal limits are not violated. During this transient, level swell is not sufficient to cause turbine trip and scram.

b. Trip of Two Recirculation Pumps

<Figure 15.3-2> shows 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 scramming the reactor. Subsequent events, such as main steam line isolation and initiation of RCIC and HPCS systems, occur late in this event and 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> indicates a basic reduction in system pressures from 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 peak pressures stay well below the 1,375 psig limit allowed by the applicable code. Therefore, the reactor coolant pressure boundary is not threatened.

15.3.1.5 Radiological Consequences

While the consequences of the events identified previously 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 <Section 15.2.4.5>. Therefore, the radiological exposures noted in <Section 15.2.4.5> cover the consequences of this event.

15.3.2 RECIRCULATION FLOW CONTROL FAILURE - DECREASING FLOW

This transient was not reanalyzed for the current reload as it had been determined to be less limiting and bounded by the analyzed transients.

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 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

a. Fast Closure of One Main Recirculation Valve

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

b. Fast Closure of Two Main Recirculation Valves

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

c. Identification of Operator Actions

1. Fast Closure of One Main Recirculation Valve

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

2. Fast Closure of Two Main Recirculation Valves

As soon as possible, the operator must verify that no operating limits are being exceeded. If they are, corrective actions must be initiated. Also, the operator must determine the cause of the trip prior to returning the system to normal.

15.3.2.2.2 Systems Operation

a. Fast Closure of One Main Recirculation Valve

Normal plant instrumentation and control is assumed to function. No protection system operation is required.

b. 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 <Section 15.3.1.2.3.b>. 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 <Section 15.1.1.3.1> is used to simulate these transient events.

15.3.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions listed in <Table 15.0-1>.

The least negative void coefficient in <Table 15.0-1> was used for these analyses.

a. Fast Closure of One Main Recirculation Valve

   Failure within either loop controller can result in a maximum stroking rate of 60 percent per second as limited by the valve hydraulics.

b. Fast Closure of Two Main Recirculation Valves

   A downscale failure of either the master controller or the flux controller will generate a zero flow demand signal to both recirculation flow controllers. Each individual valve actuator circuitry has a velocity limiter which limits maximum valve stroking rate to 11 percent per second. Recirculation loop flow is allowed to decrease to approximately 25 percent of rated. This is the flow expected when the flow control valves are maintained at a minimum open position.

15.3.2.3.3 Results

a. 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. The design 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.

b. 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 <Section 15.3.1>. The design limit on actuator velocity 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.

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 setpoint 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 1,151 psig, well below the ASME code limit. At approximately 28 seconds, the wide range level falls to the low water level trip setpoint, 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 <u>Radiological Consequences</u>

While the consequences of the events identified previously 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 <Section 15.2.4.5>. Therefore, the radiological exposures noted in <Section 15.2.4.5> cover the consequences of this event.

15.3.3 RECIRCULATION PUMP SEIZURE

This accident was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed accidents. A discussion of the recirculation pump seizure accident and why it is less severe then the LOCA is provided in GESTAR (Reference 1).

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

The seizure of a recirculation pump is considered as a design basis accident event. It is a very mild accident in relation to other design basis accidents such as the LOCA. The analysis has been conducted for both single and two loop operation.

Refer to <Chapter 5> for specific mechanical considerations and <Chapter 8> for electrical aspects.

The seizure event postulated ordinarily 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.

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 an event of greater probability (i.e., infrequent incident classification).

15.3.3.2 Sequence of Events and Systems Operations

15.3.3.2.1 Sequence of Events

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

15.3.3.2.1.1 Identification of Operator Actions

The operator should ascertain that the reactor scrams from reactor water level swell. The operator should regain control of reactor water level through RCIC operation or by restart of a feedwater pump; and he should monitor reactor water level and pressure control after shutdown.

15.3.3.2.2 Systems Operation

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

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

15.3.3.2.3 The Effect of Single Failures and Operator Errors

Single failures in the scram logic originating via the high vessel level (L8) trip are similar to the considerations in <Section 15.3.1.2.3.b>.

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 model described briefly in <Section 15.1.1.3.1> is used to simulate this event.

15.3.3.3.2 Input Parameters and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in <Table 15.0-1>.

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 one recirculation pump shaft while the reactor is operating at 105 percent NB rated steam flow. Also, the reactor is assumed to be operating at thermally limited conditions.

The void coefficient is adjusted to the most conservative value, that is, the least negative value in <Table 15.0-1>.

15.3.3.3.3 Results

<Figure 15.3-5> presents the results of the accident. MCPR does not decrease significantly before fuel surface heat flux begins dropping enough to restore greater thermal margins. The level swell produces a trip of the main turbine and feedwater pumps and scram at 3.4 seconds into the transient. 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.

15.3.3.3.4 Consideration of Uncertainties

Consideration of uncertainties is included in the GETAB analysis <Section 15.0.3.3.3>.

15.3.3.4 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.

15.3.3.5 Radiological Consequences

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

15.3.4 RECIRCULATION PUMP SHAFT BREAK

This accident was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed accidents. A recirculation pump shaft break is bounded by the more limiting case of a recirculation pump seizure which is discussed in <Section 15.3.3> and in GESTAR (Reference 1).

15.3.4.1 Identification of Causes and Frequency Classification

The breaking of the shaft of a recirculation pump is considered 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 for both single and two loop operation.

Refer to <Chapter 5> for specific mechanical considerations and <Chapter 8> for electrical aspects.

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

15.3.4.1.1 Identification of Causes

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

15.3.4.1.2 Frequency Classification

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

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

15.3.4.2.1 Sequence of Events

A postulated instantaneous break of the recirculation pump motor shaft will cause the core flow to decrease rapidly resulting in water level swell in the reactor vessel. When the vessel water level reaches the high water level setpoint (L8), reactor scram, main turbine trip and feedwater pump trip will be initiated. Subsequently, 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 Identification of Operator Actions

The operator should ascertain that the reactor scrams resulting from reactor water level swell. The operator should regain control of reactor water level through RCIC operation or by restart of a feedwater pump; and he should monitor reactor water level and pressure after shutdown.

15.3.4.2.2 Systems Operation

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

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

15.3.4.2.3 The Effect of Single Failures and Operator Errors

Effects of single failures in the high vessel level (L8) trip are similar to the considerations in <Section 15.3.1.2.3.b>.

Assumption of single active component failure (SACF) or SOE in other equipment has been examined; it was concluded 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 <u>Core and System Performance</u>

The severity of this pump shaft break event is bounded by the pump seizure event described in <Section 15.3.3>. Since this event is less limiting than <Section 15.3.3>, only qualitative evaluation is provided.

15.3.4.3.1 Qualitative Results

If this extremely unlikely event occurs, core coolant flow will drop rapidly. The level swell produces a reactor scram and trip of the main and feedwater turbines. 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 <Section 15.3.3>. 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 is slower than 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 activation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in <Section 15.2.4.5>. Therefore, the radiological exposures noted in <Section 15.2.4.5> cover the consequences of this event.

15.3.5 REFERENCES FOR SECTION 15.3

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

TABLE 15.3-1

SEQUENCE OF EVENTS FOR <FIGURE 15.3-1>

Time-sec	Event
0	Trip of one recirculation pump initiated.
5.6	Jet pump diffuser flow reverses in the tripped loop.
40	Core flow and power level stabilize at new equilibrium conditions.

TABLE 15.3-2

SEQUENCE OF EVENTS FOR <FIGURE 15.3-2>

Time-sec	Event
0	Trip of both recirculation pumps initiated.
4.1	Vessel water level (L8) trip initiates scram, turbine trip and feedwater pump trip.
4.2	Turbine trip initiates bypass operation.
6.6	Safety/relief valves open due to high pressure.
11.6	Safety/relief valves close.
25.0	Vessel water level (L2) setpoint reached.
55	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.3-3

SEQUENCE OF EVENTS FOR <FIGURE 15.3-3>

Time-sec	Event
0	Initiate fast closure of one main recirculation valve.
1.5	Jet pump diffuser flow reverses in the affected loop.
30	Core flow and power approach new equilibrium conditions.

TABLE 15.3-4

SEQUENCE OF EVENTS FOR <FIGURE 15.3-4>

Time-sec	Event
0	Initiate fast closure of both main recirculation valves.
6.5	Vessel level (L8) trip initiates scram and turbine trip.
6.5	Feedwater pumps tripped off.
6.6	Turbine trip initiates bypass operation.
28.1	Vessel water level reaches Level 2 setpoint.
58.1	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.3-5

SEQUENCE OF EVENTS FOR <FIGURE 15.3-5>

Time-sec	Event
0	Single pump seizure initiated.
0.6	Jet pump diffuser flow reverses in seized loop.
3.4	Vessel level (L8) trip initiates scram.
3.4	Vessel level (L8) trip initiates turbine trip.
3.4	Feedwater pumps are tripped off.
3.5	Turbine trip initiates bypass operation.
3.5	Turbine trip initiates recirculation pumps trip.
6.1	Safety/relief valves open due to high pressure.
11.3	Safety/relief valves close.
23.2	Main bypass valves close to regain pressure regulator control.
24.5	Vessel water level reaches Level 2 setpoint.
54.5 (est.)	HPCS/RCIC flow enters the vessel (not simulated).

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15.4.1 ROD WITHDRAWAL ERROR - LOW POWER

This transient was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed transients. This transient is less severe when initiated from the uprate power level and results in a slightly lower change in MCPR.

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

a. Initial Control Rod Removal Or Withdrawal

During refueling operations safety system interlocks provide backup to procedural controls to ensure inadvertent criticality does not occur because a control rod has been removed or is withdrawn in coincidence with another control rod.

b. Fuel Insertion With Control Rod Withdrawn

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods 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.

c. Second Control Rod Removal or Withdrawal

When the platform is not over the core (or fuel is not on the hoist) and the mode switch is in the "refuel" position, only one control rod can be withdrawn. Any attempt to withdraw a second rod results in a rod block by 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.

d. Control Rod Removal Without Fuel Removal

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

e. Identification of Operator Actions

No operator actions are required to preclude this event since the plant design, as discussed above, prevents its occurrence. If refueling interlock(s) are inoperable (other than the one-rod-out interlock), this event is precluded by the insertion of a continuous rod withdrawal block, to replace the conditional block provided by the interlocks.

f. Effect of Single Failure and Operator Errors

If any one of the operations involved in initial failure or error, is followed by any other single equipment failure (SEF) or single operator error (SOE), the necessary safety actions (e.g., scram or refueling interlock rod block) are automatically taken 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. The withdrawal of the highest worth control rod during refueling will not result in criticality. This is verified experimentally by performing shutdown margin checks. (See <Section 4.3.2> for a description of the methods and results of the shutdown margin analysis.) Additional reactivity insertion is precluded by interlocks <Section 7.6>. As a result, no radioactive material is ever released from the fuel, making it unnecessary to assess any radiological consequences.

No mathematical analysis is required for this event. Input parameters, initial conditions and consideration of uncertainties are not applicable.

15.4.1.1.4 Barrier Performance

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

15.4.1.1.5 Radiological Consequences

An evaluation of 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 this event is considered low enough to warrant its being categorized as an infrequent incident. The probability of further single failure postulated for this event is considerably lower because it is contingent upon the simultaneous failure of two redundant inputs to the rod control and information system (RC&IS), concurrent with a high worth rod, out-of-sequence rod selection, plus operator non-acknowledgement 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

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

Out-of-sequence continuous control rod withdrawal errors during reactor startup are precluded by the RC&IS. The RC&IS 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. With these RC&IS interlocks, there is no basis for a continuous out-of-sequence control rod withdrawal error in the startup and low power range. See <Section 15.4.2> for description of continuous control rod withdrawal above the preset power level. The bank position mode of the RC&IS is described in (Reference 1).

15.4.1.2.2.2 Identification of Operator Actions

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

15.4.1.2.2.3 Effects of Single Failure and Operator Errors

If any one of the operations involved in initial failure or error, is followed by another single active component failure (SACF) or single operator error (SOE), the necessary safety actions [e.g., rod pattern controller (a subsystem of RC&IS) generated block] are automatically taken prior to any limit violation. Refer to <Appendix 15A> for details.

15.4.1.2.3 Core and System Performance

The performance of the RC&IS prevents erroneous selection and withdrawal of an out-of-sequence control rod. Thus, core and system performance is not affected by such an operator error.

No mathematical analysis is required for this event. Input parameters, initial conditions and consideration of uncertainties are not applicable.

15.4.1.2.4 Barrier Performance

An evaluation of 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 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

This transient was performed as part of initial cycle analyses supporting PNPP operation in various operating modes and/or with equipment out-of-service results of which are presented in the following Chapter 15 appendices:

- <Appendix 15D> Partial Feedwater Heating Operation Analysis
- <Appendix 15E> Maximum Extended Operating Domain Analysis
- <Appendix 15F> Recirculation System Single-Loop Operation Analysis

For BWR 6's the rod withdrawal error at power event is analyzed generically or may be analyzed on a cycle-specific basis. The applicability of the generic analysis is reverified when the power operating level is increased, and as new fuel designs, methodologies, or correlations are developed, e.g., GE11 fuel type, GEXL-PLUS, etc. If the generic analysis cannot be confirmed, then a cycle-specific analysis is performed until an adequate database exists to perform generic analyses using methods previously approved by the NRC. Cycle-specific evaluations are presented in <Appendix 15B> of this chapter.

15.4.2.1 Identification of Causes and Frequency Classification

15.4.2.1.1 Identification of Causes

The rod withdrawal error (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 (RC&IS) blocks further withdrawal.

15.4.2.1.2 Frequency Classification

The frequency of occurrence for the RWE is considered 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 a majority of RWEs would be very mild.

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

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 continuously until the RWL inhibits further withdrawal. (The RWL utilizes rod position indications of the selected rod as input.)

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 (ESF) is required during this event.

15.4.2.2.3 The Effect of Single Failures and Operator Errors

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 RC&IS system. The RC&IS system is designed to be single failure proof; therefore, termination of this transient is assured <Appendix 15A>.

15.4.2.3 Core and System Performance

15.4.2.3.1 Mathematical Model

The consequences of an RWE are calculated utilizing a three dimensional, coupled-nuclear-thermal hydraulics computer program (Reference 2). 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 time for reactivity insertion is greater than the fuel thermal time constant and core-hydraulic transport times, so that the steady-state assumption is adequate.

15.4.2.3.2 Input Parameters and Initial Conditions

The reactor core is assumed to be on MCPR and MLHGR technical specification limits prior to RWE initiation. A statistical analysis of the rod withdrawal error (Reference 9) and (Reference 10) initiated from a wide range of operating conditions (exposure, power, flow, rod patterns, xenon conditions, etc.) has been performed, establishing allowable rod withdrawal increments applicable to all BWR/6 plants. These rod withdrawal increments were determined such that the ΔCPR (change in critical power ratio) for rod withdrawal errors initiated from the technical specification operation limits and mitigated by the RWL system withdrawal restrictions, provides a 95% probability at the 95% confidence level that any randomly occurring RWE will not result in violation of the MCPR (minimum critical power ratio) safety limit <Table 15.0-1>. MCPR was verified to be the limiting thermal performance parameter and therefore was used to establish allowable withdrawal increments. The 1% plastic strain limit on the clad was always a less limiting parameter.

15.4.2.3.3 Results

The calculated results demonstrate that, should a rod or gang be withdrawn a distance equal to the allowable rod withdrawal increment, there exists a 95% probability at the 95% confidence level that the MCPR safety limit will not be violated. Furthermore, the peak LHGR will be substantially less than that calculated to yield 1% plastic strain in the fuel clad.

These results of the generic analyses in (Reference 10) show that a control rod or gang can be withdrawn in increments of 12 in. at power levels ranging from 70-100% of rated, and 24 in. at power levels ranging from 20-70%. See <Section 15.4.1.2> for RWE's below 20% reactor power. The 20% and 70% reactor core power levels correspond to the Low Power Setpoint (LPSP) and High Power Setpoint (HPSP) of the RWL.

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 gangs improperly selected and withdrawn. Because of the near-infinite combinations of control patterns and reactor states, all possible states cannot be analyzed. However, because only high worth gangs were included in the statistical analysis, 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 gross core characteristics. Typically, an increase in total core power for RWE's 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.

15.4.3 CONTROL ROD MALOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

This event is covered by evaluations cited in <Section 15.4.1> and <Section 15.4.2>.

15.4.4 ABNORMAL STARTUP OF IDLE RECIRCULATION PUMP

This transient was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed transients.

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.

a. Normal Restart of Recirculation Pump at Power

This transient is categorized as an incident of moderate frequency.

b. 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 15.4-1>.

15.4.4.2.1.1 Identification of Operator Actions

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

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

b. Determine that the idle recirculation pump suction and discharge block valves are open and that the flow control valve in the idle loop is at minimum position. If not, place them in this configuration.

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

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

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

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

NOTE: The time to do the above work is approximately 1/2 hour.

15.4.4.2.2 Systems Operation

This event assumes and takes credit for normal functioning of plant instrumentation and controls. No protection systems action is anticipated. No ESF 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 <Appendix 15A>.

15.4.4.3 Core and System Performance

15.4.4.3.1 Mathematical Model

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

15.4.4.3.2 Input Parameters and Initial Conditions

This analysis has been performed unless otherwise noted with plant conditions tabulated in <Table 15.0-1>.

One recirculation loop is idle and filled with cold water (100°F). (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 MEOD and thermal limit analysis assume a 50°F delta temperature limit between the active and the idle recirculation loops. This limit ensures not only thermal stresses but also thermal limits are not exceeded during an idle loop start event) (Reference 12).

The active recirculation loop is operating with the flow control valve position that produces about 85 percent of normal rated jet pump diffuser flow in the active jet pumps.

The core is receiving 33 percent of its normal rated flow. The remainder of the coolant flows in the reverse direction through the inactive jet pumps.

The idle recirculation pump suction and discharge block valves are open and the recirculation flow control valve is closed to its minimum open position. (Normal procedure requires leaving an idle loop in this condition to maintain the loop temperature within the required limits for restart.)

15.4.4.3.3 Results

The transient response to the incorrect startup of a cold, idle recirculation loop is shown in <Figure 15.4-1>. Shortly after the pump begins to move, 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 is produced as the colder, increasing core flow reduces void volume. Surface heat flux follows the slower response of the fuel and peaks at 81 percent of rated before decreasing after the cold water is pumped out of the loop at about 19 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.

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. 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 <Figure 15.4-1>.

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

This transient was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed transients.

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.

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

a. Fast Opening of One Recirculation Flow Control Valve

<Table 15.4-4> lists the sequence of events for <Figure 15.4-2>.

b. Fast Opening to Two Recirculation Flow Control Valves

<Table 15.4-5> lists the sequence of events of <Figure 15.4-3>.

c. Identification of Operator Actions

Initial action by the operator should include:

1. Transfer flow control to manual and reduce flow to minimum.

2. Identify cause of failure.

Reactor pressure will be controlled as required, depending on whether a restart or cooldown is planned. In general, the corrective action would be to hold reactor pressure and condenser vacuum for restart after the malfunctioning flow controller has been repaired. The following is the sequence of operator actions expected during the course of the event, assuming restart. The operator should:

1. Observe that all rods are in.

2. Check the reactor water level and maintain above low Level (L1) trip to prevent MSIVs from isolating.

3. Switch the reactor mode switch to the “startup” position.

4. Continue to maintain vacuum and turbine seals.

5. Transfer the recirculation flow controller to the manual position and reduce setpoint to zero.

6. Survey maintenance requirements and complete the scram report.

7. Monitor the turbine coast down and auxiliary systems.

8. Establish a restart of the reactor per the normal procedure.

   NOTE: Time required from first trouble alarm to restart would be approximately 1 hour.

15.4.5.2.2 Systems Operation

The analysis of this transient assumes and takes credit for normal functioning of plant instrumentation and controls, and the reactor protection system. Operation of engineered safeguards is not expected.

15.4.5.2.3 The Effect of Single Failures and Operator Errors

Both of these transients lead 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 <Appendix 15A>. 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.

15.4.5.3 Core and System Performance

15.4.5.3.1 Mathematical Model

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

15.4.5.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in <Table 15.0-1>.

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 NB rated power and 33 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 flow control valve for a loop controller failure is limited by hydraulics to 30 percent per second.

15.4.5.3.3 Results

a. Fast Opening of One Recirculation Flow Control Valve

   <Figure 15.4-2> 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 1.4 seconds. The peak neutron flux reached was 215 percent of NB rated value, while the accompanying average fuel surface heat flux reaches 71 percent of NB rated at approximately 2.3 seconds. MCPR remains considerably above the safety limit and the average fuel temperature increases only 104°F. Reactor pressure is discussed in <Section 15.4.5.4>.

b. Fast Opening of Two Recirculation Flow Control Valves

<Figure 15.4-3> illustrates the failure where both recirculation loop flow control 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 1.9 seconds, peaking at 149 percent of NB rated while the average surface heat flux reaches 66 percent of NB rated at approximately 2.6 seconds. MCPR remains considerably above the safety limit and average fuel center temperature increases 78°F.

15.4.5.3.4 Consideration of Uncertainties

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.

15.4.5.4 <u>Barrier Performance</u>

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 15.4-2> and therefore represents no threat to the RCPB.

15.4.5.4.2 Fast Opening of Two Recirculation Flow Control Valves

This transient results in a very slight increase in reactor vessel pressure as shown in <Figure 15.4-3> and therefore represents no threat to the RCPB.

15.4.5.5 Radiological Consequences

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

15.4.6 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTIONS

Not applicable to BWRs.

15.4.7 MISPLACED BUNDLE ACCIDENT

Analysis of the misplaced bundle accident (fuel loading error) for the 3,758 MWt core considered only the mislocated bundle accident. For reload cores, refer to <Appendix 15B>, Reload Safety Analysis, for evaluation of the mislocated bundle accident and the misoriented bundle accident.

15.4.7.1 Identification of Causes and Frequency Classification

15.4.7.1.1 Identification of Causes

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

15.4.7.1.2 Frequency of Occurrence

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

15.4.7.2 Sequence of Events and Systems Operation

The postulated sequence of events for the misplaced bundle accident (MBA) is 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 Effect of Single Failures 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. 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

A three-dimensional BWR simulator model is used to calculate the core performance resulting from this event.

15.4.7.3.2 Input Parameters and Initial Conditions

The initial core consists of bundles with average enrichments that are high, medium or low with correspondingly different gadolinia concentrations. The fuel bundle loading error with the severest consequences occurs at beginning-of-cycle (BOC) when a low-enriched bundle (which should be loaded at the periphery) is interchanged with a high-enriched bundle located adjacent to an LPRM which is predicted to have the highest LHGR and/or lowest CPR in the core. After the loading error is made and has gone undetected, it is assumed for purposes of conservatism that the operator uses a control rod pattern that places the limiting bundle in the four bundle array containing the misplaced bundle, on design thermal limits as recorded by the LPRM.

As a result of loading the low-enriched bundle in an improper location, the reading of the adjacent LPRM decreases. Consequently, because there are no instruments in the 3 mirror images of this four bundle array, the operator believes these arrays are operating at the same power as the instrumented one, when in fact they are not (since no loading error occurred in these quadrants). As a result of placing the instrumented array on limits, the 3 mirror-image arrays exceed the design limit. By replacing the high-enriched bundle with the greatest power peaking, by the low-enriched bundle, it is assured that the difference in power peaking between the instrumented and the non-instrumented arrays is maximum, or rather, that the ΔCPR and ΔLHGR is the upper bound for this error.

Other input parameters assumed are given in <Table 15.4-7> and <Figure 15.4-4>.

15.4.7.3.3 Results

Results of analyzing the worst fuel bundle loading error are reported in <Table 15.4-8>. As can be seen, MCPR remains well above the point where boiling transition would be expected to occur, and the MLHGR does not exceed the 1 percent plastic strain limit for the clad. Therefore, no fuel damage occurs as a result of this event.

15.4.7.3.4 Consideration of Uncertainties

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. This assures that the minimum CPR and maximum LHGR are conservatively bounded.

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 reactor pressure would occur.

15.4.7.5 Radiological Consequences

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

15.4.8 SPECTRUM OF ROD EJECTION ACCIDENTS

Not applicable to BWRs.

The BWR has precluded this event by incorporating into its design mechanical equipment which restricts any movement of the control rod drive system assemblies. The control rod drive housing support assemblies are described in <Chapter 4>.

15.4.9 CONTROL ROD DROP ACCIDENT (CRDA)

Certain limiting safety analyses are reperformed each operating cycle to determine and/or verify safety margins. The methods, input conditions, and results for the current cycle for the control rod drop accident are presented in <Appendix 15B> - Reload Safety Analysis.

The NRC approved generic bounding Control Rod Drop Accident (CRDA) analysis for Banked Position Withdrawal Sequence (BPWS) plants (such as PNPP) described in GESTAR (Reference 9) is applied and therefore, the CRDA is not reanalyzed on a reload - specific basis. As new fuel designs, methodologies or correlations are developed (e.g., GEMINI methods) the applicability of the generic analysis is reverified. For the second cycle the CRDA was reverified due to GEMINI methods being used. The impact of GEMINI methods on the results of the generic analysis is negligible. Also, the effect of increasing core thermal power to 3,758 MWt on the generic CRDA analysis is negligible due to the considerable margin present in the generic analysis.

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, within the constraints of the banked position rod control and information system (RC&IS), drops from the fully inserted or intermediate position in the core. The high worth rod becomes decoupled from its drive mechanism. The mechanism is withdrawn but the decoupled control rod is assumed to be stuck inplace. At a later moment, the control rod suddenly falls free and drops to the control rod drive position. This results in a localized power excursion.

A more detailed discussion is given in (Reference 3).

It is important to note that, because of Perry's rod pattern control system (RPCS) <Section 7.6.1.5>, the consequences of a rod drop become insignificant. The fission product release would be so small that no reactor isolation is expected to occur. This is based on analysis of similar control systems as described in (Reference 1). Therefore, due to RPCS a control rod drop accident resulting in radiological consequences even remotely approaching <10 CFR 50.67> guidelines is not considered credible at Perry.

15.4.9.1.2 Frequency Classification

The CRDA is classified 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 Systems 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 control and information system (RC&IS) 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 RC&IS will only allow bank position mode rod withdrawals or insertions.

The RC&IS uses redundant input to provide absolute assurance on control rod drive position. If either of the diverse inputs were to fail, the other would provide the necessary information.

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

15.4.9.2.3 Effect of Single Failures and Operator Errors

Systems mitigating the consequences of this event are RC&IS and APRM scram. The RC&IS is designed as a redundant system and therefore provides single failure protection. The APRM scram system is also designed to 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 analytical methods, assumptions and conditions for evaluating the control rod drop accident are described in detail in (Reference 3), (Reference 4), (Reference 5), and (Reference 11). They are considered to provide a realistic yet conservative assessment of the associated consequences. The bounding analyses are presented in (Reference 6). Compliance checks are made to verify that the maximum rod worth does not exceed 1 percent Δk, so that the maximum local core power increase cannot cause more than the number of fuel rod failures assumed in the following radiological evaluations.

15.4.9.3.2 Input Parameters and Initial Conditions

The core at the time of rod drop is assumed to be at the point in 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). Removing xenon, which has a high neutron absorption cross section, increases the fraction of neutrons absorbed, or worth of the control rods. The 50 percent control rod density which nominally occurs at the hot startup condition, ensures that withdrawal of a rod results in the maximum increment of reactivity.

<Table 15.4-10> provides other input parameters and initial conditions for the CRDA analysis.

15.4.9.3.3 Results

Control rod worth calculations were performed for a representative, high energy, equilibrium GNF2 core design. All control rod worths were less than the confirmation criterion of 1 percent Δk.

Peak fuel enthalpy is the parameter used to determine the severity of a transient and the onset of fuel pin failure. As reference points, the following design and fuel failure criteria have been established by General Electric:

Enthalpy = 170 calories per gram, cladding failure threshold
Enthalpy = 280 calories per gram, specific energy design limit
Enthalpy = 425 calories per gram, prompt fuel dispersal threshold

A conservative number of pins that could reach the cladding failure threshold of 170 calories per gram is less than 1,200 for all plant operating conditions. The PNPP radiological evaluations are conservatively based on an assumed failure of 1,376 fuel pins for GNF2 fuel.

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 reactor temperature or pressure.

15.4.9.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

a. The first analysis assumes the accident occurs at a low reactor power level with the mechanical vacuum pumps in operation or at any power level with a coincident loss of offsite power. This is referred to as Scenario 1.

b. The second analysis assumes the accident occurs at a higher reactor power level with the Steam Jet Air Ejectors in operation such that the condenser gases are processed through the Offgas filtration system. This is referred to as Scenario 2.

Each analysis is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet <10 CFR 50.67> guidelines and is referred to as the “Design Basis Analysis.”

The exposures were calculated using the computer code RADTRAD (Reference 13).

A schematic of the leakage path for each analysis is shown in <Figure 15.4-5>.

15.4.9.5.1 Scenario 1 Analysis

The Scenario 1 analysis is based on the NRC’s Standard Review Plan 15.4.9 (Reference 8) where a loss of offsite power (LOOP) occurs coincident with the CRDA. The LOOP condition results in the automatic closure of the MSIVs, thereby stopping the transport of the fission products to the condenser. Radioactive release to the environment follows due to the condenser leakage. Alternatively, if the mechanical vacuum pumps are in operation (i.e., low reactor power level) at the time of the CRDA and a LOOP does not occur, the high radiation condition will be detected by the MSLRMs causing an automatic isolation of the operating mechanical vacuum pump line. Once again, the radioactive release to the environment will occur as a function of condenser leakage. Specific parametric values used in the evaluation are presented in <Table 15.4-12>.

15.4.9.5.1.1 Fission Product Release from Fuel

The failure of 1,376 fuel pins 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 2,842°C) is estimated to be 0.0077.

Fuel reaching melt conditions is assumed to release 100 percent of the noble gas inventory, 50 percent of the iodine inventory, and 25 percent of the alkali metals (Cs, Rb). The remaining fuel in the damaged pins is assumed to release 10 percent of both the noble gas and iodine inventories, and 12 percent of the alkali metals.

A maximum equilibrium inventory of fission products in the core is based on 1,000 days of continuous operation at 3,833 MWt. No delay time is considered between departure from the above power condition and the initiation of the accident.

15.4.9.5.1.2 Fission Product Transport to the Environment

The transport pathway is shown in <Figure 15.4-5> and consists of carryover with steam to the turbine condenser and leakage from the condenser to the environment. No credit is taken for decay during retention in 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 remaining radionuclides are assumed to be carried to the condenser.

Of the activity reaching the condenser, 100 percent of the noble gases, 10 percent of the iodines, and 1 percent of the particulate radionuclides (due to partitioning and plateout) 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 a period of 24 hours. Radioactive decay is accounted for during residence in the condenser, however, it is neglected after release to the environment.

The activity airborne in the condenser is presented in <Table 15.4-13>.

15.4.9.5.1.3 Results

The calculated exposures from the Scenario 1 design basis analysis are presented in <Table 15.4-14> and are well within the guidelines of <10 CFR 50.67>.

15.4.9.5.2 Scenario 2 Analysis

The Scenario 2 analysis is based on the NRC’s Standard Review Plan 15.4.9 (Reference 8) with the exception that a LOOP is not assumed to occur. The specific models and assumptions used for the evaluation are described in (Reference 11). Specific parametric values used in the evaluation are presented in <Table 15.4-12>.

15.4.9.5.2.1 Fission Product Release from Fuel

The fission product release from fuel is the same as presented in <Section 15.4.9.5.1.1> for Scenario 1.

15.4.9.5.2.2 Fission Product Transport to the Environment

The transport pathway is shown in <Figure 15.4-5> and consists of carryover with steam to the turbine condenser and leakage from the condenser to the environment via the Offgas System.

Of the activity released from the fuel, 100 percent of the noble gases, 10 percent of the iodines, and 1 percent of the remaining radionuclides are assumed to be carried to the condenser.

Of the activity reaching the condenser, 100 percent of the noble gases, 10 percent of the iodines, and 1 percent of the particulate radionuclides remain airborne. The activity airborne in the condenser is processed by the Offgas filtration system prior to release to the environment. Only the noble gases are released from the offgas system.

Radioactive decay is accounted for during residence in the condenser and the Offgas System, however, it is neglected after release to the environment.

The activity airborne in the condenser is presented in <Table 15.4-13>.

15.4.9.5.2.3 Results

The calculated exposures from the Scenario 2 design basis analysis are presented in <Table 15.4-16> and are well within the guidelines of <10 CFR 50.67>.

15.4.10 REFERENCES FOR SECTION 15.4

1. Paone, C. J., "Banked Position Withdrawal Sequence," January 1977, (NEDO-21231).

2. Woolley, J. A., "Three Dimensional Boiling Water Reactor Core Simulator," May 1976, (NEDO-20953).

3. R. C. Stirn, et al., "Rod Drop Accident Analysis for Large BWRs," March 1972, (NEDO-10527).

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

5. R. C. Stirn, et al., "Rod Drop Accident Analysis for Large BWRs," January 1973, (NEDO-10527 Supplement 2).

6. "GE BWR Generic Reload Application for 8 x 8 Fuel" (NEDO-20360 Supplement 3 to Revision 1).

7. (Deleted)

8. USNRC Standard Review Plan, <NUREG-75/087>, Washington, D.C., Revision 1.

9. General Electric Company “General Electric Standard Application for Reactor Fuel” including the “United States Supplement,” NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision).

10. GESSAR II BWR/6 238 Nuclear Island Design, 22A7007, Revision 21, Appendix 15B - BWR/6 Generic Rod Withdrawal Error Analysis (or GE report EAS 69-0687).

11. General Electric Company “Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor” NEDO-31400, May 1987.

12. GE Services Information Letter, SIL No. 517 Supplement 1, “Analysis Basis for Idle Recirculation Loop Startup”.

13. S.L. Humphreys et al., “RADTRAD: A Simplified Model for Radionuclide Transport And Removal and Dose Estimation,” NUREG/CR-6604, USNRC, April 1998.

TABLE 15.4-1

SEQUENCE OF EVENTS - RWE IN POWER RANGE

Elapsed Time	Event
0	Core is assumed to be operating at rated conditions.
0	Operator selects and withdraws the maximum worth single control rod or rod gang.
~1 sec	The total core power and the local power in the vicinity of the control rod increase.
~6 sec	The RWL mode blocks withdrawal.
~25 sec	Reactor core stabilizes at slightly higher core power level.
~45 sec	Operator re-inserts control rod to reduce core power level.
~60 sec	Core stabilizes at rated conditions.

<TABLE 15.4-2>

DELETED

TABLE 15.4-3

SEQUENCE OF EVENTS FOR <FIGURE 15.4-1>

Time-sec	Event
0	Start pump motor.
0.73	Jet pump diffuser flows on started pump side become positive.
3.1	Pump motor at full speed and drive flow at about 21% of rated.
17.6 (est.)	Last of cold water leaves recirculation drive loop.
18.0 (est.)	Peak value of core inlet subcooling.
50	Reactor variables settle into new steady-state.

TABLE 15.4-4

SEQUENCE OF EVENTS FOR <FIGURE 15.4-2>

Time-sec	Event
0	Initiate failure of single loop control.
1.4	Reactor APRM high flux scram trip initiated.
3.4 (est.)	Turbine control valves start to close upon falling turbine pressure.
12.0 (est.)	Turbine control valves closed. Turbine pressure below pressure regulator setpoints.
14.0 (est.)	Vessel water level reaches Level 2 (L2) setpoint.
14.0 (est.)	Recirculation pump drive motors trip due to L2.
>100	Reactor variables settle into new steady-state.

TABLE 15.4-5

SEQUENCE OF EVENTS FOR <FIGURE 15.4-3>

Time-sec	Event
0	Initiate failure of master controller.
1.9	Reactor APRM high flux scram trip initiated.
3.7 (est.)	Turbine control valves start to close upon falling turbine pressure.
9.0 (est.)	Turbine control valves closed. Turbine pressure below pressure regulator setpoints.
9.0	Vessel water level reaches Level 2 (L2) setpoint.
9.0	Recirculation pump drive motors trip due to L2.
>100 (est.)	Reactor variables settle into new steady-state.

TABLE 15.4-6

SEQUENCE OF EVENTS FOR MISPLACED BUNDLE ACCIDENT

1. During core loading operation, bundle is placed in the wrong location.

2. Subsequently, the bundle intended for this location is placed in the location 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.

TABLE 15.4-7

INPUT PARAMETERS AND INITIAL CONDITIONS
FOR FUEL BUNDLE LOADING ERROR

(3758 MWt Core)

1.	Power, % rated	100
2.	Flow, % rated	91.7
3.	MCPR	1.378
4.	MLHGR, kW/ft	10.01
5.	Average core exposure, MWd/t	11,900
6.	Location of minimum CPR bundle	(12, 15)
7.	Location of maximum LHGR bundle	(8, 13)
8.	Control Rod Pattern	<Figure 15.4-4>

NOTE:

(1) Core conditions are assumed to be normal for a hot, operating core near EOC.

TABLE 15.4-8

MISPLACED BUNDLE ANALYSIS

Bundle (11, 13) Replaced with Fresh Bundle

MCPR	MCPR With Misplaced Bundle	ΔCPR
1.378	1.253	-0.125

MLHGR	MLHGR With Misplaced Bundle	ΔLHGR
10.01 kW/ft	10.92 kW/ft	0.91 kW/ft

TABLE 15.4-9

SEQUENCE OF EVENTS FOR ROD DROP ACCIDENT

Approximate Elapsed Time	Event
	Reactor is operating at 50% rod density pattern.
	Maximum worth control rod blade becomes decoupled from the CRD.
	Operator selects and withdraws the control rod drive of the decoupled rod either individually or along with other control rods assigned to the RC&IS 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 the 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

1.	Reactor power, % rated	100
2.	Reactor flow, % rated	100
3.	Core average exposure, MWd/t	8.0
4.	Control rod density, percent	~50
5.	Average fuel temperature, °C	286
6.	Average moderator temperature, °C	286
7.	Xenon state	No xenon

TABLE 15.4-11

INCREMENTAL WORTH OF THE MOST REACTIVE ROD USING BPWS[1]

Control Rod Group[2]	Banked At Notch	Control Rod (I, J)	Drops From-To	Increase In k_{eff}
10	4	(30, 55)	0→8	0.0010
10	8	(30, 55)	0→12	0.0022
10	12	(30, 55)	0→48	0.0042
10	48	(30, 55)	0→48	0.0005

NOTES:

(1) Sequence A, Rod Groups 1-4 withdrawn.
(2) For definition of rod groups, see (Reference 5).

TABLE 15.4-12

CONTROL ROD DROP ACCIDENT
EVALUATION PARAMETERS

	Scenario 1 Assumptions	Scenario 2 Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents.		
A. Power level	3,833 MWt	3,833 MWt
B. Burnup	N/A	N/A
C. Fuel damaged	1,376 rods[(1)]	1,376 rods
D. Release of activity by nuclide	<Table 15.4-13>	N/A
E. Iodine fractions, %		
(1) Organic	0.15	0.15
(2) Elemental	4.85	4.85
(3) Particulate	95	95
F. Reactor coolant activity before the accident.	N/A	N/A
II. Data and assumptions used to estimate activity released.		
A. Condenser leak rate (%/day)	1.0	N/A
B. Turbine building leak rate (%/day)	N/A	N/A
C. Valve closure time (sec)	N/A	N/A
D. Adsorption and filtration efficiencies		
(1) Organic iodine	N/A	N/A
(2) Elemental iodine	N/A	N/A
(3) Particulate iodine	N/A	N/A
(4) Particulate fission products	N/A	N/A
E. Recirculation system parameters		
(1) Flow rate	N/A	N/A
(2) Mixing efficiency	N/A	N/A
(3) Filter efficiency	N/A	N/A
F. Containment spray parameters (flow rate, drop size, etc.)	N/A	N/A
G. Containment volumes	N/A	N/A

TABLE 15.4-12 (Continued)

	Scenario 1 Assumptions	Scenario 2 Assumptions
H. All other pertinent data and assumptions.	None	Holdup Time in Offgas Pretreatment System a. Xe = 4.37 days b. Kr = 5.5 hours
III. Dispersion Data		
A. Boundary and LPZ distances (m)	863/4,002	863/4,002
B. X/Q's (sec/m^3) for time intervals of:		
(1) 0-2 hr - SB/LPZ	4.3E-4/4.8E-5	4.3E-4/4.8E-5
(2) 2-8 hr - LPZ	4.8E-5	4.8E-5
(3) 8-24 hr - LPZ	3.3E-5	3.3E-5
(4) 1-4 days - LPZ	1.4E-5	1.4E-5
(5) 4-30 days - LPZ	4.1E-6	4.1E-6
C. Control Room X/Q (sec/m^3) for time intervals of:		
(1) 0-2 hours	3.5E-4	3.5E-4
(2) 2-8 hours	3.5E-4	3.5E-4
(3) 8-24 hours	2.1E-4	2.1E-4
(4) 24-96 hours	1.1E-4	1.1E-4
(5) 96-720 hours	5.75E-5	5.75E-5
IV. Data and assumptions used to estimate dose to the Control Room		
Volume (ft^3)	3.71E+5	3.71E+5
Flow Rate – unfiltered inleakage (cfm)	6600	6600
Recirculation flow (cfm)	0	0
Recirculation filter efficiencies (%)		
Particulate	0	0
Elemental and organic Iodine	0	0

TABLE 15.4-12 (Continued)

		Scenario 1 Assumptions	Scenario 2 Assumptions
V.	Dose Data		
A.	Method of dose calculation	<Section 15.0.3.5>	<Section 15.0.3.5>
B.	Dose conversion assumptions	<Section 15.0.3.5>	<Section 15.0.3.5>
C.	Peak activity concentrations in condenser	<Table 15.4-13>	<Table 15.4-13>
D.	Doses	<Table 15.4-14>	<Table 15.4-16>

NOTE:

(1) GNF2 Fuel.

TABLE 15.4-13

CONTROL ROD DROP ACCIDENT - SCENARIO 1
(DESIGN BASIS ANALYSIS)
ACTIVITY AIRBORNE IN CONDENSER (CURIES)[1]

Isotope	Activity Curies
BR-82	3.0E+1
BR-83	5.5E+2
BR-84	9.4E+2
I-128	6.9E+1
I-130	1.7E+2
I-131	4.6E+3
I-132	6.6E+3
I-133	9.3E+3
I-134	1.0E+4
I-135	8.7E+3
CS-132	1.5E-2
CS-134	1.4E+1
CS-134M	3.2E+0
CS-136	4.3E+0
CS-137	8.3E+0
CS-138	9.9E+1
RB-86	1.4E-1
RB-88	3.6E+1
RB-89	4.7E+1
KR-83M	5.7E+4
KR-85	6.6E+3
KR-85M	1.2E+5
KR-87	2.2E+5
KR-88	3.2E+5
XE-129M	3.5E+0
XE-131M	5.3E+3
XE-133	9.3E+5
XE-133M	3.0E+4
XE-135	3.4E+5
XE-135M	1.9E+5
XE-138	7.9E+5

NOTE:

[1] GNF2 Fuel, 1350 EFPD @ 3,833 MWt.

TABLE 15.4-14

CONTROL ROD DROP ACCIDENT - SCENARIO 1
(DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

	TEDE Dose (rem)	Licensing Basis Limit (TEDE, rem)
Exclusion area (863 Meters)	1.61E-1	6.3
Low population zone (4,002 Meters)	1.62E-1	6.3
Control Room	2.63E-1[1]	5

NOTE:

[1] Limiting case presumed no control room isolation or filtration.

<TABLE 15.4-15>

DELETED

TABLE 15.4-16

CONTROL ROD DROP ACCIDENT - SCENARIO 2
(DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

	TEDE Dose (rem)	Licensing Basis Limit (TEDE, rem)
Exclusion area (863 Meters)	1.77E-4[(1)]	6.3
Low population zone (4,002 Meters)	1.87E-4[(1)]	6.3
Control Room	8.28E-5[(2)]	5

NOTES:

(1) Iodine releases are negligible because of their retention in the charcoal beds of the Offgas Treatment System as noted in NEDO-31400, “Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function of the Main Steam Line Radiation Monitor.”
(2) Limiting case presumed no control room isolation or filtration.

15.5 INCREASE IN REACTOR COOLANT INVENTORY

15.5.1 INADVERTENT HPCS STARTUP

This transient was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed transients.

15.5.1.1 Identification of Causes and Frequency Classification

15.5.1.1.1 Identification of Causes

Manual startup of the HPCS system is postulated for this analysis, i.e., 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

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

15.5.1.2.1.1 Identification of Operator Actions

With the recirculation system in either the automatic or manual mode, relatively small changes would be experienced in plant conditions. The operator should, after hearing the alarm that the HPCS has commenced operation, check reactor water level and drywell pressure. If conditions are normal, the operator should shut down the system.

15.5.1.2.2 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, 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 recirculation system is assumed to be in the manual flow control mode of operation.

15.5.1.2.3 The Effect of Single Failures and Operator Errors

Inadvertent operation of the 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 <Section 15.1.3> and <Section 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.

15.5.1.3 Core and System Performance

15.5.1.3.1 Mathematical Model

The nonlinear dynamic model described briefly in <Section 15.1.2.3.1> 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 15.0-1>.

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

Inadvertent startup 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 5.1 percent of rated feedwater flow rate. This flow is nearly 102 percent of 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 remains above the safety limit and 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 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.

15.5.3 BWR TRANSIENTS WHICH INCREASE REACTOR COOLANT INVENTORY

These events are discussed and considered in <Section 15.1> and <Section 15.2>.

TABLE 15.5-1

SEQUENCE OF EVENTS FOR <FIGURE 15.5-1>

Time-sec	Event
0	Initiate HPCS cold water injection.
3	Full flow established for HPCS.
5	Depressurization effect stabilized.

15.6 DECREASE IN REACTOR COOLANT INVENTORY

15.6.1 INADVERTENT SAFETY/RELIEF VALVE OPENING

This event is discussed and analyzed in <Section 15.1.4>.

15.6.2 INSTRUMENT LINE PIPE BREAK

This event involves the postulation of a small steam or liquid line pipe break inside containment. In order to bound the event, it is assumed that a small instrument line, instantaneously and circumferentially, breaks at a location where it may not be isolated and where immediate detection is not automatic or apparent.

Obviously, this event is far less limiting than the postulated events in <Section 15.6.4> "Steam System Piping Break Outside Containment", <Section 15.6.5> "Loss-of-Coolant Accidents – Inside Containment, and <Section 15.6.6> "Feedwater Line Break – Outside Containment". Accordingly, this accident was not reanalyzed for the current reload as it has been determined to be less limiting and bounded by the analyzed accidents described in the previously listed sections.

This postulated event represents the envelope evaluation for small line failure inside containment, relative to sensitivity to detection.

15.6.2.1 Identification of Causes and Frequency Classification

15.6.2.1.1 Identification of Causes

There is no specific event or circumstance identified which results in the failure of an instrument line. These lines are designed to high quality, engineering standards, seismic, and environmental requirements. However, for the purpose of evaluating the consequences of a small line rupture, the failure of an instrument line is assumed to occur.

A circumferential rupture of an instrument line which is connected to the primary coolant system is postulated to occur outside the drywell but inside the containment structure. This failure results in the release of primary system coolant to containment until the reactor is depressurized. This event could also conceivably occur in the drywell. However, the associated effects would not be as significant as those from a failure in the containment.

15.6.2.1.2 Frequency Classification

This event is categorized as a limiting fault.

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

15.6.2.2.1 Sequence of Events

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

15.6.2.2.1.1 Identification of Operator Actions

The operator should isolate the affected instrument line. Depending on which line is broken, the operator should determine whether to continue plant operation until a scheduled shutdown can be made or to proceed with an immediate, orderly plant shutdown.

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

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

b. By annunciation of the control function, either high or low in the main control room.

c. By a half-channel scram if rupture occurred on a reactor protection system instrument line.

d. By a general increase in the area radiation monitor readings.

e. By an increase in the ventilation process radiation monitor readings.

f. By increases in area temperature monitor readings in the containment.

g. Leak detection system actuations.

Upon receiving one or more of the above signals and having made the decision to shut down the plant, the operator should proceed to shut down in an orderly manner.

15.6.2.2.2 Systems Operation

Normal plant instrumentation and controls are assumed to be fully operational during the entire plant transient to ensure positive identification of the break and safe shutdown of the plant. Minimum reactor and plant protection system operations are assumed for the analysis, e.g., minimum ECCS flow, and pool cooling capability. As a consequence of the accident, the reactor is scrammed and the reactor vessel cooled and depressurized over a 5 hour period.

15.6.2.2.3 The Effect of Single Failures and Operator Errors

The initiating event is handled by a protection sequence which can accommodate additional single failures. See <Appendix 15A> for a more detailed discussion of this subject.

15.6.2.3 Core and System Performance

15.6.2.3.1 Qualitative Summary - Results

Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint than the events examined in <Section 15.6.4>, <Section 15.6.5> and <Section 15.6.6>. Consequently instrument line breaks are considered to be bounded specifically by the steam line break, <Section 15.6.4>. Details of this calculation, including those pertinent to core and system performance are discussed in detail in <Section 15.6.4.3>.

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

15.6.2.3.2 Quantitative Results

Instrument line breaks, because of their small size, are substantially less limiting from a core and system performance standpoint than the steam line break outside containment. Similarly, instrument line breaks are considered within the spectrum considered in ECCS performance calculations discussed in detail in <Section 6.3.3>.

Therefore, all information concerning ECCS models employed, input parameters and detailed results for a more limiting (steam line break) event may be found in <Section 6.3>.

15.6.2.3.3 Consideration of Uncertainties

The approach toward conservatively analyzing this event is discussed in detail for a more limiting case in <Section 6.3>.

15.6.2.4 Barrier Performance

15.6.2.4.1 General

The release of primary coolant through the orificed instrument line could result in an increase in containment pressure and the potential for isolation of the normal ventilation system, if operating.

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

a. Shutdown and depressurization initiated at 10 minutes after break occurs.

b. Normal depressurization and cooldown of reactor pressure vessel.

c. Line contains a 1/4-inch diameter flow restricting orifice inside the drywell.

d. Moody critical blowdown flow model (Reference 1) is applicable and flow is critical at the orifice.

The total integrated mass of fluid released into the containment via the break during the blowdown is 25,000 pounds. Of this total, 6,000 pounds flash to steam. Release of this mass of coolant results in a containment pressure which is well below the design pressure.

15.6.2.4.2 Containment Effects

Following the postulated failure of an instrument line in the containment, the containment pressure will rise due to the release of primary system fluid and will continue until the reactor is depressurized. The containment pressure increase is evaluated based on the calculated mass release. The calculation is based on the assumptions outlined above and includes the heat losses to the containment structures that will occur.

15.6.2.5 Radiological Consequences

15.6.2.5.1 Design Basis Analysis

While the NRC has developed a standard review plan (Reference 2) for this event, a specific regulatory guide calculation method has not been issued, and as noted above, instrument line breaks are considered to be bounded specifically by the steam line break <Section 15.6.4>, which releases 141,687 pounds of fluid in a manner that is treated as direct to the environment. For these reasons, only the Realistic Analysis performed for initial plant licensing is provided.

15.6.2.5.2 Realistic Analysis

This Realistic Analysis was performed for initial plant licensing, and is retained for historical information (not updated).

The realistic analysis is based on a realistic but still conservative assessment of this accident. The specific models, assumptions and the program used for evaluation are described in <Section 15.0.3.5>. Specific values of parameters used in the evaluation are presented in <Table 15.6-2>.

15.6.2.5.2.1 Fission Product Release from Fuel

The quantity of activity released as a consequence of reactor scram and vessel depressurization is based in part on measurements during plant shutdowns (Reference 3). These measurements have been used to develop an empirical model which predicts, during the depressurization transient, I-131 releases of 0.42 Ci/bundle at the 50th percentile to 2.14 Ci/bundle at the 95th percentile. For the purpose of this evaluation, the 95th percentile values are used. The release of other iodine isotopes is considered to be proportional to the fission yields, that is

$$I_{132} = \frac{(2.14)(F_r I_{132})}{F_r I_{131}}$$

The activity airborne in the break location structure is presented in <Table 15.6-3>.

15.6.2.5.2.2 Fission Product Release to the Environment

The fission product activity released to the environment as a result of this accident is based upon the methods and assumptions outlined below:

a. The failure of an instrument line results in a relatively small release of activity to the containment over a blowdown period of approximately 5 hours. Therefore, it is conservatively assumed that the total iodine activity airborne inside containment presented in <Table 15.6-3> is instantaneously released to the environs through the containment purge exhaust system.

b. Charcoal filter efficiency for the containment purge exhaust system is conservatively assumed to be 90 percent for iodine.

15.6.2.5.2.3 Results

The calculated inhalation doses for the realistic analysis are 2.03 rem at the exclusion area boundary (863 meters) and 0.249 rem at the low population zone boundary (4,002 meters). These were calculated for initial plant licensing, and are retained for historical information (not updated).

15.6.3 STEAM GENERATOR TUBE FAILURE

This section is not applicable to the BWR.

15.6.4 STEAM SYSTEM PIPING BREAK OUTSIDE CONTAINMENT

The accident was not reanalyzed for the current reload since the original analysis is still applicable.

This event involves the postulation of a large steam line break outside containment. It is assumed that the largest steam line, instantaneously and circumferentially, breaks at a location downstream of the outboard containment isolation valve <Figure 15.6-1>. The plant is designed to immediately detect such an occurrence, initiate isolation of the broken line and actuate the necessary protective features. This postulated event represents the envelope evaluation of steam line failures outside 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 is given in <Table 15.6-6>.

15.6.4.2.1.1 Identification of Operator Actions

Normally the reactor operator will maintain reactor vessel water inventory and, therefore, core cooling with the RCIC system. Without operator action, the RCIC would initiate automatically on low water level following isolation of the main steam supply system (i.e., MSIV closure). The core would be covered throughout the accident and there would be no fuel damage. Without taking credit for the RCIC water makeup capability and assuming HPCS failure, the ADS will automatically initiate on low water level to ensure termination of the accident without fuel damage.

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 restrictor area and finally terminates the mass loss when full closure is reached.

A discussion of plant and reactor protection system action and ESF action is given in <Section 6.3>, <Section 7.2>, <Section 7.3>, and <Section 7.6>.

15.6.4.2.3 The Effect of Single Failures and Operator Errors

The effect of single failures has been considered in analyzing this event. The ECCS aspects are covered in <Section 6.3>. The break detection and isolation considerations are defined in <Section 7.3> and <Section 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 <u>Core and System Performance</u>

Quantitative results (including mathematical models, input parameters and consideration of uncertainties) for this event are given in <Section 6.3>. The temperature and pressure transients resulting 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 6.3> for initial conditions.

15.6.4.3.2 Results

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

Refer to <Section 6.3> for ECCS analysis.

15.6.4.3.3 Consideration of Uncertainties

<Section 6.3> and <Section 7.3> contain discussions of the uncertainties associated with 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.

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

a. The reactor is operating at the power level associated with maximum mass release.

b. Nuclear system pressure is 1,060 psia and remains constant during closure.

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

d. Isolation valves start to close at 0.5 seconds on high flow signal and are fully closed at 5.5 seconds. The analysis conservatively assumes flow through the valves for 6.05 seconds.

e. The Moody critical flow model (Reference 1) is applicable.

f. Level rise time is conservatively assumed to be 1.0 second. Mixture quality is conservatively taken to be a constant 7.0 (steam weight percentage) during mixture flow.

Initially only steam will issue from the broken end of the steam line. The flow in each line is limited by critical flow at the limiter to a maximum of 200 percent of rated flow for each line. Rapid depressurization of the RPV causes the water level to rise resulting in a steam-water mixture flowing from the break until the valves are closed. The total integrated mass leaving the RPV through the steam line break is 141,687 pounds of which 127,376 pounds is liquid and 14,311 pounds is steam.

15.6.4.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

a. The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet <10 CFR 50.67> guidelines. This analysis is referred to as the "design basis analysis."

b. The second is based on assumptions considered to provide a realistic conservative estimate of the radiological consequences. This analysis is referred to as the "realistic analysis."

The exposures were calculated utilizing the computer code RADTRAD (Reference 21).

A schematic of the release path is shown in <Figure 15.6-1>.

15.6.4.5.1 Design Basis Analysis

The design basis analysis is based on NRC Standard Review Plan 15.6.4 and NRC <Regulatory Guide 1.183>. The specific models, assumptions and the program used for computer evaluation are described in <Section 15.0.3.5>. Specific values of parameters used in the evaluation are presented in <Table 15.6-9>.

15.6.4.5.1.1 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 iodine concentration in the reactor coolant is assumed to be 4.0 μCi per gram dose equivalent I-131:

I-131	9.02 E-1
I-132	9.02 E+0
I-133	6.24 E+0
I-134	1.95 E+1
I-135	9.02 E+0

15.6.4.5.1.2 Fission Product Transport to the Environment

The transport pathway is a direct unfiltered release to the environment. The MSIV detection and closure time of 6.05 seconds results in a discharge of 14,311 pounds of steam and 127,376 pounds of liquid from the break. Assuming all the activity in this discharge becomes airborne, the release of activity to the environment is presented in <Table 15.6-7>.

15.6.4.5.1.3 Results

The calculated exposures for the design basis analysis are presented in <Table 15.6-8> and are a small fraction of the guidelines of <10 CFR 50.67>.

15.6.4.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are described in <Section 15.0.3.5>. Specific values of parameters used in the evaluation are presented in <Table 15.6-9>.

15.6.4.5.2.1 Fission Product Release from Fuel

There is no fuel rod damage as a consequence of this event, therefore, the only activity released to the environment is that associated with the steam and liquid discharged from the break.

15.6.4.5.2.2 Fission Product Transport to the Environment

The activity released from the accident is a function of the coolant activity, valve closure time and mass of coolant released. A portion of the released coolant exists as steam prior to the blowdown, and as such does not contain the same concentration per unit of mass as does the steam generated as a consequence of the blowdown. Therefore, it is necessary to subtract the initial steam mass from the total mass released and assign to it only 2 percent of the iodine activity contained by an equivalent mass of primary coolant.

The following assumptions are used in the calculation of the quantity and types of radioactive material released from the reactor coolant pressure boundary.

a. The amount of coolant discharged is that calculated in the analysis of the nuclear system transient.

b. The concentrations of biologically significant radionuclides contained in the primary coolant is assumed to be 0.2 μCi per gram dose equivalent I-131:

I-131	4.51 E-2
I-132	4.51 E-1
I-133	3.12 E-1
I-134	9.77 E-1
I-135	4.51 E-1

c. The analysis conservatively takes no credit for decay of noble gases to better reflect the instantaneous release from the steam line.

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

Based on the above considerations, the amount of activity which is available for atmospheric dispersion is presented in <Table 15.6-10>.

15.6.4.5.2.3 Results

The calculated exposures for this event are presented in <Table 15.6-11>. As noted, these values are a small fraction 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

This accident was evaluated as part of the analyses supporting PNPP operation in various operating modes and/or with equipment out-of-service results of which are presented in the following Chapter 15 appendices:

- <Appendix 15D> Partial Feedwater Heating Operation Analysis
- <Appendix 15E> Maximum Extended Operating Domain Analysis
- <Appendix 15F> Recirculation System Single-Loop Operation Analysis

This accident was re-analyzed for cycle 8 using the SAFER/GESTR-LOCA methodology. Subsequent fuel cycles will be re-analyzed to ensure the new fuel types remain bounding with the initial (Cycle 8) SAFER/GESTR-LOCA analysis. The results for these cycles with respect to the loss-of-coolant accident are presented in <Appendix 15B> of this chapter.

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 coincident with an SSE earthquake.

The event has been analyzed quantitatively in <Section 3.6>, <Section 6.2>, <Section 6.3>, <Section 7.3>, and <Section 8.3>. Therefore, the following discussion provides only new information not presented in the subject sections. All other information is covered by cross-referencing.

The postulated event represents the envelope evaluation for liquid or steam line failures inside containment.

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 requirements. The subject piping is designed of high quality for severe seismic and environmental conditions to strict codes and standards. However, since such an accident provides an upper limit estimate to the resultant effects for this category of pipe breaks, it is evaluated without the causes being identified.

15.6.5.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.5.2 Sequence of Events and Systems Operation

15.6.5.2.1 Sequence of Events

The sequence of events associated with this accident is shown in <Table 6.3-2> for ECCS performance and <Table 6.2-9> for barrier (containment) performance.

Following the pipe break and scram, the low-low water level (Level 2) or high drywell pressure signal will initiate RCIC and HPCS systems at time 0 plus approximately 30 seconds, and the MSIVs will begin closing on the low-low-low level (Level 1) signal. LPCS and LPCI systems will begin injecting once a pressure permissive is reached.

15.6.5.2.1.1 Identification of Operator Actions

Since automatic actuation and operation of the ECCS is a system design basis, no operator actions are required for the accident. However, the operator should perform the following described actions.

The operator should, after assuring that all rods have been inserted, determine plant condition by observing the annunciators. After observing that the ECCS flows are initiated, the operator should check that the diesel generators have started and are on standby condition. When possible (less than half an hour later), the operator should initiate operation of the RHR system heat exchangers in the suppression pool cooling mode and check that the emergency service water system has been automatically initiated. After the RHR system and other auxiliary systems are in proper operation, the operator should monitor the hydrogen concentration in the drywell for proper activation of the recombiner and mixer, if necessary. The operator should initiate the Feedwater Leakage Control System as described in <Section 6.9.2>.

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 reactor coolant pressure boundary pipe breaks. Possibilities for all pipe breaks sizes and locations are examined in <Section 6.2> and <Section 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 reactor and plant protection systems are discussed in <Section 6.2>, <Section 6.3>, <Section 7.3>, <Section 7.6>, <Section 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 considered in the analysis of the entire spectrum of primary system breaks. The consequences of a LOCA with consideration of single failures 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 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 <Section 6.3>, <Section 7.3>, <Section 7.6>, <Section 8.3>, and <Appendix 15A>.

15.6.5.3.2 Input Parameters and Initial Conditions

Input parameters and initial conditions used for ECCS analysis of this event are given in <Table 6.3-1>.

15.6.5.3.3 Results

Results of this event are given in detail in <Section 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. Postaccident tracking instrumentation and control is assured. Continued long term core cooling is demonstrated. Radiological release is minimized and within limits.

15.6.5.3.4 Consideration of Uncertainties

This event was conservatively analyzed; see <Section 6.3>, <Section 7.3>, <Section 7.6>, <Section 8.3>, and <Appendix 15A> for details.

15.6.5.4 Barrier Performance

The design basis for the containment is to maintain its integrity, as defined by ASME Code Criteria <Section 3.8.2.5>, 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 occurring. Therefore, any postulated loss-of-coolant accident does not result in exceeding the containment design limits. For details and results of the analyses, see <Section 3.8>, <Section 3.9> and <Section 6.2>.

15.6.5.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

a. The "design basis analysis" is based on the alternative accident source term methodology of <Regulatory Guide 1.183>. This conservative analysis is used for the purpose of determining adequacy of the plant design to meet the licensing basis limits for offsite consequences and control room consequences (25 rem TEDE and 5 rem TEDE, respectively). This analysis is referred to as the "design basis analysis."

b. The post-LOCA equipment qualification, vital area access, PASS access, control room dose due to radiation shine, and containment purge isolation analyses are based on the "original licensing basis analysis". This "original licensing basis analysis" is based on the source terms and methodology of <Regulatory Guide 1.3>, Revision 2 and <Regulatory Guide 1.7>, Revision 2 and SRP 15.6.5 (Reference 2).

The exposures were calculated using the computer code RADTRAD (Reference 21).

15.6.5.5.1 Design Basis Analysis

<10 CFR 100> required, in support of the reactor siting, that a fission product release into containment be postulated and that offsite radiological consequences be evaluated against the guideline dose values specified in that regulation. The fission product releases into containment are used for evaluating the acceptability of both the plant

site and the effectiveness of Engineered Safety Feature (ESF) components and systems. As discussed in <Section 15.6.5.1.1>, 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 requirements. In addition, the analysis in <Section 6.3> demonstrates that even in such an unlikely event, the event does not result in failed fuel. 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. The radiological consequence analysis even assumes (without the cause being identified) that ECCS water makeup does not reach the core for two hours postaccident. This produces a source term comparable in quantity to the original licensing basis source term, but different in the timing of the releases and the radionuclide composition (Reference 19). The "original" licensing basis source term (based on Standard Review Plan 15.6.5, <Regulatory Guide 1.3>, and <Regulatory Guide 1.7>) was taken from information published in 1962 by the U.S. Atomic Energy Commission in Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactors" (TID source term).

The alternative Accident Source Term (AST), which is still a very conservative assessment, is based on the advances in the understanding of the timing, magnitude, and chemical forms of fission product releases from severe reactor accidents as published in <Regulatory Guide 1.183>, July 2000 (Reference 19). This Guide reflects the extensive research and experience that culminated in the development of the alternative accident source term. In addition, due to the nature of the analysis that uses a spectrum of postaccident isotopes, the LOCA dose analysis is evaluated using a Total Effective Dose Equivalent (TEDE) methodology. The acceptance criteria for the LOCA is 25 rem TEDE, offsite, and 5 rem TEDE to the control room operators.

The LOCA analysis is based on the following:

- using a reactor accident source term developed from <Regulatory Guide 1.183>,
- credit for decay during the two (2) minute onset of the gap release,
- relying on natural deposition of fission product aerosol in the drywell and unsprayed regions of the containment,
- relying on natural deposition of fission product aerosol in the main steam lines,
- controlling the pH of the water in the containment to prevent iodine re-evolution,
- operating the containment spray system for up to 24 hours <Section 6.5.2.3>,
- not crediting iodine removal by charcoal adsorbers in the Annulus Exhaust Gas Treatment System (AEGTS),
- delaying actuation of the control room emergency recirculation system for up to 30 minutes,
- utilizing elemental and organic iodine removal efficiencies of 80 percent for the control room emergency recirculation system charcoal adsorbers,
- utilizing an engineered safety feature system leakage outside primary containment value of 15 gallons per hour (gph), and
- utilizing a maximum allowable secondary containment bypass leakage of $0.1008L_a$.

The analysis considers the following four potential fission product release pathways following the design basis LOCA:

- main steam isolation valve leakage,
- containment leakage,

- secondary containment bypass leakage, and
- post-LOCA water leakage from engineered safety features systems outside containment.

The fission product transport model used to calculate radiological consequences is shown in <Figure 15.6-2>. The most limiting LOCA with respect to offsite and control room radiological consequences is different than the most limiting LOCA for ECCS analysis. The most limiting event for radiological consequences is a guillotine pipe break in one of the four main steam lines upstream of the inboard MSIV, because this break minimizes the amount of steam line length available for particulate deposition. The results of the radiological consequence analyses are provided in <Table 15.6-15>. <Table 15.6-12a>, <Table 15.6-12b>, and <Table 15.6-12c> summarize the input parameters for the LOCA analysis.

For the LOCA analysis it is assumed that the fraction of core inventory given in Table 1 of (Reference 19) are released from an equilibrium core operating at a power level of 3,833 MWt for 1375 days prior to the accident. This assumed release of the core activity implies substantial fuel damage. Even though this condition is inconsistent with operation of the ECCS systems <Section 6.3>, it is assumed applicable for the evaluation of this accident. Of this release, 100 percent of the noble gases and five percent of the iodine are gaseous. The remainder of the activity released is assumed to be in particulate form as stated in (Reference 19). (Reference 19) also establishes that the spectrum of sources need to be considered as opposed to the limited isotopes of a TID-14844 evaluation.

15.6.5.5.1.1 Main Steam Isolation Valve Leakage Pathway

There are four main steam lines; each line has an inboard MSIV, an outboard MSIV, and a third isolation valve. These valves isolate the reactor coolant system in the event of a break in a steam line outside the primary containment, a design basis LOCA, or other events requiring containment isolation. These MSIVs along with the main steam lines, up to and including the third isolation valve, are designed as Seismic Category I.

The analysis conservatively assumes that the fission product leakage from the main steam lines is released directly into the environment. The leakage past the MSIVs is conservatively assumed to begin immediately after the accident. In actuality, the three intact steam lines would contain trapped steam which would be relatively cooler and more dense as compared to the atmosphere in the reactor vessel upper head during the overheating of the core. This condition would greatly inhibit mixing between the activity released from the core and the steam leaking through the three intact steam lines and the three associated sets of MSIVs. However, for conservatism, all of the lines are assumed to be leaking contaminated drywell atmosphere.

Other significant conservatisms in the analysis of steam line transport include:

(1) No consideration of reduced steam line mass leak rate with decreasing drywell pressure in the region between the inboard and outboard MSIVs (no reduction in flow for this portion of the main steam lines),

(2) No consideration of steam line mass leak rate for two closed MSIVs in series, and

(3) No consideration of particulate removal and even plugging of the extremely small MSIV leak paths due to particulate deposition at the entrance to or within the leak path as the gas flow accelerates to sonic or near-sonic conditions.

Two configurations were analyzed to cover all single-failure possibilities. In the first configuration (Configuration 1), the inboard MSIV on the affected line was assumed to fail open, and this line was assumed to leak at 100 scfh. The three intact lines were then assumed to leak at 100 scfh, 50 scfh, and 0 scfh to maximize flow rates through the lines, which in turn maximizes the activity release. At 20 minutes after the start of release the third safety-related and seismically-qualified isolation valves (just outboard of the outboard

MSIVs) were assumed to be manually closed in all four lines. This configuration was evaluated to be less limiting than a second configuration (Configuration 2) in which all MSIVs successfully closed, but in which the third isolation valves remained open due either to operator error or a failure of the common power supply.

In both cases (i.e., Configuration 1 and Configuration 2) particulate deposition is credited in all volumes of the steam line upstream of closed isolation valves. This is in accordance with Section 5.2.3 of (Reference 18). For the more limiting Configuration 2 this means deposition is considered in the space between the closed MSIVs for the affected line and between the reactor vessel and the closed inboard MSIVs as well as between the closed MSIVs in the three intact lines. Therefore, leakage past the outboard MSIV is assumed to be released directly to the environment. As with Configuration 1, the affected line is assumed to leak at 100 scfh and the three intact lines at 100, 50, and 0 scfh.

For the three intact steam lines, the space between the reactor vessel and the inboard MSIV is assumed to be well-mixed. For all four lines, the space between closed MSIVs is considered to exhibit plug flow as long as unequal cooling of the line does not create the potential for internal circulation (as compared to the magnitude of the plug flow velocity). As the cooling of the line continues and this potential is approached, the effective length of the line between the MSIVs is assumed to be "shortened" so as to ignore the portion where circulation may be occurring and, therefore, to avoid any potential for overestimating the filtration effect of this portion of the steam line. A simple integrable model (similar to the DEPOSITION computer code, (Reference 16)) is used to treat the plug flow. For both models, the input, time-dependent particulate flow rates and the particle size distributions are taken from the results of the "upstream" STARNAUA analyses (Reference 15).

For conservatism, the main steam line aerosol removal efficiency (the ability of the steam lines to retain aerosol fission products) was slightly reduced in the analysis. This aerosol removal efficiency is equivalent to an increase in aerosol penetration of 10 percent. This was done to further increase the dose from the main steam line pathway.

Elemental iodine retention efficiency in the main steam lines is based on a comparison of deposition and resuspension rates from (Reference 17) and was initially set at 50%. Similar to the aerosol removal efficiency reduction, the elemental iodine retention efficiency was also reduced by 10% for conservatism, to 45%.

15.6.5.5.1.2 Fission Product Transport in Drywell

The most limiting DBA, with respect to the offsite and control room radiological consequences, is considered a large-break LOCA as a result of a double guillotine pipe rupture in one of the four main steam lines upstream of the inboard MSIV. It is further conservatively assumed that all fission products are released directly to the drywell and leaked into the primary containment and into the main steam lines, bypassing the suppression pool. The analysis also assumes that at a point two hours after accident initiation (when the ECCS is assumed to be able to reach the core and reflood it) the fission products are homogeneously distributed between the drywell and the primary containment. The objective of this well mixed approach is to achieve an appropriate balance for the design of drywell leakage mitigative devices such as the MSIVs as well as containment leakage mitigative features such as the HEPA filters in the annulus exhaust gas treatment system.

As characterized in (Reference 19), the gap releases and the early in-vessel fission product releases terminate 2 hours after accident initiation. For the fission product releases to terminate, the reactor vessel would need to be reflooded. In lieu of evaluating all of the potential steaming rates due to various reflooding scenarios, the analysis assumes that a substantial amount of fission products will end up in the primary containment as well as in the drywell, and as such, mitigative features such as the HEPA filters in the annulus exhaust gas treatment system are designed to accommodate a significant portion of the source term. The 2-hour assumption for the homogeneous mixture of the source term between the drywell and the containment is used since it provides an appropriate balance, because the "worst 2 hours" are considered for the EAB radiological dose results, as opposed to simply the first 2 hours as was done when the TID source term was used.

The radiological consequences are dependent upon the drywell bypass leakage prior to the termination of fission product release at 2 hours. Because of this sensitivity, the analysis uses a steaming rate of an intact core without relocation to the lower head region, on the order of 3,000 cfm. For the period prior to 2 hours, the analysis conservatively does not credit steaming due to relocation, cooling from alternative water sources, or the release of hydrogen gas, all of which would provide a higher steaming rate and remove more of the fission products from the drywell region.

15.6.5.5.1.3 Elemental Iodine and Aerosol Deposition

Activity released to the drywell as a result of the design basis loss-of-coolant-accident is initially airborne and can be removed from the atmosphere in one of four ways:

(1) Convection from the drywell to the containment

(2) Natural removal within the drywell (e.g., particulate sedimentation)

(3) Leakage into the broken steam line and through the MSIVs

(4) Leakage back into the reactor vessel and through the MSIVs

The depletion due to MSIV leakage is small by design; and therefore, the two principal mechanisms for depletion of activity in the drywell atmosphere (other than by radioactive decay) is convection from the drywell to the containment and natural removal within the drywell.

Following the fuel release phase of the accident, the restoration of ECCS (thus arresting further core damage) would quench the core debris, and results in a rapid sweep-out of the drywell into the containment as discussed in Section 5.2.3 of (Reference 18).

For the design basis analysis, a negotiated licensing basis was established for the transport of activity between the containment and the drywell. The negotiated basis in effect mixes activity between the regions and does not consider a sweep-out of the activity after two hours. The negotiated parameters are in <Table 15.6-12b>.

Elemental iodine removal is credited in the drywell and containment volumes. Airborne elemental iodine is removed by deposition to the walls. This process is driven by the temperature differences between the surfaces and the atmosphere. The calculated removal constants are applied until a decontamination factor (DF) of up to 200 has been obtained. Aerosol removal in the unsprayed regions of containment (including drywell) is modeled using the Power's removal model as given in <NUREG/CR-6189> (Reference 20). The lower bound decontamination coefficient associated with the 10^{th} percentile uncertainty was used for conservatism.

15.6.5.5.1.4   Containment Leakage Pathway

The primary containment consists of a drywell, a wetwell, and supporting systems to limit fission product leakage during and following the postulated LOCA with isolation of the containment boundary penetrations. The design basis leak rate of the primary containment is 0.2 volume percent per day. The analysis reduces the design basis leak rates at 24 hours as permitted by <Regulatory Guide 1.183> for the remaining duration of the accident (30 days) 69 percent of the leak rates assumed during the first 24 hours.

The secondary containment (shield building) which surrounds the primary containment will collect and retain fission product leakage from the primary containment and will release fission products to the environment in a controlled manner through the AEGTS. AEGTS will maintain the secondary containment pressure negative following a DBA by the time the gap release could migrate outside the containment structure. Therefore, if a short period of time exists post-LOCA when the annulus pressure is not negative, the dose calculations would not be affected.

Although the primary containment is enclosed by the secondary containment, there are systems that penetrate both the primary containment and the shield building boundaries that could create potential pathways through which fission products in the primary containment could bypass the leakage collection and filtration systems associated with the shield building. The analysis conservatively assumes 10.08% of the primary containment leakage bypasses the secondary containment.

The analysis assumes 89.92 percent of the primary containment leak rate goes into the secondary containment for its radiological consequence analysis. This leakage is collected in the shield building and processed through the AEGTS HEPA filters before being released into the environment. The remaining 10.08 percent of the primary containment leak rate is assumed to bypass the shield building and to be released directly to the environment for the entire duration of the postulated LOCA.

15.6.5.5.1.5 Annulus Exhaust Gas Treatment System

The AEGTS is an engineered safety features system and is designed to collect, process, and release the fission product leakage from the primary containment into the shield building. The AEGTS is a redundant system consisting of two 100 percent capacity subsystems. Each subsystem has a design capacity of 2000 cfm and consists of, among other things, a HEPA pre-filter, one 4-inch deep charcoal adsorber, and a HEPA post-filter. The system is designed to Seismic Category I standards and is located in a Seismic Category I structure.

The system is operated continuously during normal plant operation, and it maintains a slight negative pressure in the shield building. The analysis assumes a 99 percent removal efficiency for fission products in aerosol form for HEPA filters. The analysis however does not consider any fission product removal by the charcoal adsorbers in the AEGTS. The analysis also conservatively assumes that the entire 2000 cfm flow is discharged directly to the environment with no recirculation (holdup) of iodine in the annulus.

15.6.5.5.1.6 Containment Spray

The containment sprays are an engineered safety feature mode of RHR, designed to provide containment cooling, pressure reduction and fission product removal in the containment following a postulated LOCA. The containment sprays consists of two redundant and independent loops. Each loop has a design spray water flow capacity of 5250 gpm. The system is designed to Seismic Category I standards and is located in a Seismic Category I structure. No chemical additives are used in the containment sprays, other than the pH buffering chemical (boron solution) from the existing Standby Liquid Control System <Section 3.4> following a LOCA.

To support the analysis, the containment sprays will be operated post-LOCA for up to 24 hours based upon plant emergency guidelines <Section 6.5.2.3>. The containment sprays will automatically initiate 10 minutes following a LOCA signal if containment pressure exceeds the high pressure setpoint. Otherwise, the sprays will be manually initiated post-accident within the 30-minute time assumed in the analysis, based on readings from the containment high range radiation monitor or other emergency operating procedure guidance.

See <Section 6.5.2> for additional design basis of the containment sprays.

15.6.5.5.1.7 Post-LOCA Leakage Pathway from Engineered Safety Features Outside Containment

Any leakage of water from ESF components located outside the primary containment releases fission products during the recirculation phase of long-term core cooling following a postulated LOCA. The PNPP administrative controls limit this leakage to less than half of the value used in the radiological dose calculations. The analysis is conservatively based on a leakage rate of 15 gallons per hour of ESF leakage for the entire duration of the accident (30 days). Additionally, leakage from a gross failure of a passive component is assumed to occur at a rate of 50 gpm starting 24 hours into the accident and lasting for 30 minutes. Ten percent of iodine (all forms) contained in the leakage is assumed to be released directly to the environment and the pH of water leakage is assumed to be above 7.

15.6.5.5.1.8 Postaccident Containment Water Chemistry Management

(Reference 19) concludes that iodine entering the containment from the reactor coolant system during an accident would be composed of at least 95 percent cesium iodide (CsI), with no more than 5 percent of iodine (I) and hydrogen iodide (HI). Once in the containment, highly soluble cesium iodide will readily dissolve in water pools forming iodide (I^-) in solution. (Reference 19) also considers the radiation-induced conversion of iodide in water into elemental iodine (I_2,) to be strongly dependent on the pH, and without pH control, a large fraction of iodide dissolved in water pools in ionic form will be converted to elemental iodine and will be released into the containment atmosphere if the pH is less than 7. On the other hand, if the pH is maintained above 7, very little (less than 1 percent) of the dissolved iodides will be converted to elemental iodine.

The Standby Liquid Control System (SLCS) is used for controlling and maintaining long-term suppression pool water pH levels to 7 or above following the postulated DBA. The SLCS is a safety-related system and designed as a Seismic Category I system. Its primary function is as a reactivity control system to provide backup capability to be able to shut down the reactor if the normal control rods become inoperable. The system is manually initiated from the main control room to pump a boron neutron absorber solution into the reactor.

The SLCS contains a borax-boric acid solution. Such boron solutions act as pH buffers. Buffering will cause only a small decrease in pH with addition of an acid so long as the buffer capacity is not exceeded. The analysis used a containment water pool volume (which includes the suppression pool and reactor coolant inventory) of 8.357E+5 gallons and assumed all cesium iodide released into the drywell is directly deposited in the containment water pool. The analysis of pH levels in the containment water pool considered the following factors:

(1) cesium hydroxide formed from the fission products released from the core (basic-raises pH)

(2) the addition of the boron solution from SLCS (buffer)

(3) nitric acid produced by irradiation of water and air in the containment (acid-lowers pH)

(4) hydrochloric acid generated from electrical cable degradation (acid-lowers pH)

The analyses demonstrate that with the amount of the boron solution provided in the containment, the pH of the postaccident water in the containment will remain above 7 for the duration of the postulated LOCA.

15.6.5.5.1.9 Control Room Habitability

Upon receipt of an ESF actuation system signal or high radiation, the control room Heating, Ventilation, and Air Conditioning (HVAC) system is designed to automatically switch to the emergency recirculation mode of operation (CRERS). The analysis conservatively assumes a 30-minute delay in actuation of the CRERS.

The CRERS is a redundant system and each subsystem has a design flow capacity of 30,000 cfm. The analysis uses a conservative recirculation flow rate of 27,000 cfm. Each subsystem consists of, among other things, a High-Efficiency Particulate Air (HEPA) filter, charcoal adsorbers, and a HEPA post-filter. The analysis also uses a conservative HEPA filter efficiency of 99 percent for aerosol particulate and an 80 percent charcoal filter removal efficiency for iodine in elemental and organic forms.

During normal operation, the HVAC system is designed to pressurize the control room envelope with 45,000 cfm recirculation airflow and with 6,000 cfm outside makeup air. During an emergency, when the system operates in the emergency recirculation mode, the outside makeup air is isolated and the control room envelope is not pressurized relative to adjacent areas. To be conservative, the analysis uses 6,600 cfm unfiltered inleakage to the control room during the first 30 minutes, followed by 1,375 cfm unfiltered inleakage in the emergency recirculation mode after 30 minutes. The major parameters and assumptions used in the analysis are listed in <Table 15.6-14>.

15.6.5.5.1.10 Atmospheric Relative Concentrations at Control Room, Exclusion Area Boundary and Low Population Zone

The atmospheric dispersion factors used in the Control Room Habitability analysis were determined based on several analyses including NRC ARCON96 calculations in conjunction with the NUS Tracer Gas Study (Reference 8). The NUS Tracer Gas Study was performed to characterize the atmospheric dispersion within the building complex at PNPP. Prior estimates of atmospheric relative dispersion (X/Q) values had been made for postulated releases to the control room using the Murphy-Campe methodology referenced in Standard Review Plan 6.4. The objective of conducting the tracer gas tests was to demonstrate more site specific/realistic control room air intake X/Q values.

The NRC reviewed and compared the results of the tracer gas study with calculations made using the ARCON96 methodology described in <NUREG/CR-6631>, Revision 1, "Atmospheric Relative Concentrations in Building Wakes" (Reference 13).

For the postulated release point resulting in the largest X/Q values, the calculated X/Q values from the tracer gas study were as much as 50 times lower than the original X/Q values calculated using the Murphy-Campe methodology. For the same postulated release point, X/Q values using the ARCON96 methodology were approximately two times lower than the Murphy-Campe values.

The ARCON96 methodology assumes that the effluent travels the shortest distance possible between the postulated release point and the control room air intake. While the model calculates dispersion within building complexes, it is not intended to provide an exact model of postulated scenarios for complex site-specific flow paths around obstructions. Meander and building effects are implicitly factored in, based on the field test studies used in the development of ARCON96.

At PNPP, the effluent from a release postulated from the plant vent or containment building would need to disperse over or around an obstruction, down the side of a building and around a missile shield to be drawn into the control room air intake. For the limiting case, the X/Q values calculated from field tests performed by the licensee are about a factor of two or three lower for the control room air intake than for measurements made at the top of the building on which the intake is located. Thus, it was determined that results using ARCON96 would overestimate X/Q values for this scenario at PNPP.

The tracer gas field tests were conducted over a period of approximately one week in September 1985. While care was taken to assure that the tests were made under adequately limiting meteorological conditions, there is some likelihood that testing may not have captured the full range of poor dispersion conditions. Also, the field measurements may include some off centerline conditions, and due to solar heating in the building complex, better dispersion may have occurred during the tests than might occur at some other times of the year.

After discussing the tracer gas test limitations with the NRC, the X/Q values in <Table 15.6-13> were accepted as the PNPP design basis.

The AST analysis also reviewed the X/Q's for the EAB and LPZ. The analyses are based upon the <Regulatory Guide 1.145>. The dispersion factors used in the analyses for the offsite dose analysis are in <Table 15.6-13>.

15.6.5.5.1.11 Results

The calculated exposures for the design basis analysis are presented in <Table 15.6-15> and are within the licensing basis limits of 25 rem TEDE (offsite) and 5 rem TEDE (control room).

15.6.6 FEEDWATER LINE BREAK - OUTSIDE CONTAINMENT

This accident was not reanalyzed for the current reload since the original analysis is still applicable.

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 to this postulated event. The postulated break of the feedwater line, representing the largest liquid line outside containment, provides the envelope evaluation relative to this type of occurrence. The break is assumed to be instantaneous, circumferential and upstream of the outermost isolation valve.

A more limiting event from a core performance evaluation standpoint (feedwater line break inside containment) has been quantitatively analyzed in <Section 6.3>, "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 topics in <Section 6.3>.

15.6.6.1 Identification of Causes and Frequency Classification

15.6.6.1.1 Identification of Causes

A feedwater line break is assumed without the cause being identified. The subject piping is designed to high quality, to strict engineering codes and standards and to severe seismic environmental requirements.

15.6.6.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.6.2 Sequence of Events and Systems Operation

15.6.6.2.1 Sequence of Events

The sequence of events is shown in <Table 15.6-19>.

15.6.6.2.1.1 Identification of Operator Actions

Since automatic actuation and operation of the ECCS is a system design basis, no operator actions are required for this accident. However, the operator should perform the following actions:

a. Determine that a line break has occurred and evacuate the area of the turbine building.

b. The operator is not required to take any action to prevent reactor coolant loss, but should ensure that the reactor is shutdown and that RCIC and/or HPCS are operating normally.

c. Implement appropriate emergency procedures.

d. If possible, shut down the feedwater system and de-energize any electrical equipment which may be damaged by water from the feedwater system in the turbine building.

e. Continue to monitor reactor water level and performance of the ECCS systems while emergency procedures are being implemented, and begin normal reactor cooldown.

f. When the reactor pressure has decreased below 150 psia, initiate RHR in the shutdown cooling mode to continue cooling the reactor.

The above operations occur over an elapsed time of 3-4 hours.

15.6.6.2.2 Systems Operations

It is assumed that the normally operating plant instruments and controls, reactor protection, ADS, containment and reactor vessel isolation, ECCS and RHR systems function properly to assure a safe shutdown as described in <Appendix 15A>.

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 <Section 6.3.3.3>. For the feedwater line break outside containment, since the break is isolable, 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 analysis.

15.6.6.3 Core and System Performance

15.6.6.3.1 Qualitative Summary

The accident evaluation qualitatively considered in this section is considered to be a conservative, envelope assessment of the consequences of the postulated failure (severance) of one of the feedwater lines external to the containment. The accident is postulated to occur at the input parameters and initial conditions in <Table 6.3-1>.

15.6.6.3.2 Qualitative Results

The feedwater line break outside the containment is less limiting than either of the steam line breaks outside the containment (analysis presented in <Section 6.3> and <Section 15.6.4> or the feedwater line break inside the containment (analysis presented in <Section 6.3.3>. It is far less limiting than the design basis accident (the recirculation line break analysis presented in <Section 6.3.3> and <Section 15.6.5>.

The reactor vessel is isolated on low-low-low water level (L1), and HPCS restores 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 <Section 6.3>.

15.6.6.4 Barrier Performance

Accidents that result in the release of radioactive materials outside the containment are the results of postulated breaches in the reactor coolant pressure boundary or the steam power conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the main steam lines as described in <Section 15.6.4>. The feedwater system pipe break is less severe than the main steam line break.

15.6.6.5 Radiological Consequences

15.6.6.5.1 Design Basis Analysis

The NRC provides no specific regulatory guidelines for the evaluation of this accident, therefore, no design basis analysis will be presented.

15.6.6.5.2 Realistic Analysis

The realistic analysis is based on a realistic, but still conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are described in (Reference 6). Parameters used in the evaluation are presented in <Table 15.6-20>. A schematic diagram of the leakage path for this accident is shown in <Figure 15.6-3>.

15.6.6.5.2.1 Fission Product Release

There is no fuel damage as a consequence of this accident. In addition, an insignificant quantity of activity (compared to that existing in the main condenser hotwell prior to occurrence of the break) is released from the contained piping system prior to isolation closure.

The iodine concentration in the main condenser hotwell is consistent with an offgas release rate of 100,000 μCi/sec at 30 minutes delay, and is 0.02 (2 percent carryover) times the concentration in the reactor coolant. Noble gas activity in the condensate is negligible since the air ejectors remove practically all noble gas from the condenser.

15.6.6.5.2.2 Fission Product Transport to the Environment

The transport pathway consists of liquid release from the break, carryover to the turbine building atmosphere due to flashing and unfiltered release to the environment through the turbine building ventilation system.

The total integrated mass of coolant leaving the break is 1.454 E6 lbs of condensate. For the purposes of this evaluation, the conservative assumption is made that the activity of iodine per pound of steam is equal to 2 percent of the activity per pound of water.

Taking no credit for holdup, decay or plateout during transport through the turbine building, the release of activity to the environment is presented in <Table 15.6-21>. The total release is assumed to take place within 2 hours of the occurrence of the break.

15.6.6.5.2.3 Results

The calculated exposures for the realistic analysis are presented in <Table 15.6-22> and are a small fraction of <10 CFR 100> guidelines (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <10 CFR 50.67>.

15.6.6.5.2.4 Sensitivity Analysis

As described in <Section 6.2.4.2.2.1.a.1>, should a break occur in a feedwater line, the control closure check valves prevent significant loss of reactor coolant inventory and provide immediate isolation. A sensitivity analysis was performed to estimate the amount of leakage that would have to occur through the control closure check valves in order for the consequences of a feedwater line break outside containment event to exceed the consequences of the main steam line break outside containment. The results of the sensitivity analysis are that the leakage through the control closure check valves would have to exceed 200 gallons per minute for each feedwater line (400 gallons per minute total) for 2 hours in order for the consequences of the feedwater line break outside containment to exceed the consequences of the main steam line break outside containment <Table 15.6-8> and <Table 15.6-11>. The alternative non-Type C testing performed on these check valves per the Inservice Testing Program will verify proper closure of these valves to prevent significant leakage of this order of magnitude. The “exercise closed” (EC) testing will include a water leak rate test with an acceptance criterion of ≤200 gallons per minute per Feedwater penetration, when tested at ≥1.1 P_a), with no significant valve seat orifice defects [those large enough to result in leakage greater than the 200 gpm limit (400 gpm total) during a high pressure transient].

15.6.7 REFERENCES FOR SECTION 15.6

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20. D. A. Powers, K. E. Washington, S. B. Burson, and J. L. Sprung, “A Simplified Model of Aerosol Removal by Natural Process in Reactor Containments,” NUREG/CR-6189, SAND 94-0407, Sandia National Laboratories, Albuquerque, NM, 1995.

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TABLE 15.6-1

SEQUENCE OF EVENTS FOR INSTRUMENT LINE BREAK

Time	Event
0	Instrument line fails.
0-10 minutes	Identification of break.
10 minutes	Activate RHR and initiate orderly shutdown.
5 hours	Reactor vessel depressurized and break flow terminated.

TABLE 15.6-2

INSTRUMENT LINE BREAK ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions[1]	Realistic Basis Assumptions[1]
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	None	N/A
B. Burn-up	None	N/A
C. Fuel damaged	None	None
D. Release of activity by nuclide	None	<Table 15.6-3>
E. Iodine fractions, %		
(1) Organic	None	0
(2) Elemental	None	100
(3) Particulate	None	0
F. Reactor coolant activity before the accident	None	<Section 15.6.2.5.2.1>
II. Data and assumptions used to estimate activity released		
A. Primary containment leak rate (%/day)	None	N/A
B. Secondary containment leak rate (%/day)	None	N/A
C. Valve movement times	None	N/A
D. Adsorption and filtration efficiencies		
(1) Organic iodine	None	90
(2) Elemental iodine	None	90
(3) Particulate iodine	None	90
(4) Particulate fission products	None	N/A
E. Recirculation system parameters		
(1) Flow rate	None	N/A
(2) Mixing efficiency	None	N/A
(3) Filter efficiency	None	N/A
F. Containment spray parameters (flow rate, drop size, etc.)	None	N/A
G. Containment volumes	None	N/A
H. All other pertinent data and assumptions	None	None

TABLE 15.6-2 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
III. Dispersion Data		
A. Boundary and LPZ distance (m)	None	863/4002
B. X/Q's (sec/m^3) for time intervals of		
(1) 0-2 hr - SB/LPZ	None	6.7E-4/8.2E-5
(2) 2-8 hr - LPZ	None	8.2E-5
(3) 8-24 hr - LPZ	None	5.2E-5
(4) 1-4 days - LPZ	None	1.9E-5
(5) 4-30 days - LPZ	None	4.7E-6
IV. Dose Data		
A. Method of dose calculation	N/A	<Section 15.0.3.5>
B. Dose conversion assumptions	N/A	<Section 15.0.3.5>
C. Peak activity concentrations in containment	N/A	<Section 15.6.2.3>
D. Doses	N/A	<Section 15.6.2.5>

NOTE:

[1] As detailed in this table, no Design Basis radiological consequence analysis was performed for initial plant licensing. The Realistic Basis Assumptions presented herein are retained for historical information (not updated).

TABLE 15.6-3

INSTRUMENT LINE FAILURE (REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN INSTRUMENT LINE BREAK STRUCTURE (CURIES)
(Historical)

Isotope	Activity
I-131	7.46E + 1
I-132	1.15E + 2
I-133	1.79E + 2
I-134	1.97E + 2
I-135	1.70E + 2

<TABLE 15.6-4>

<TABLE 15.6-5>

DELETED

TABLE 15.6-6

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 isolation valves.
<1.0	Reactor begins scram.
≤5.5	Main steam isolation valves fully closed.
18	Safety/relief valves open on high vessel pressure. The valves open and close to maintain vessel pressure at approximately 1,000 psi.
43	RCIC and HPCS initiate on low-low water level (RCIC considered unavailable, HPCS assumed single failure and therefore, not available).
340	ADS signal to initiate on low-low-low water level.
379	Reactor water level begins to drop slowly due to loss of steam through the safety/relief valves. Reactor pressure still at approximately 1,000 psi.
460	ADS initiated, vessel depressurizes rapidly.
611 <Section 6.3.3>	Low pressure ECCS systems initiated. Reactor fuel uncovered partially.
<Section 6.3.3>	Core effectively reflooded and clad temperature heatup terminated. No fuel rod failure.

TABLE 15.6-7

STEAM LINE BREAK ACCIDENT
(DESIGN BASIS ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT (CURIES)

Isotope	Activity
I-131	5.259E +1
I-132	5.277E +2
I-133	3.637E +2
I-134	1.148E +3
I-135	5.263E +2
Kr-83m	1.028E -1
Kr-85m	2.261E -1
Kr-85	8.447E -4
Kr-87	6.740E -1
Kr-88	6.697E -1
Kr-89[1]	4.288E +0
Xe-131m	8.582E -4
Xe-133m	9.007E -3
Xe-133	2.581E -1
Xe-135m	9.974E -1
Xe-135	7.042E -1
Xe-137[1]	4.873E +0
Xe-138	2.840E +0
Br-83	6.151E +1
Br-84	1.337E +2
Br-85[1]	9.826E +1

NOTE:

[1] Not included in dose calculation since no dose conversion factors for these isotopes are listed in FGR 11 or FGR 12.

TABLE 15.6-8

STEAM LINE BREAK ACCIDENT
(Iodine Concentration in Coolant = 4.0 μCi/gm dose - equivalent I-131)
RADIOLOGICAL EFFECTS

	TEDE Dose (rem)	Licensing Basis Limit (TEDE, rem)
Exclusion area (863 Meters)	1.17E+0	25
Low population zone (4,002 Meters)	1.31E-1	25
Control Room[(1)]	6.22E-1	5

NOTE:

(1) Normal HVAC flow w/no control room isolation.

TABLE 15.6-9

STEAM LINE BREAK ACCIDENT - PARAMETERS
TABULATED FOR POSTULATED ACCIDENT ANALYSES

		Design Basis Assumptions	Realistic Basis Assumptions
I.	Data and assumptions used to estimate radioactive source from postulated accidents		
	A. Power level	N/A	N/A
	B. Burn-up	N/A	N/A
	C. Fuel damaged	None	None
	D. Release of activity by nuclide	<Table 15.6-7>	<Table 15.6-10>
	E. Iodine fractions (%)		
	(1) Organic	0.15	0.15
	(2) Elemental	4.85	4.85
	(3) Particulate	95	95
	F. Reactor coolant activity before the accident	<Section 15.6.4.5.1.1>	<Section 15.6.4.5.2.1>
II.	Data and assumptions used to estimate activity released		
	A. Primary containment leak rate (%/day)	N/A	N/A
	B. Secondary containment leak rate (%/day)	N/A	N/A
	C. Isolation valve closure time (sec)	6.05	6.05
	D. Adsorption and filtration efficiencies		
	(1) Organic iodine	N/A	N/A
	(2) Elemental iodine	N/A	N/A
	(3) Particulate iodine	N/A	N/A
	(4) Particulate fission products	N/A	N/A
	E. Recirculation system parameters		
	(1) Flow rate	N/A	N/A
	(2) Mixing efficiency	N/A	N/A
	(3) Filter efficiency	N/A	N/A
	F. Containment spray parameters (flow rate, drop size, etc.)	N/A	N/A

TABLE 15.6-9 (Continued)

		Design Basis Assumptions	Realistic Basis Assumptions	
G.	Containment volumes	N/A	N/A	
H.	All other pertinent data and assumptions	None	None	

III. Dispersion Data

A.	Boundary and LPZ distance (m)	863/4002	863/4002	
B.	X/Q's for total (sec/m^3)			
	For time intervals of:	EAB	LPZ	CR
	(1) 0-8 hrs	4.3E-4	4.8E-5	3.5E-4
	(2) 8-24 hrs	4.3E-4	3.3E-5	2.1E-4
	(3) 24-96 hrs	4.3E-4	1.4E-5	1.1E-4
	(4) 96-720 hrs	4.3E-4	4.1E-6	5.75E-5

IV. Data and assumptions used to estimate dose to the Control Room

Volume (ft^3)	4.205E+5	4.205E+5
Flow Rate – unfiltered inleakage (cfm)	6600	6600
Recirculation flow (cfm)	0	0
Recirculation filter efficiencies (%)		
Particulate	0	0
Elemental and organic Iodine	0	0

V. Dose Data

A.	Method of dose calculation	<Section 15.0.3.5>	<Section 15.0.3.5>
B.	Dose conversion assumptions	<Section 15.0.3.5>	<Section 15.0.3.5>
C.	Peak activity concentrations in containment	N/A	N/A
D.	Doses	<Table 15.6-8>	<Table 15.6-11>

TABLE 15.6-10

STEAM LINE BREAK ACCIDENT
(REALISTIC ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT (CURIES)

Isotope	Activity
I-131	2.630E +0
I-132	2.639E +1
I-133	1.819E +1
I-134	5.739E +1
I-135	2.632E +1
Kr-83m	1.028E -1
Kr-85m	2.261E -1
Kr-85	8.447E -4
Kr-87	6.740E -1
Kr-88	6.697E -1
Kr-89[1]	4.288E +0
Xe-131m	8.582E -4
Xe-133m	9.007E -3
Xe-133	2.581E -1
Xe-135m	9.974E -1
Xe-135	7.042E -1
Xe-137[1]	4.873E +0
Xe-138	2.840E +0
Br-83	3.075E +0
Br-84	6.687E +0
Br-85[1]	4.913E +0

NOTE:

[1] Not included in dose calculation since no dose conversion factors for these isotopes are listed in FGR 11 or FGR 12.

TABLE 15.6-11

STEAM LINE BREAK ACCIDENT
(Iodine Concentration in Coolant = 0.2 μCi/gm dose – equivalent I-131)
RADIOLOGICAL EFFECTS

	TEDE Dose (rem)	Licensing Basis Limit (TEDE, rem)
Exclusion area (863 Meters)	5.90E-2	2.5
Low population zone (4,002 Meters)	6.59E-3	2.5
Control Room[(1)]	3.11E-2	5

NOTE:

(1) Normal HVAC flow w/no control room isolation.

TABLE 15.6-12a

LOSS-OF-COOLANT ACCIDENT PARAMETERS AND ASSUMPTIONS
USED IN RADIOLOGICAL CONSEQUENCE CALCULATIONS
MAIN STEAM ISOLATION VALVE LEAKAGE PATHWAY

Parameter	Value
Reactor power (3758 MWt x 1.02)	3833 MWt
Drywell volume	2.765×10^5 ft^3
Containment (excluding drywell) volume	1.165×10^8 ft^3
Volume of one main steam line between MSIV's	146 ft^3
Volumetric flow rate, drywell to broken steam line	
0 to 7484 seconds	1.987 ft^3/min
7484 seconds to 24 hours	1.647 ft^3/min
24 hours to 30 days	1.371 ft^3/min
Volumetric flow rate, drywell to intact steam lines	
0 to 7484 seconds	2.98 ft^3/min
7484 seconds to 24 hours	2.47 ft^3/min
24 hours to 30 days	2.056 ft^3/min
Volumetric flow rate (maximum) in one main steam line, between the MSIV's, then to environment, t = 0 to 30 days	3.183 ft^3/min
Volumetric flow rate in intact main steam lines, between the MSIVs, then to environment, t = 0 to 30 days	4.775 ft^3/min

TABLE 15.6-12b

LOSS-OF-COOLANT ACCIDENT PARAMETERS AND ASSUMPTIONS USED IN RADIOLOGICAL CONSEQUENCE CALCULATIONS CONTAINMENT LEAKAGE PATHWAY

Parameter	Value
Reactor power (3758 x 1.02)	3833 MWt
Volume of sprayed region	4.812×10^5 ft^3
Volume of containment unsprayed region	6.842×10^5 ft^3
Volume of total unsprayed region (including drywell)	9.607×10^5 ft^3
Flow rate from drywell to containment unsprayed region	
0 – 0.5 hrs	0 ft^3/min
0.5 – 2 hours	3000 ft^3/min
2 hours – 30 days	2.77×10^5 ft^3/min
Flow rate from containment unsprayed region to drywell	
0 – 2 hours	0 ft^3/min
2 hours – 30 days	2.77×10^5 ft^3/min
Flow rate between drywell and sprayed region	0 ft^3/min
Flow rate from sprayed region to containment unsprayed region	71,400 ft^3/min
Flow rate from containment unsprayed region to sprayed region	71,400 ft^3/min
Containment leak rate to environment from sprayed region	
40 seconds – 24 hours	0.067 ft^3/min
24 hours – 30 days	0.0462 ft^3/min
Spray removal rate for particulate	10 percent uncertainty distribution
Spray fall height	54.05 ft
Spray removal rate for elemental iodine (sprayed region only)	<Table 6.5-11>
Containment leak rate to environment from total unsprayed region	
40 seconds – 24 hours	0.135 ft^3/min
24 hours – 30 days	0.0932 ft^3/min
Containment leak rate to annulus from sprayed region	
40 seconds – 24 hours	0.601 ft^3/min
24 hours – 30 days	0.415 ft^3/min
Containment leak rate to annulus from total unsprayed region	
40 seconds – 24 hours	1.20 ft^3/min
24 hours – 30 days	0.828 ft^3/min
Annulus volume	1.96×10^5 ft^3
Flow rate from annulus to environment	2000 ft^3/min

TABLE 15.6-12b (Continued)

LOSS-OF-COOLANT ACCIDENT PARAMETERS AND ASSUMPTIONS
USED IN RADIOLOGICAL CONSEQUENCE CALCULATIONS
CONTAINMENT LEAKAGE PATHWAY

Parameter	Value
Annulus exhaust gas treatment system filter efficiency	
particulate	99 percent
elemental and organic iodine	0

TABLE 15.6-12c

LOSS-OF-COOLANT ACCIDENT PARAMETERS AND ASSUMPTIONS
USED IN RADIOLOGICAL CONSEQUENCE CALCULATIONS
ENGINEERED SAFETY FEATURE (ESF) LEAKAGE PATHWAY

ECCS Leakage Model

Parameter	Value
Plant power (3758 MWt x 1.02)	3833 MWt
Release fractions and timing	As specified for BWR in <Regulatory Guide 1.183> (gap and early in-vessel iodine releases only)
Release location	Directly from core to suppression pool
Suppression pool water volume	114,379 ft^3
ECCS leak rate	
0 – 24 hours	15 gph
24 – 24.5 hours	15 gph and 50 gpm for 30 minutes
24.5 hours – 30 days	15 gph
Partition factor	10

<TABLE 15.6-12d>

DELETED

<TABLE 15.6-12e>

DELETED

TABLE 15.6-13

METEOROLOGICAL DATA

Exclusion Area Boundary

Time (hr)	X/Q (sec/m^3)
0-720	$4.3x10^{-4}$

Low Population Zone Distance

Time (hr)	X/Q (sec/m^3)
0-8	$4.8x10^{-5}$
8-24	$3.3x10^{-5}$
24-96	$1.4x10^{-5}$
96-720	$4.1x10^{-6}$

Control Room

Time (hr)	X/Q (sec/m^3)
0-8	$3.5x10^{-4}$
8-24	$2.1x10^{-4}$
24-96	$1.1x10^{-4}$
96-720	$5.75x10^{-5}$

TABLE 15.6-14

CONTROL ROOM MODEL

Parameter	Value
Volume	3.90×10^5 ft^3
Flow rate - unfiltered inleakage	
0 - 0.5 hour	6600 ft^3/min
0.5 hour - 30 days	1375 ft^3/min
Flow rate - exhaust	
0 - 0.5 hour	4800 ft^3/min
0.5 hour - 30 days	1375 ft^3/min
Recirculation flow rate	
0 - 0.5 hour	0
0.5 hour - 30 days	2.7×10^4 ft^3/min
Recirculation filter efficiencies	
particulate	99%
elemental and organic iodine	80%
Occupancy Factors	
0 - 24 hours	1.0
24 - 96 hours	0.6
96 - 720 hours	0.4

TABLE 15.6-15

LOSS-OF-COOLANT ACCIDENT
(DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS[1]

		Dose (Expressed as TEDE, Rem)	Licensing Basis Limit (TEDE, Rem)
1.	Offsite Doses		
	Exclusion area (863 Meters)	21.32	25
	Low population zone (4,002 Meters)	6.9	25
2.	Control Room Doses (0-30 days)	3.01	5

NOTE:

[1] GNF2 Fuel, 1375 Effective Full Power Days (EFPD).

<TABLE 15.6-16>

<TABLE 15.6-17>

<TABLE 15.6-18>

DELETED

TABLE 15.6-19

SEQUENCE OF EVENTS FOR FEEDWATER LINE BREAK
OUTSIDE CONTAINMENT

Time	Event
0 seconds	One feedwater line breaks.
0+ seconds	Feedwater line check valves isolate the reactor from the break.
<30 seconds	At low water level, the reactor would scram. At low-low water level, RCIC would initiate, HPCS would initiate and recirculation pumps would trip. MSIVs will close at water Level 1.
~2 minutes	The safety/relief valves would cycle open and close to maintain reactor vessel pressure at approximately 1,100 psig.
1 to 2 hours	Normal reactor cooldown procedure established.

TABLE 15.6-20

FEEDWATER LINE BREAK ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	N/A	N/A
B. Burn-up	N/A	N/A
C. Fuel damaged	N/A	None
D. Release of activity by nuclide	N/A	<Table 15.6-21>
E. Iodine fractions		
(1) Organic	N/A	0
(2) Elemental	N/A	1
(3) Particulate	N/A	0
F. Reactor coolant activity before the accident	N/A	<Section 15.6.6.5.2.1>
II. Data and assumptions used to estimate activity released		
A. Primary containment leak rate (%/day)	N/A	N/A
B. Secondary containment leak rate (%/day)	N/A	N/A
C. Isolation valve closure time (sec)	N/A	N/A
D. Adsorption and filtration efficiencies		
(1) Organic iodine	N/A	N/A
(2) Elemental iodine	N/A	N/A
(3) Particulate iodine	N/A	N/A
(4) Particulate fission products	N/A	N/A
E. Recirculation system parameters		
(1) Flow rate	N/A	N/A
(2) Mixing efficiency	N/A	N/A
(3) Filter efficiency	N/A	N/A
F. Containment spray parameters (flow rate, drop size, etc.)	N/A	N/A
G. Containment volumes	N/A	N/A

TABLE 15.6-20 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
II. Data and assumptions used to estimate activity released (Continued)		
H. All other pertinent data and assumptions	N/A	None
III. Dispersion Data		
A. Boundary and LPZ distance (m)	863/4002	863/4002
B. X/Q's for total dose - SB/LPZ (sec/m^3)	6.7E-4/8.2E-5	6.7E-4/8.2E-5
IV. Dose Data		
A. Method of dose calculation	N/A	<Section 15.0.3.5>
B. Dose conversion assumptions	N/A	<Section 15.0.3.5>
C. Peak activity concentrations in containment	N/A	N/A
D. Doses	N/A	<Table 15.6-22>

TABLE 15.6-21

FEEDWATER LINE BREAK
(REALISTIC ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT (CURIES)

Isotope	Activity
I-131	2.64E - 1
I-132	3.42
I-133	1.98
I-134	6.33
I-135	3.17

TABLE 15.6-22

FEEDWATER LINE BREAK
RADIOLOGICAL EFFECTS

	Inhalation Dose (rem)
EXCLUSION AREA (863 Meters)	9.37E - 1
Low population zone (4,002 Meters)	1.147E - 1

NOTE:

These results do not account for feedwater check valve leakage. Refer to <Section 15.6.6.5.2.4> for sensitivity analysis.

15.7 RADIOACTIVE RELEASE FROM SUBSYSTEMS AND COMPONENTS

15.7.1 RADIOACTIVE GAS WASTE SYSTEM LEAK OR FAILURE

This accident is not affected by the reload analysis.

The following radioactive gas waste system components are examined under several failure mode conditions:

a. Main condenser gas treatment system failure.

b. Malfunction of main turbine gland sealing system.

c. Failure of air ejector lines.

15.7.1.1 Main Condenser Offgas Treatment System Failure

15.7.1.1.1 Identification of Causes and Frequency Classification

Those events which could cause a gross failure in the offgas treatment system are:

a. A seismic occurrence exceeding the seismic capabilities of the equipment.

b. A hydrogen detonation which ruptures the system pressure boundary.

c. A fire in the filter assemblies.

d. Failure of adjacent equipment which could subsequently compromise offgas equipment.

The seismic event is considered to be the most probable and is the only conceivable 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 decay heat on the filters is insignificant and cannot serve as an ignition source for the filters.

The system is 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>.

This seismic event, more severe than the design requirements, is categorized as a limiting fault.

15.7.1.1.2 Sequence of Events and Systems Operation

a. Sequence of Events

   The probable sequence of events following this failure is shown in <Table 15.7-1>.

   1. 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 should monitor the turbine generator auxiliaries and break vacuum as soon as possible. The operator should 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.

b. 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:

1. Capability to detect the failure itself as indicated by an alarmed increase in radioactivity levels seen by the area radiation monitoring system, an alarmed loss of flow in the offgas system, and an alarmed increase in activity at the vent release.

2. Capability to isolate the system and shutdown the reactor.

3. Operational indicators and annunciators in the main control room.

c. The Effect of Single Failure and Operator Errors

The seismic event which is assumed to occur beyond the present plant design basis for nonsafety equipment will cause the tripping of the 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. However, for conservatism, the SJAE will be assumed to continue pumping process gas for 30 minutes.

15.7.1.1.3 Core and System Performance

The postulated failure results in a system isolation necessitating reactor shutdown because of loss of vacuum in the main condenser. This transient has been analyzed in <Section 15.2.5>.

15.7.1.1.4 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 the next section.

15.7.1.1.5 Radiological Consequences

Two separate radiological analyses are provided for the seismic accident:

a. The first analysis is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet <10 CFR 100> guidelines (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <10 CFR 50.67>. This analysis is referred to as the "design basis analysis."

b. The second is based on assumptions considered to provide a realistic conservative estimate of radiological consequences. This analysis is referred to as the "realistic analysis."

Both are based on the following equipment characteristics with respect to retention of radioactive solid daughter products during normal operation of the offgas system.

a. Offgas condenser - 100 percent retained and continuously washed out with condensate.

b. Water separator - (included with offgas condenser).

c. Holdup pipe - 60 percent retained and continuously washed out with condensate.

d. Cooler condenser - 100 percent retained and continuously washed out with condensate.

e. Moisture separator - (included with cooler condenser).

f. Prefilter - 100 percent retained, element changed annually.

g. Desiccant dryer - 100 percent retained, desiccant replaced approximately once every five years.

h. Charcoal adsorbers - 100 percent retained.

i. After-filter - 100 percent retained, element changed annually.

Components not listed are assumed to have zero retention of solid daughter products.

Both analyses assume that the SJAE continues to pump the process gas out of a break near the failed component for 30 minutes after the accident. The release rates for breaks at the holdup pipe exit are given in <Table 15.7-3>.

a. Design Basis Analysis

1. Fission Product Release Assumptions

The activity in the offgas system is based on 2 scfm air inleakage and 100,000 μCi/sec noble gas after 30 minutes delay for a period of 11 months, followed by 1 month of 350,000 μCi/sec at 30 minutes.

Depending upon the assumptions as to radionuclide release fractions for each equipment piece, the assumed single failure of any one of several components could be controlling with respect to dose consequences. The assumed release fractions for the design basis analysis are found in <Table 15.7-4>. The bases for the failure assumptions of that equipment expected to have the worst dose consequences follow.

(a) Charcoal Adsorbers and Desiccant Vessel

These vessels are designed with thick walls for detonation resistance. The only credible failure that could result in loss of carbon is a vessel nozzle failure due to excessive nozzle loads during the seismic event. Assuming the vessel supports fail along with the nozzle failure, it is expected that no more than 10-15 percent of the carbon would be displaced from the vessel. This percentage of the carbon is assumed to be from the top of the first bed, and therefore, would contain virtually all of the activity stored in the beds. Because iodine is strongly bonded to the charcoal, it is not expected to be removed by exposure to the air. However, the conservative assumption is made that 1 percent of the iodine activity contained in the adsorber tank is released to the vault containing the offgas equipment. Additionally, the conservative assumption is made that 1 percent of the solid daughters retained in the charcoal is released.

It is further assumed that 10 percent of the noble gas activity is released from a failed vessel because of the small fraction of carbon exposed to the air. Measurements made at KRB (plant in Germany) indicate that offgas is about 30 percent richer in Kr than air. Therefore, if this carbon is exposed to air, it will eventually reach equilibrium with the noble gases in the air. However, the first few inches of carbon will blanket the underlying carbon from the air.

(b) Prefilters

Because of the design features of the prefilter vessel, (approximately 24 inch diameter, 4 feet height, 350 psig design pressure, 1/2 inch wall thickness, and collapsible filter media) a failure mechanism cannot be postulated that will result in emission of filter media or daughter products from this vessel. However, to illustrate the consequences of a radioactivity loss from this vessel, one percent release of particulate activity is assumed.

(c) Holdup pipe

Pipe rupture and depressurization of the holdup pipe is considered. For the design basis analysis, 100 percent of the noble gases and all of the remaining solid daughters after a 60 percent washout are assumed to be released.

2. Fission Product Transport to the Environment

The transport pathway consists of direct release of fission products from the failed component to the environment through the building ventilation system based on the design basis release fractions given in <Table 15.7-4>. The inventory of activities (design basis values) in each component, before the assumed failure, is presented in <Table 15.7-3a>. The release rates due to the continued operation of the SJAE are given in <Table 15.7-3>.

3. Results

Dose consequences due to failure of the worst single component [the holdup pipe] and assuming the SJAE continues to pump for 30 minutes after the break are presented in <Table 15.7-5>. The doses are a small fraction of the limits specified in <10 CFR 100> (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <10 CFR 50.67>.

b. Realistic Analysis

The realistic analysis is still a conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are described in (Reference 5). Specific values of parameters used in the evaluation are presented in <Table 15.7-6>.

1. Fission Product Release Assumptions

The activity in the offgas system is based on normal operating conditions of 30 scfm air inleakage and 100,000 μCi/sec noble gas after 30 minutes delay.

The activity stored in the various components before failure is given in <Table 15.7-3a> (Normal/Realistic Values).

The assumed release fractions for the realistic analysis are found in <Table 15.7-4>. The basis for the failure assumptions of those components which could have the worst dose consequence are as follows:

(a) Charcoal Adsorbers and Desiccant Vessel

Assumptions are the same as the design basis analysis except for the solid daughters. There is no reason to believe that any of the solid daughter products formed and retained within the micropore structure of the carbon will be released. Hence, no such release is assumed for the realistic analysis.

(b) Holdup pipe

Pipe rupture and depressurization of the pipe is considered. Normally, the pipe will operate at less than 16 psia and depressurize to 14.7 psia. The possible loss of solid daughters and noble gases is conservatively taken as 20 percent. The model used assumes retention and washout of 60 percent of the particulate daughters for the calculation of the holdup pipe inventory.

(c) Prefilter

Same as design basis analysis.

2. Fission Product Transport to the Environment

The release of activity to the environment is determined by applying the Realistic release fractions in <Table 15.7-4> to the Normal Value inventories in <Table 15.7-3a>. The release rates from the holdup pipe due to the continued operation of the SJAE are given in <Table 15.7-3>. The release of the indicated radioisotopes is assumed to continue for 30 minutes after the postulated break at the rates indicated in this table.

3. Results

The calculated exposures for the realistic analysis <Table 15.7-7> are due to the failure of the holdup pipe and continued operation of the SJAE for 30 minutes after the postulated failure.

15.7.1.2 Malfunction of Main Turbine Gland Sealing System

15.7.1.2.1 Identification of Causes and Frequency Classification

Plausible malfunctions of the turbine gland sealing system include the failure of the steam seal evaporator and its backup steam supply, failure of the steam packing exhauster fan and excessive pressure in the steam seal header.

This event is categorized as a limiting fault.

15.7.1.2.2 Sequence of Events and Systems Operation

It is assumed that the system fails near the 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.

The operator should initiate normal shutdown of the reactor to reduce the gaseous activity being discharged. A loss of main condenser vacuum will result in a turbine trip and reactor shutdown.

See <Appendix 15A> for further details on single failures and operator errors.

15.7.1.2.3 Core and System Performance

The failure of this power-conversion system does not directly affect the nuclear steam supply system (NSSS). It will, of course, lead to decoupling of the NSSS and the power-conversion system.

The tripping of the main turbine via main condenser signals will result in an anticipated operational transient examined earlier in <Chapter 15>.

This failure has no applicable effect on the core or the NSSS safety performance.

15.7.1.2.4 Barrier Analysis

This release occurs outside the containment and does not involve any barrier integrity aspects. However, a discussion of the release of the radioactivity to the environment is presented in order to assess the radiological impact relative to applicable safety limits.

15.7.1.2.5 Radiological Consequences

Failure of the steam seal evaporator and its backup steam supply would result in air leakage through the low pressure shaft seals to the condenser and in the discharge of a small amount of contaminated steam from the high pressure shaft seals to the steam packing exhauster. The loss of seal steam to the low pressure seals requires that the turbine be shut down to prevent excessive cooling of the turbine shaft. The small amount of contaminated steam that would be discharged to the atmosphere during the short period before the turbine shutdown is assumed to be inconsequential.

Failure of the steam packing exhauster fan results in the escape of clean steam from the high pressure and low pressure shaft seals. The most undesirable result of operating in this condition is that some condensate from the escaping seal steam could leak into the lube oil system.

Excessive pressure in the sealing steam header as a result of a malfunction of the seal steam evaporator or the backup steam supply valve is prevented by a relief valve so that there is no detrimental effect on the operation of the shaft seals.

15.7.1.3 Failure of Main Turbine Steam Air Ejector Lines

15.7.1.3.1 Identification of Causes and Frequency Classification

Those events which could cause a failure in the main turbine steam air ejector system are:

a. Failure of the steam line to the air ejectors.

b. Failure of the air ejector suction line.

c. Failure of the air ejector discharge line to the offgas system.

In each of these failures it is assumed that the worst case condition exists and that the failure is in a section of line common to both air ejectors so as to negate the use of the standby air ejector.

This event is categorized as a limiting fault.

15.7.1.3.2 Sequence of Events and Systems Operation

<Table 15.7-31> lists the sequence of events.

Failure of the steam line to the air ejectors would result in loss of condenser vacuum and the discharge of radioactive steam to the atmosphere. The high air activity would result in an alarm on the atmospheric radiation monitors and loss of condenser vacuum would result in a turbine trip and a reactor scram.

Failure of the air ejector suction line would result in the loss of condenser vacuum which would result in a turbine trip and a reactor scram.

Failure of the air ejector discharge line to the offgas system would result in the discharge of radioactive gas into the atmosphere. This failure would result in a "loss-of-flow to the offgas system" after which the operator should initiate shutdown of the reactor to reduce the amount of gaseous activity being discharged to the atmosphere.

See <Appendix 15A> for further details on single failures and operator errors.

15.7.1.3.3 Core and System Performance

The failure of this power-conversion system does not directly affect the nuclear steam supply systems (NSSS). It will, of course, lead to decoupling of the NSSS and the power-conversion system.

Tripping of the main turbine via main condenser pressure signals will result in an anticipated operational transient examined earlier in <Chapter 15>.

This failure has no applicable effect on the core or the NSSS safety performance.

15.7.1.3.4 Barrier Analysis

This release occurs outside the containment and does not involve any barrier integrity aspects. However, a discussion of the release of the radioactivity to the environment is presented in order to assess the radiological impact relative to applicable safety limits.

15.7.1.3.5 Radiological Consequences

15.7.1.3.5.1 Fission Product Release

Of the three lines considered to fail in <Section 15.7.1.3.1>, the most severe radiological consequences offsite would be due to failure of the air ejector discharge line. The assumptions used in calculating the amount of gaseous radioactive materials released from this break follow:

a. Loss of flow in the offgas system will be indicated by an alarm in the control room. It is conservatively assumed that it takes 15 minutes after the break for the operators to shut down the plant.

b. During this period, the noble gas activity is conservatively assumed to be released from the break at the same rate it is released from the reactor vessel (i.e., no credit is taken for decay of the isotopes while in transit from the reactor to the point of the break).

c. The iodine activity released from the break is based on 2 percent carryover from the reactor water to the steam and a mass loss of approximately 1,725 pounds through the break before termination of the accident.

d. No credit is taken for plateout of radioiodine.

e. It is assumed that an equilibrium coolant concentration consistent with an offgas release rate of 100,000 μCi/second after 30 minutes exists prior to the accident.

15.7.1.3.5.2 Fission Product Transport to the Environment

The following assumptions are used in calculating the amount of activity released to the environs:

a. It is conservatively assumed that all of the iodine and noble gas activity released from the break is instantaneously released to the environment via the offgas building ventilation system where it is treated by a series of roughing, HEPA and charcoal filters.

b. The charcoal filter efficiency is assumed to be 90 percent for the removal of iodine.

c. All other assumptions relating to this event are tabulated in <Table 15.7-8>. The activity released to the environment is presented in <Table 15.7-9>.

15.7.1.3.5.3 Results

The calculated exposures for this analysis are presented in <Table 15.7-10> and are a very small fraction of <10 CFR 100> guidelines (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <10 CFR 50.67>).

15.7.2 RADIOACTIVE LIQUID WASTE SYSTEM FAILURES (RELEASE TO ATMOSPHERE)

This accident is not affected by the reload analysis.

15.7.2.1 Identification of Causes and Frequency Classification

The liquid radwaste treatment systems are classified as quality Group D. Radioactive releases considered include rupture of radwaste tanks, equipment malfunction or small leaks in the system process lines that transport liquid radwaste.

15.7.2.2 Sequence of Events and Systems Operations

The sequence of events and systems operations is as follows:

a. Event begins - postulated failure of system component occurs.

b. Area radiation alarms alert plant personnel. No credit for any operator action is considered in evaluating this event.

15.7.2.3 Core and System Performance

This event has no effect on the core or NSSS safety functions.

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 failure of the liquid radwaste system are given in <Table 15.7-11>, and the radioactive inventory in the system is listed in <Table 15.7-12>.

The amount of activity airborne from this event is negligible. <Section 15.7.3> evaluates the radiological consequences due to liquid releases.

15.7.3 POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID-CONTAINING TANK FAILURES

This accident is not affected by the reload analysis.

15.7.3.1 Identification of Causes and Frequency Classification

It is considered highly improbable for significant cracks to develop in the Seismic Category I safety class buildings containing radioactive waste materials and equally improbable for a waste tank to fail. However, it is postulated that an unspecified event causes a failure of a tank in the radwaste building and subsequent failure of the radwaste building.

This failure is classified as a limiting fault.

15.7.3.2 Sequence of Events and Systems Operation

The sequence of events and systems operations follows:

a. Failure occurs - contents of failed component is released into the radwaste building.

b. Area radiation alarms alert plant personnel.

c. For the evaluation of this failure no credit is taken for operator action and it is assumed that liquid leaks from the building into the ground. However, the plant operating procedures require that upon indication of a seismic event or unexpected high radiation levels in the radwaste building, the service and backup underdrain pumps are manually tripped with a positive, safety-related cutoff switch. In addition, radiation monitors located in the underdrain system effluent discharge will alarm in the control room and automatically stop the service and backup underdrain pumps upon detection of high radioactivity <Section 11.5>. The radwaste building is then inspected to determine whether a gross failure of any components housing radioactive liquids has occurred. If no failure has occurred, the underdrain pumps are reactivated. If failure is discovered, the pumps will not be reactivated until it can be determined that contaminated water has not entered the underdrain system. If radioactivity has been released to the underdrain system, the pumps will not be reactivated, and the groundwater will be allowed to rise to the gravity drain discharge system (Elevation 582.6’).

15.7.3.3 Core and System Performance

The failure of these radwaste components does not directly affect the core or NSSS safety function.

15.7.3.4 Barrier Performance

This event does not involve any containment barrier integrity.

15.7.3.5 Radiological Consequences

The following methods and assumptions are applied in the analysis of the offsite exposures resulting from the release of liquids to the groundwater from a failure in the liquid radwaste system:

a. For each piece of failed equipment, it is conservatively assumed that 80 percent of the design capacity is immediately released from the building, i.e., no credit is taken for retention of any of the released liquids in the Seismic Category I radwaste building. The radioactive inventories in the system are listed in <Table 15.7-12>. The system is operated as provided in (Reference 8).

b. After the liquids leave the building, they enter the porous concrete mat and are mixed with clean, non-contaminated groundwater. Activity bound up in solids such as resins is assumed to remain in the Radwaste Building.

c. The time required for groundwater to reach gravity drain discharge elevation is approximately 6.25 days. During this time, credit is taken for radioactive decay of the released radioisotopes. The quantity of clean groundwater available for dilution at this time is conservatively calculated to be 720,000 gallons.

d. This decayed and diluted mixture then drains, via the gravity drain system to the emergency service water pumphouse bay area at a rate of 80 gpm (design groundwater flow rate into underdrain system).

e. For all postulated tank ruptures, the isotopic concentrations are then further reduced by mixing with a conservative emergency service water flow of 19,000 gpm. No credit is assumed for any dilution with the non-contaminated water in the emergency service water pumphouse bay.

f. The emergency service water pump would normally discharge to Lake Erie, via the plant discharge tunnel, where the radioactive liquids would be well mixed (diluted) with the non-contaminated lake water. In the event of a collapse or blockage of the non-seismic portion of the ESW discharge piping, however, the emergency service water system will discharge via a standpipe to the yard outside of the auxiliary building. At this location a grass swale is provided to carry the flow from the auxiliary building area, between the cooling towers, to the minor stream diversion on the east side of the plant. This water then flows in the stream diversion over the sediment control dam and ultimately enters Lake Erie at the shoreline. If this path were used by the effluents following the postulated accident, dilution of the radioactive liquids would occur in the minor stream diversion and in Lake Erie with the non-contaminated lake water. In calculating the resultant individual exposures from this pathway, it was conservatively assumed that no dilution occurred in the grass swale.

g. No credit is taken for any settling or plating out of the radioisotopes.

h. The dose conversion factors for the isotopes considered are taken from (Reference 7).

i. For the purposes of calculating the average fraction of <10 CFR 20, Appendix B> effluent concentration, the total release of the radioisotopes into the lake is averaged over a one year period. (Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the <10 CFR 20> regulations prior to October 4, 1993. Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised <10 CFR 20> dated October 4, 1993.)

j. The resultant ingestion exposure is calculated for an individual drinking potentially contaminated water for a period of one year at a rate of 2,000 cc/day. The isotopic concentrations in this water are conservatively assumed to be the concentrations calculated at the nearest drinking water intake.

k. Credit is taken for dilution in the lake to the nearest drinking water intake (0.5 miles ENE of the plant) as presented in Table 5.1-10 of the Perry Nuclear Power Plant Environmental Report (Operating License Stage).

The resulting exposures from liquid releases to the groundwater are presented in <Table 15.7-14>.

The individual isotopic concentrations and fraction of effluent water concentrations (FEWC) for the radionuclides released by a postulated failure of the condensate filter backwash settling tank are given in <Table 15.7-15a>. (Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the <10 CFR 20> regulations prior to October 4, 1993. Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised <10 CFR 20> dated October 4, 1993.) A summary of the total isotopic concentration and total FEWC for each component postulated to fail is given in <Table 15.7-16>.

As indicated by these results the concentrations are well within the <10 CFR 20> effluent concentration limits for unrestricted areas <10 CFR 20, Appendix B>. Likewise, the resultant exposures are a small fraction of acceptable limits for this type of event.

15.7.4 FUEL HANDLING ACCIDENT OUTSIDE CONTAINMENT

This accident is not reanalyzed as part of the reload analyses as the fuel handling accident inside containment analysis is bounding <Section 15.7.6>. Radiological exposures were recalculated incorporating GNF2 fuel resulting in exposures within the licensing basis limits of 6.3 rem TEDE (offsite) and 5 rem TEDE (control room).

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 onto stored 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

15.7.4.2.1 Sequence of Events

The most severe fuel handling accident from a radiological release viewpoint is the drop of a channeled spent fuel bundle onto unchanneled spent fuel in the spent fuel racks in the fuel handling building. During the spent fuel dry cask loading operations, the fuel handling bridge is used to transfer spent fuel from the spent fuel pool to the MPC canister located in the HI-TRAC at the cask loading area of the cask pit pool. The top of the MPC canister fuel basket is slightly higher in elevation than the spent fuel racks located in the spent fuel pool. Therefore the actual drop height of a fuel bundle over the MPC-68 (less than 8 feet) is comparable to the actual drop height of a fuel bundle over the spent fuel in the spent fuel pool racks. The sequence of events which is assumed to occur is as follows:

Event	Approximate Elapsed Time
a. Channeled fuel bundle is being handled by a crane over spent fuel pool. Crane motion changes from horizontal to vertical and the fuel grapple releases, dropping the bundle. The channeled bundle strikes unchanneled bundles in the rack.	0

	Event	Approximate Elapsed Time
b.	Some rods in both the dropped and struck bundles fail, releasing radioactive gases to the pool water.	0
c.	Gases pass from the water to the fuel handling building.	0
d.	The fuel handling building ventilation system high radiation alarm alerts plant personnel.	<1 Min

15.7.4.2.1.1 Identification of Operator Actions

The accident analysis does not assume any operator actions for the mitigation of this event.

15.7.4.2.2 Systems Operation

Operation of plant, reactor protection or ESF systems is not taken into account.

15.7.4.2.3 The Effect of Single Failures and Operator Errors

The FHAES is designed to single failure criteria and safety requirements. No credit is taken for the FHAES.

15.7.4.3 Core and System Performance

15.7.4.3.1 (Deleted)

15.7.4.3.2 (Deleted)

15.7.4.3.3 (Deleted)

(Deleted)

(Deleted)

15.7.4.3.4 Results

The total number of failed fuel rods for the bounding fuel handling accident is given in <Section 15.7.6.3>.

15.7.4.4 Barrier Performance

This failure occurs in the fuel handling building outside the normal barriers (RCPB and containment). Therefore, this section is not directly applicable. The transport of fission products to the environment is discussed in the next section.

15.7.4.5 Radiological Consequences

The analysis is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet <Regulatory Guide 1.183> dose criteria. This analysis is referred to as the "design basis analysis," and is based on a 24 hour radioactive decay period of the fuel. The design basis analysis is based on assumptions and methods contained in NRC <Regulatory Guide 1.183>.

When comparing a fuel handling accident inside containment with a fuel handling accident in the Fuel Handling Building, the inside containment event would be bounding due to higher kinetic energy and the greater number of fuel pins damaged. The assumption is made that the activity which escapes from the pool is released immediately and directly to the environment. Thus, for this analysis, refer to <Section 15.7.6.4> on the bounding analyses for the fuel handling accident inside containment.

The fission product inventory in the fuel rods assumed to be damaged is based on operation at 3,833 MWt.

15.7.4.5.1 Design Basis Analysis Assuming 24 Hour Radioactive Decay of the Fuel

15.7.4.5.1.1 Fission Product Release from Fuel

Refer to <Section 15.7.6.4.1>.

15.7.4.5.1.2 Fission Product Transport to the Environment

Refer to <Section 15.7.6.4.1>.

15.7.4.5.1.3 Results

The calculated exposures for the bounding fuel handling accident are presented in <Table 15.7-35> and are within the dose criteria of <Regulatory Guide 1.183>.

15.7.5 SPENT FUEL CASK DROP ACCIDENTS

This accident is not affected by the reload analysis.

15.7.5.1 Cask Drop from Transport Vehicle

In the unlikely event that the fuel transportation cask falls from the transport vehicle, the maximum height which the cask will drop should be in general less than 10 ft. Since the cask is designed to withstand a 30 ft drop onto a non-yielding surface without failure, the fall from the transport vehicle will cause no failure of the cask.

15.7.5.2 Cask Drop from Crane

The Mark III containment design includes a separate fuel handling building. The spent fuel storage pools in this building are arranged so that the overhead crane which handles the cask cannot possibly move the cask above the spent fuel storage pool. This precludes the possibility of the cask falling on the stored spent fuel bundles. Also, the pools are arranged so that a rupture of the cask loading pool floor will not drain water from the spent fuel storage pool if the pool gate is installed. The cask loading area design and operating procedures are specifically formulated so that a cask drop will not result in failure of the cask.

As discussed in <Section 9.1.4.2.2.2>, Fuel Handling Area Crane, the hoist for this crane's main hook that handles spent fuel casks has been upgraded to single-failure-proof in accordance with applicable guidelines of NRC <NUREG-0554> (Single-Failure-Proof Cranes for Nuclear Power Plants, May 1979)and NRC <NUREG-0612> (Control of Heavy Loads at Nuclear Power Plants, July 1980) to support spent fuel dry storage cask handling activities. With the hoist for the crane's main hook qualified as single-failure-proof, a cask drop accident is not a credible event and need not be postulated.

15.7.6 FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

The radiological exposures were recalculated incorporating GNF2 Fuel and power uprate analysis resulting in exposures within the licensing basis limits of 6.3 rem TEDE (offsite) and 5 rem TEDE (control room).

The analysis in <Section 15.7.6.4> assumes a 24 hour decay time prior to the accident occurring, i.e., the fuel has not been in a critical reactor core within the previous 24 hours. The 24 hour value forms the definition of "recently irradiated fuel" as identified in the Technical Specifications. This analysis is reviewed each cycle to verify that the 24 hour assumption is valid. As a result of this review, the definition of "recently irradiated fuel" may change, and would be reflected herein. Note that although the Technical Specifications retain the term "recently irradiated fuel" and could be interpreted to permit fuel handling before 24 hours if certain buildings and ventilation systems are operable, this is not the case. Handling of "recently irradiated fuel" is prohibited, because no dose calculations exist to address a fuel handling accident within the first 24 hours after the core is subcritical.

15.7.6.1 Identification of Causes and Frequency Classification

15.7.6.1.1 Identification of Causes

Various mechanisms for fuel failure during refueling have been investigated. Procedural controls, backed up by the refueling interlocks, impose restrictions on the movement of refueling equipment and control rods, to prevent an inadvertent criticality during refueling operations. In addition, the reactor protection system is able to initiate a reactor scram in time to prevent fuel damage for errors or malfunctions occurring during planned criticality tests with the reactor vessel head off. It is concluded that the accident that results in the release of significant quantities of fission products during this mode of operation with the greatest analyzed radiological consequences is one resulting from the accidental dropping of a fuel bundle onto the top of the core.

The movement of non-fuel items over irradiated fuel inside containment will be administratively controlled such that the radiological consequences associated with their accidental dropping with or without primary or secondary containment will remain bounded by that of a dropped fuel bundle.

15.7.6.1.2 Frequency Classification

This event has been categorized as a limiting fault.

15.7.6.2 Sequence of Events and Systems Operation

15.7.6.2.1 Sequence of Events

The sequence of events which is assumed to occur is as follows:

	Event	Approximate Elapsed Time
a.	Channeled fuel bundle being removed from reactor vessel by the refueling crane. Fuel bundle is dropped from maximum height allowed by the refueling equipment. Fuel bundle strikes core.	0
b.	Some rods in both dropped and struck bundles fail releasing radioactive gases to pool water.	0
c.	Gases pass from water immediately to building.	0
d.	Containment vessel and drywell purge ventilation system isolates due to high radiation signal. Not credited in the dose analysis.	20 sec

15.7.6.2.1.1 Identification of Operator Actions

The accident analysis does not assume any operator actions for the mitigation of this event.

15.7.6.3 Core and System Performance

The methods used for this evaluation are the same as those presented in <Section 15.7.4.3>, for the initial cycle. The analysis credits 150 failed fuel rods based on GE methodology for GNF2 bundles and a triangular fuel handling mast.

15.7.6.4 Radiological Consequences

Three separate radiological cases are provided for this accident:

a. The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet <Regulatory Guide 1.183> dose criteria. This analysis is based on a 24 hour radioactive decay period of the fuel. This is considered the base case and takes no credit for engineered safety features, isolations, or filtration.

b. The second case is identical to the first case but was performed to determine the effect of control room isolation and fresh air intake. The control room dose was calculated assuming that once the available activity is introduced into the control room, the fresh air intake was isolated without any additional inleakage. At two hours, outside ventilation air is re-established and continues for the remainder of the 30-day dose analysis.

c. The third case is also identical to the first case but was performed to determine the effect of control room isolation and emergency recirculation filtering. The control room dose was calculated assuming that once the available activity is introduced into the control room, the fresh air intake was isolated without any additional inleakage. At two hours, the control room emergency recirculation system was assumed to initiate and continue for the remainder of the 30-day dose analysis. A filtration efficiency for the charcoal of 80 percent was assumed in order to be consistent with <Table 15.6-14>.

NOTE: The second and third cases were performed to examine the flexibility the control room operators have in using ventilation, to ensure that there were no dose outliers. The results show that even if the operators take 2 hours to initiate an action, (i.e., re-initiate normal intake or utilize recirculation filtration) the doses remain below the licensing basis limits.

For all cases, the fission product inventory in the fuel rods assumed to be damaged is based on operation at 3,833 MWt. Specific values for parameters used are provided in <Table 15.7-32>.

The exposures were calculated utilizing the computer code RADTRAD (Reference 16).

An additional scenario was reviewed in the event a fuel bundle was dropped when travelling through the refueling shield inside containment. This reduces the amount of water coverage resulting in a lower decontamination factor. The resultant dose is less than the cases described above when considering that the damage is limited to the number of rods in one fuel bundle.

15.7.6.4.1 Design Basis Analysis Assuming 24 Hour Radioactive Decay of the Fuel

a. Fission Product Release from Fuel

The fission product activity released from the fuel damaged as a result of a fuel handling accident is calculated using the methods below:

1. The fuel rod gap activity is assumed to consist of 5% of the total halogen and noble gas activity in the rods at the time of the accident, except for KR-85, which is assumed to be 10% and I-131 which is assumed to be 8%.

2. Twelve percent of the alkali metals are available for release but are retained in the water.

A total of 150 fuel rods fail as a result of the accident given a core loaded with GNF2 fuel, while using a triangular fuel handling mast.

b. Fission Product Activity Released

The following assumptions and initial conditions are used in calculating the fission product activity released to the environs.

1. The iodine gap inventory is composed of elemental species (99.85 percent) and organic species (0.15 percent).

2. The minimum water depth between the top of the damaged fuel rods and the containment pool surface is 23 feet.

3. The pool decontamination factors for the elemental and organic species of iodine are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5 percent of the total iodine released from the damaged rods is retained by the pool water). This difference in decontamination factors for elemental and organic iodine species results in the iodine above the fuel pool being composed of 57 percent elemental and 43 percent organic species.

4. The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water.

5. The effects of plateout and fallout are neglected.

6. The activities within the failed rods are assumed to have been decayed 24 hours prior to the accident.

7. All activity released from the pool to the containment is released directly to the environment (i.e., filtering and containment integrity is not credited).

Based on these assumptions, the activity released from the pool to the environment is listed in <Table 15.7-34>.

c. Results

Based on these assumptions, the Total Effective Dose Equivalent (TEDE) at the exclusion boundary, low population zone and control room are summarized in <Table 15.7-35>. The doses at these distances are within the licensing basis limits of 6.3 rem TEDE (offsite) and 5 rem TEDE (control room).

15.7.7 REFERENCES FOR SECTION 15.7

1. Nguyen, D., "Realistic Accident Analysis - The RELAC Code," October 1977, (NEDO-21142).

2. Bunch, F. D., "Dose to Various Body Organs from Inhalation or Ingestion of Soluble Radionuclides," IDO-12054, AEC Research and Development Report, TID-4500, August 1966.

3. N. R. Horton, W. A. Williams, J. W. Holtzclaw, "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor," March 1969, (APED 5756).

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

5. <Regulatory Guide 1.109>, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the purpose of evaluating Compliance with <10 CFR 50, Appendix I>, Revision 1, October 1977.

6. <Regulatory Guide 1.183>, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

7. Environmental Protection Agency (EPA) Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion." Second printing, 1989.

8. Perry calculation 3.1.3.3 Revision 6 G50 Declassification - Source Terms.

9. PNPP Calculation G58-H-HI-2084201, "Analysis of Mechanical Accidents During MPC Loading in the Perry Cask Pit Pool."

10. PNPP Specification PRS-1917, “Perry Nuclear Power Plant Spent Fuel Dry Storage Project Cask & Ancillaries Procurement Specification.”

11. PNPP Drawing 515-0083-00000, Fuel Handling Building Liner Plate Details for Fuel Cask Storage Pit.

12. HOLTEC International Report No. HI-2002444, Docket 72-1014, Final Safety Analysis Report for the HOLTEC International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System), Revision 7, August 2008.

13. 10 CFR 72 Certificate of Compliance No. 1014, HI-STORM 100 Cask System, Amendment No. 5, Effective Date July 14, 2008, including attached Appendix A - Technical Specifications for the HI-STORM 100 Cask System, and attached Appendix B – Approved Contents and Design Features for the HI-STORM 100 Cask System, with accompanying NRC Safety Evaluation Report.

14. General Electric (GE) Drawing 107E1592, Fuel Bundle for GE 14 BWR Fuel, Rev. 1.

15. PNPP Drawing 23_-0271-00003-001 PAR Systems Fuel Storage Racks.

16. S. L. Humphreys et. al., “RADTRAD: A Simplified Model For Radionuclide Transport and Removal and Dose Estimation.” NUREG/CR-6604, USNRC April 1998.

TABLE 15.7-1

PROBABLE SEQUENCE OF EVENTS FOR MAIN CONDENSER GAS 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

DELETED

TABLE 15.7-2 (Continued)

DELETED

TABLE 15.7-3

RECHAR SYSTEM HOLDUP PIPE BREAK ACTIVITY RELEASE RATE

INPUT: 100,000 µCi/sec Mix

Isotope	µCi/sec	Isotope	µCi/sec	Isotope	µCi/sec
T-3	1.893E-2	SE-88	0.	ZR-95	2.841E-15
N0-13	2.108E-3	BR-88	2.760E-11	NB-95m	6.113E-17
A-13m	1.070E-1	KR-88	2.144E+4	NB-95	2.758E-15
NO-13	2.319E-12	RB-88	1.923E+3	ZR-97	2.179E-14
C-14	3.488E-2	SE-89	0.	NB-97m	1.769E-14
N0-16	3.874E-14	BR-89	1.157E-34	NB-97	3.089E-13
A-16m	3.320E-18	KR-89	4.045E+4	ZR-99	0.
NO-16	2.826E-29	RB-89	1.016E+4	NB-99m	2.072E-13
N0-17	2.484E-32	SR-89	3.465E-1	NB-99	5.507E-22
A-17m	2.718E-36	Y-89m	5.511E-28	MO-99	2.258E-14
		BR-90	0.	TC-99m	1.133E-11
F-18	2.821E-2	KR-90	1.908E+1	TC-99	1.627E-18
O-19	5.452E-5	RB-90m	5.417E+2	MO-101	7.727E-13
NA-24	1.557E-13	RB-90	3.442E+3	TC-101	9.883E-12
P-32	1.562E-15	SR-90	1.572E-3	MO-102	7.224E-13
CR-51	3.996E-14	Y-90m	2.792E-19	TC-102m	3.571E-12
MN-54	3.119E-15	Y-90	8.503E-7	TC-102	0.
MN-56	3.870E-12	BR-91	0.	TC-103	1.152E-14
CO-58	3.976E-13	KR-91	4.464E-11	RU-103	1.598E-15
FE-59	6.381E-15	RB-91	1.960E+1	RH-103m	1.511E-13
CO-60	3.764E-14	SR-91	2.814E+0	MO-104	6.388E-14
NI-65	2.320E-14	Y-91m	6.607E-2	TC-104	3.712E-12
ZN-65	1.546E-16	Y-91	2.820E-5	MO-105	2.816E-15
ZN-69m	2.335E-15	BR-92	0.	TC-105	1.427E-12
AS-83	1.758E-23	KR-92	0.	RU-105	3.224E-14
SE-83m	5.018E-15	RB-92	1.541E-31	RH-105m	7.955E-17
SE-83	2.691E-14	SR-92	4.758E-5	RH-105	2.075E-15
BR-83	1.257E-2	Y-92	4.682E-7	TC-106	1.376E-16
KR-83m	4.205E+3	KR-93	0.	RU-106	1.087E-16
AS-84	7.202E-38	RB-93	2.757E-28	AG-110m	4.696E-15
SE-84	1.375E-13	SR-93	3.335E-7	SB-129	8.072E-15
BR-84m	1.873E-4	Y-93	1.645E-9	TE-129m	4.090E-16
BR-84	2.380E-2	ZR-93	1.148E-20	TE-129	4.958E-14
AS-85	0.	NB-93m	5.175E-22	I-129	7.961E-11
SE-85m	2.328E-20	KR-94	0.	SB-131	2.675E-13
SE-85	2.277E-16	RB-94	0.	TE-131m	1.746E-15
BR-85	4.016E-3	SR-94	2.628E-13	TE-131	3.121E-13
KR-85m	6.446E+3	Y-94	1.384E-12	I-131	9.498E-3
KR-85	1.669E+1	KR-95	0.	XE-131m	3.446E+1
AS-87	0.	RB-95	0.	TE-132	1.371E-14

TABLE 15.7-3 (Continued)

Isotope	μCi/sec	Isotope	μCi/sec	Isotope	μCi/sec
SE-87	3.154E-38	SR-95	5.534E-17	I-132	1.165E-1
BR-87	9.678E-5	Y-95	1.531E-12	SB-133	1.973E-13
KR-87	2.112E+4				
TE-133m	2.606E-13	CS-137	1.822E-2	LA-142	3.552E-8
TE-133	5.813E-13	BA-137m	1.830E-3	XE-143	0.
IM-133	1.532E-19	XE-138	8.926E+4	CS-143	0.
I-133	6.941E-2	CS-138m	1.046E-13	BA-143	6.023E-22
XE-133m	3.391E+2	CS-138	5.981E+3	LA-143	1.221E-12
XE-133	8.274E+3	XE-139	1.376E+2	CE-143	1.431E-14
TE-134	5.219E-13	CS-139	6.160E+3	PR-143	3.537E-15
IM-134	2.060E-3	BA-139	1.939E+2	XE-144	0.
I-134	2.000E-1	XE-140	1.334E-5	CS-144	0.
XE-134m	0.	CS-140	9.379E+1	BA-144	8.949E-22
I-135	1.101E-1	BA-140	2.898E-1	LA-144	1.356E-13
XE-135m	2.733E+4	LA-140	2.492E-4	CE-144	1.896E-15
XE-135	2.470E+4	XE-141	0.	ND-147	1.472E-15
CS-135m	1.470E-15	CS-141	3.358E-8	P-147m	3.685E-16
CS-135	4.303E-8	BA-141	7.230E-5	ND-149	3.275E-14
TE-136	4.662E-19	LA-141	7.350E-7	P-149m	1.620E-15
I-136m	2.613E-5	CE-141	3.300E-11	W-187	2.397E-13
I-136	1.096E-3	XE-142	0.	NP-239	1.664E-11
I-137	6.587E-8	CS-142	0.		
XE-137	6.439E-4	BA-142	1.270E-7	Total	3.388E+5

TABLE 15.7-3a

OFFGAS SYSTEM COMPONENT INVENTORY ACTIVITIES

DESIGN BASIS AND NORMAL VALUES[1]

Design Basis Values in Ci[2]

OFFGAS PREHEATER

N16	9.81E-00	KR87	8.20E-02	TOTAL = 1.62E+01	
KR88	8.24E-02	KR89	6.25E-01	KR90	1.40E-00
RB90	2.13E-02	KR91	6.82E-01	RB91	4.35E-02
XE133	2.87E-02	XE135M	1.22E-01	XE135	9.49E-02
XE137	7.66E-01	XE138	4.35E-01	XE139	1.21E-00
XE140	8.76E-01	CS140	4.87E-02		

RECOMBINER

N16	1.10E+01	AM16	3.99E-02	TOTAL = 1.74E+01	
KR87	1.36E-01	KR88	9.67E-02	KR89	1.25E-01
KR90	1.29E-00	RB90	2.95E-02	KR91	7.23E-01
RB91	5.84E-02	XE133	3.37E-02	XE135M	1.43E-01
XE135	1.11E-01	XE137	8.95E-01	XE138	5.10E-01
XE139	1.38E-00	XE140	9.60E-01	CS140	6.62E-02

OFFGAS CONDENSER

N16	1.08E+02	AM16	4.25E-01	TOTAL = 3.90E+02	
KR85M	1.52E-00	KR87	5.02E-00	KR88	5.04E-00
KR89	3.49E+01	RB89	1.47E-00	KR90	4.15E+01
RB90M	6.89E-01	RB90	8.49E-01	KR91	9.60E-00
RB91	6.59E-00	XE133	1.77E-00	XE135M	7.31E-00
XE135	5.82E-00	XE137	4.34E+01	XE138	2.61E+01
XE139	4.83E+01	CS139	3.42E-00	XE140	1.81E+01
CS140	1.03E+01	N13	6.61E-01		

WATER SEPARATOR

N13	3.10E-01	N16	8.36E-01	TOTAL = 9.50E+01	
KR83M	5.29E-01	KR85M	9.62E-01	KR87	2.83E-00
KR88	2.86E-00	KR89	1.71E+01	RB89	3.95E-01
KR90	9.64E-00	RB90M	8.45E-02	RB90	1.06E-00
KR91	1.75E-01	RB91	6.62E-02	XE133	9.72E-01
XE135M	3.95E-00	XE135	3.21E-00	XE137	2.13E+01
XE138	1.38E+01	CS138	1.53E-01	XE139	1.29E+01
CS139	9.84E-01	XE140	1.10E-00	CS140	3.11E-01

TABLE 15.7-3a (Continued)

OFFGAS HOLDUP PIPE

N13	7.52E-00	N16	3.97E-02	TOTAL = 4.01E+03	
KR83M	8.35E+01	KR85M	1.64E+02	KR87	3.89E+02
KR88	5.03E+02	RB88	4.99E+02	KR89	1.44E+02
RB89	1.60E+02	KR90	1.03E+01	RB90M	2.11E-00
RB90	1.68E+01	RB91	1.40E-01	XE133M	9.30E-00
XE133	2.13E+02	XE135M	1.65E+02	XE135	6.80E+02
XE137	2.19E+02	XE138	5.21E+02	CS138	5.12E+02
XE139	1.82E+01	CS139	3.06E+01	BA139	1.89E+01

COOLER CONDENSER

N13	2.86E-03	KR83M	8.67E-00	TOTAL = 3.12E+02	
KR85M	2.20E+01	KR87	3.26E+01	KR88	6.05E+01
RB88	3.36E+01	KR89	1.63E-01	RB89	5.22E-01
XE133M	1.45E-00	XE133	3.36E+01	XE135M	6.25E-00
XE135	8.00E+01	XE138	1.30E-00	CS138	1.63E+01
BA139	1.68E-00				

MOISTURE SEPARATOR

KR83M	2.86E-01	KR85M	7.52E-01	TOTAL = 8.89E-00	
KR87	1.05E-00	KR88	2.04E-00	XE133M	5.03E-02
XE133	1.09E-01	XE135M	3.14E-01	XE135	3.44E-00
XE138	2.80E-02				

OFFGAS PREFILTER - GAS COMPONENT

KR83M	1.63E-00	KR85M	4.84E-00	TOTAL = 5.61E+01	
KR87	5.39E-00	KR88	1.24E+01	XE133M	3.55E-01
XE133	8.23E-00	XE135	2.32E+01	XE138	3.75E-02

OFFGAS PREFILTER - ONE YEAR MEDIA COMPONENT

RB88	4.08E+01	CS138	1.07E+01	TOTAL = 4.12E+01

OFFGAS DRYER

KR83M	1.30E+01	KR85M	3.61E+01	TOTAL = 4.91E+02	
KR87	4.54E+01	KR88	9.55E+01	RB88	6.90E+01
XE133	5.88E+01	XE135	1.69E+02		

TABLE 15.7-3a (Continued)

CHARCOAL BEDS

KR83M	5.08E+01	KR85M	3.98E+02	TOTAL = 1.27E+04	
KR87	1.07E+02	KR88	6.16E+02	RB88	6.45E+02
XE133M	2.22E+02	XE133	6.60E+03	XE135	4.01E+03

AFTER-FILTER - GAS COMPONENT

XE131M	1.81E-02	XE133M	5.54E-03	TOTAL = 5.45E-00
XE133	5.21E-00	XE135	9.67E-02	

AFTER-FILTER - ONE YEAR MEDIA COMPONENT

RB88	1.33E-09	I133	1.52E-09	TOTAL = 8.69E-07
CS135	8.51E-07			

TABLE 15.7-3a (Continued)

Normal Values in Ci[3]

OFFGAS PREHEATER

N16	4.40E-00	KR87	1.83E-02	TOTAL = 6.66E-00	
KR88	1.79E-02	KR89	1.87E-01	KR90	4.37E-01
RB90	8.21E-03	KR91	1.91E-02	RB91	1.91E-02
XE133	6.62E-03	XE135M	2.67E-02	XE135	1.96E-02
XE137	2.18E-01	XE138	1.06E-01	XE139	4.56E-01
XE140	3.72E-01	CS140	2.07E-02		

RECOMBINER

N16	4.54E-00	AM16	1.99E-02	TOTAL = 7.10E-00	
KR87	2.14E-02	KR88	2.10E-02	KR89	2.19E-01
KR90	4.98E-01	RB90	1.14E-02	KR91	3.18E-01
RB91	2.57E-02	XE133	7.77E-03	XE135M	3.15E-02
XE135	2.30E-02	XE137	2.55E-01	XE138	1.24E-01
XE139	5.23E-01	XE140	4.08E-01	CS140	2.81E-02

OFFGAS CONDENSER

N16	4.83E+01	AM16	2.18E-01	TOTAL = 1.43E+02	
KR85M	3.25E-01	KR87	1.12E-00	KR88	1.09E-00
KR89	1.04E+01	RB89	4.38E-01	KR90	1.60E+01
RB90M	2.65E-01	RB90	1.23E-01	KR91	4.22E-00
RB91	1.90E-00	XE133	4.06E-01	XE135M	1.62E-00
XE135	1.20E-00	XE137	1.24E+01	XE138	6.35E-00
XE139	1.82E+01	CS139	1.29E-00	XE140	7.72E-00
CS140	4.37E-00				

WATER SEPARATOR

N13	7.02E-03	N16	6.77E-02	TOTAL = 2.76E-00	
KR83M	8.29E-03	KR85M	1.33E-02	KR87	4.56E-02
KR88	4.48E-02	KR89	3.89E-01	RB89	1.82E-02
KR90	3.58E-01	RB90M	1.04E-02	RB90	1.23E-01
KR91	1.34E-02	RB91	8.64E-02	XE133	1.67E-02
XE135M	6.50E-02	XE135	4.92E-02	XE137	4.70E-01
XE138	2.55E-01	CS138	5.31E-03	XE139	4.60E-01
CS139	5.26E-02	XE140	6.51E-02	CS140	1.35E-01

TABLE 15.7-3a (Continued)

OFFGAS HOLDUP PIPE

N13	1.20E-00	N16	3.06E-01	TOTAL = 2.91E+02	
KR83M	1.78E-00	KR85M	2.90E-00	KR87	9.69E-00
KR88	9.70E-00	RB88	2.78E-00	KR89	4.18E+01
RB89	1.55E+01	KR90	7.96E-00	RB90M	1.93E-00
RB90	1.80E+01	RB91	3.58E-00	XE133	3.66E-00
XE135M	1.21E+01	XE135	4.92E-02	XE137	5.64E+01
XE138	4.69E+01	CS138	8.08E-00	XE139	1.26E+01
CS139	1.43E+01	BA139	9.46E-01	XE140	5.91E-01
CS140	6.54E-00				

COOLER CONDENSER

N13	1.40E-01	KR83M	2.76E-01	TOTAL = 2.66E+01	
KR85M	4.57E-01	KR87	1.48E-00	KR88	1.52E-00
RB88	4.41E-01	KR89	2.30E-00	RB89	2.14E-00
RB90M	1.43E-01	RB90	8.13E-01	XE133	5.83E-00
XE135M	1.57E-00	XE135	1.73E-00	XE137	3.80E-00
XE138	6.00E-00	CS138	1.32E-00	CS139	1.32E-00
BA139	1.62E-00	CS140	1.67E-02		

MOISTURE SEPARATOR

N13	4.69E-03	KR83M	9.64E-03	TOTAL = 6.65E-01	
KR85M	1.60E-02	KR87	5.17E-02	KR88	5.30E-02
RB89	7.02E-02	XE133M	8.42E-04	XE133	2.05E-02
XE135M	5.37E-02	XE135	6.05E-02	XE137	1.91E-01
XE138	2.04E-01				

OFFGAS PREFILTER - GAS COMPONENT

N13	2.18E-02	KR83M	6.66E-02	TOTAL = 3.90E-00	
KR85M	1.11E-01	KR87	3.55E-01	KR88	3.68E-01
KR89	2.99E-01	XE133M	5.90E-03	XE133	1.43E-01
XE135M	3.40E-01	XE135	4.24E-01	XE138	1.28E-00

OFFGAS PREFILTER - ONE YEAR MEDIA COMPONENT

BR83	2.26E-02	RB88	2.84E-00	TOTAL = 2.34E+01	
RB89	2.90E-00	SR89	3.16E-00	I131	1.38E-00
I132	1.97E-01	I133	1.10E-00	I134	1.27E-01
I135	5.51E-01	CS137	2.50E-01	BA137M	6.04E-02
CS138	1.07E+01				

TABLE 15.7-3a (Continued)

OFFGAS DRYER

N13	2.18E-01	KR83M	4.79E-01	TOTAL = 3.17E+01	
KR85M	7.89E-01	KR87	2.56E-00	KR88	2.64E-00
RB88	2.12E-01	KR89	2.80E-00	RB89	2.62E-01
XE133M	4.21E-02	XE133	1.02E-00	XE135M	2.56E-00
XE137	4.93E-00	XE138	9.70E-00	CS138	4.38E-01

CHARCOAL BEDS

KR83M	2.98E+01	KR85M	7.55E+01	TOTAL = 1.34E+03	
KR87	1.20E+02	KR88	2.07E+02	RB88	2.09E+02
KR89	3.22E-00	RB89	4.43E-00	XE133M	5.47E-00
XE133	1.36E+02	XE135M	2.37E+01	XE135	3.48E+02
XE137	7.74E-00	XE138	8.24E+01	CS138	8.84E+01

AFTER-FILTER - GAS COMPONENT

KR83M	1.14E-02	KR85M	5.34E-02	TOTAL = 6.53E-01	
KR87	2.64E-02	KR88	1.14E-01	XE133M	5.54E-03
XE133	1.40E-01	XE135	3.02E-01		

AFTER-FILTER - ONE YEAR MEDIA COMPONENT

RB88	1.29E-03	SR89	8.90E-03	TOTAL = 1.11E-02
CS137	4.62E-04	BA139	4.00E-04	

NOTES:

(1) Component activity inventories are based on operation of the charcoal adsorbers at 0°F utilizing the corresponding dynamic adsorption coefficients of charcoal for retention of krypton and xenon. At temperatures greater than 0°F, the adsorption coefficients decrease, resulting in a reduction in radionuclide inventory in the charcoal adsorbers. Thus, the inventory activities utilized are based on maximum inventories, i.e., operation at 0°F.

(2) The design basis case source terms are based on 2 scfm flow, 100,000 μCi/sec mix for 11 months plus 350,000 μCi/sec mix for 30 days, total buildup time of 1 year. This case represents abnormal operating conditions of low flow and an activity release spike of 250,000 μCi/sec for 30 days.

(3) The normal case is based on 30 scfm flow, 100,000 μCi/sec mix, and 1 year buildup time. This case represents the normal design offgas system operating conditions.

TABLE 15.7-4

EQUIPMENT FAILURE RELEASE ASSUMPTIONS
RELEASE FRACTIONS ASSUMED FOR DESIGN BASIS/
REALISTIC ANALYSIS

Equipment Piece	Noble Gases	Solid Daughters	Radioiodine
Preheater	1.00/1.00	1.00/1.00	N/A
Catalytic Recombiner	1.00/1.00	1.00/1.00	N/A
Offgas Condenser	1.00/1.00	1.00/1.00	N/A
Water Separator	1.00/1.00	1.00/1.00	N/A
Holdup Pipe	1.00/0.20	1.00/0.20	N/A
Cooler Condenser	1.00/1.00	1.00/1.00	N/A
Moisture Separator	1.00/1.00	1.00/1.00	N/A
Desiccant Dryer	1.00/0.10	0.01/0.01	N/A
Prefilter	1.00/1.00	0.01/0.01	N/A
Charcoal Adsorbers	0.10/0.10	0.01/0.0	0.01/0.01
After-filter	1.00/1.00	0.01/0.01	N/A

TABLE 15.7-5

GASEOUS RADWASTE SYSTEM FAILURE
HOLDUP PIPE
(DESIGN BASIS ANALYSIS)
OFFSITE RADIOLOGICAL ANALYSIS (mrem)

Distance (m)	Whole Body	Thyroid
863	338	2.07E-2
4,002	41.2	2.53E-3

TABLE 15.7-6

GASEOUS RADWASTE SYSTEM FAILURE - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

		Design Basis Assumptions	Realistic Basis Assumptions
I.	Data and assumptions used to estimate radioactive source postulated accidents		
A.	Power level	N/A	N/A
B.	Burn-up	N/A	N/A
C.	Fuel damage	None	None
D.	Inventory of activity by nuclide	<Table 15.7-3a>	<Table 15.7-3a>
E.	Iodine fractions, %		
	(1) Organic	0	0
	(2) Elemental	100	100
	(3) Particulate	0	0
F.	Reactor coolant activity before the accident	N/A	N/A
II.	Data and assumptions used to estimate activity released		
A.	Containment leak rate (% day)	N/A	N/A
B.	Secondary containment leak rate (% day)	N/A	N/A
C.	Isolation valve closure time (sec)	N/A	N/A
D.	Adsorption and filtration efficiencies:	N/A	N/A
	(1) Organic iodine	N/A	N/A

TABLE 15.7-6 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
(2) Elemental iodine	N/A	N/A
(3) Particulate Iodine	N/A	N/A
(4) Particulate fission products	N/A	N/A
E. Recirculation system parameters	N/A	N/A
(1) Flow rate	N/A	N/A
(2) Mixing Efficiency	N/A	N/A
(3) Filter Efficiency	N/A	N/A
F. Containment spray parameters (flow rate, drop size, etc.)	N/A	N/A
G. Containment volumes	N/A	N/A
H. All other pertinent data and assumptions	None	None
III. Dispersion data		
A. Boundary and LPZ distances (m)	863/4002	863/4002
B. X/Q's for SB/LPZ (sec/m^3)	6.7E-4/8.2E-5	3.8E-5/3.8E-6
IV. Dose Data		
A. Method of dose calculation	N/A	
B. Dose conversion assumptions	(Reference 5)	(Reference 5)
C. Peak activity concentrations in containment	N/A	N/A
D. Doses	<Table 15.7-5>	<Table 15.7-7>

TABLE 15.7-7

GASEOUS RADWASTE SYSTEM FAILURE
HOLDUP PIPE
(REALISTIC ANALYSIS)
OFFSITE RADIOLOGICAL ANALYSIS (mrem)

Distance (m)	Whole Body	Thyroid
863	3.34	7.33E-4
4,002	0.334	7.33E-5

TABLE 15.7-8

FAILURE OF THE AIR EJECTOR LINE - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	None	N/A
B. Burn-up	None	N/A
C. Fuel damage	None	None
D. Release of activity by nuclide	None	<Table 15.7-9>
E. Iodine fractions, %		
(1) Organic	None	0
(2) Elemental	None	100
(3) Particulate	None	0
F. Reactor coolant activity before the accident	None	<Section 15.6.4.5.2.2>
II. Data and assumptions used to estimate activity released		
A. Containment leak rate (% day)	None	N/A
B. Secondary containment leak rate (% day)	None	N/A
C. Isolation valve closure time (sec)	None	N/A
D. Adsorption and filtration efficiencies:		
(1) Organic iodine	None	N/A

TABLE 15.7-8 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
(2) Elemental iodine	None	N/A
(3) Particulate Iodine	None	N/A
(4) Particulate fission products	None	N/A
E. Recirculation system parameters		
(1) Flow rate	None	N/A
(2) Mixing Efficiency	None	N/A
(3) Filter Efficiency	None	N/A
F. Containment spray parameters (flow rate, drop size, etc.)	None	N/A
G. Containment volumes	N/A	
H. All other pertinent data and assumptions	None	N/A
III. Dispersion data		
A. Boundary and LPZ distances (m)	None	863/4,002
B. X/Q's for total dose - SB/LPZ	None	6.7E-4/8.2E-5
IV. Dose Data		
A. Method of dose calculation	None	<Section 15.0.3.5>
B. Dose conversion assumptions	None	<Section 15.0.3.5>
C. Peak activity concentrations in containment	None	N/A
D. Doses	None	<Table 15.7-10>

TABLE 15.7-9

FAILURE OF AIR EJECTOR LINE
(REALISTIC ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT (CURIES)

Isotope	Activity
I-131	3.13E-5
I-132	4.07E-4
I-133	2.35E-4
I-134	7.51E-4
I-135	3.75E-4
Kr-83m	3.79E+0
Kr-85	2.59E-2
Kr-85m	6.64E+0
Kr-87	2.07E+1
Kr-88	2.12E+1
Kr-89	8.83E+1
Xe-131m	2.12E-2
Xe-133m	3.17E-1
Xe-133	8.87E+0
Xe-135m	2.59E+1
Xe-135	2.39E+1
Xe-137	1.17E+2
Xe-138	8.83E+1

TABLE 15.7-10

FAILURE OF AIR EJECTOR LINE
(REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area (863 Meters)	6.33E-2	1.11E-4
Low Population Zone (4,002 Meters)	7.75E-3	1.32E-5

TABLE 15.7-11

LIQUID WASTE SYSTEM FAILURE PARAMETERS
TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	None	N/A
B. Burn-up	None	N/A
C. Fuel damage	None	None
D. Inventory of activity by nuclide	None	<Table 15.7-12>
E. Iodine fractions		
(1) Organic	None	0
(2) Elemental	None	1
(3) Particulate	None	0
F. Reactor coolant activity before the accident based on	None	15 μCi/MWt-sec
II. Data and assumptions used to estimate activity released		
A. Primary containment leak rate (% day)	None	N/A
B. Secondary containment leak rate (% day)	None	N/A
C. Isolation valve closure time (sec)	None	N/A
D. Adsorption and filtration efficiencies - radwaste bldg.		
(1) Organic iodine	None	N/A

TABLE 15.7-11 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
(2) Elemental iodine	None	N/A
(3) Particulate iodine	None	N/A
(4) Particulate fission products	None	N/A
E. Recirculation system parameters		
(1) Flow rate	None	N/A
(2) Mixing efficiency	None	N/A
(3) Filter efficiency	None	N/A
F. Containment spray parameters (flow rate, drop size, etc.)	None	N/A
G. Containment volumes	None	N/A
H. All other pertinent data and assumptions	None	(Reference 8)
III. Dispersion data		
A. Boundary and LPZ distances (m)	None	N/A
B. X/Q's for total dose - SB/LPZ	None	N/A
IV. Dose Data		
A. Method of dose calculation	None	15.0.3.5
B. Dose conversion assumptions	None	(1)
C. Peak activity concentrations in containment	None	N/A

TABLE 15.7-11 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
D. Doses	None	<Table 15.7-14>

NOTE:

(1) The dose conversion factors for the other isotopes considered are taken from (Reference 7).

TABLE 15.7-12

INVENTORY OF ACTIVITIES FOR LIQUID RADWASTE SYSTEM COMPONENTS
(microcuries)

Isotope	Waste Collector Tank	Floor Drain Collector Tank	Waste Sample WST	Floor Drains Storage Tank	Chemical Waste Tank	Chemical Waste Distillate Tank	RWCU Backwash Settling Tank[1]	Condensate Filter Backwash Receiving Tank[1]	Condensate Filter Backwash Settling Tank[1][2]	Fuel Pool Filter/Demin Backwash Receiving Tank[1]	Spent Resin Tank[1]
F-18	6.02E+06	6.02E+06	5.57E+06	5.57E+06	3.30E+06	3.47E+04	1.50E+04	2.08E+03	1.90E+04	0.00E+00	6.64E+04
Na-24	3.01E+05	3.01E+05	2.79E+05	2.79E+05	1.65E+05	1.58E+02	6.14E+03	7.78E+01	7.78E+03	2.80E-01	2.72E+04
P-32	3.01E+03	3.01E+03	2.79E+03	2.79E+03	1.65E+03	1.58E+00	1.47E+03	1.78E+01	1.78E+03	3.00E+01	5.98E+03
Cr-51	7.53E+03	7.53E+03	6.97E+04	6.97E+04	4.13E+04	3.94E+01	7.54E+04	8.60E+02	8.62E+04	9.60E+02	2.54E+05
Mn-54	6.02E+03	6.02E+03	5.57E+03	5.57E+03	3.30E+03	3.15E+00	6.28E+04	3.32E+02	7.80E+04	9.60E+01	6.18E+04
Mn-56	7.53E+06	7.53E+06	6.97E+06	6.97E+06	4.13E+06	3.94E+03	2.64E+04	3.34E+02	3.34E+04	0.00E+00	1.17E+05
Co-58	7.53E+05	7.53E+05	6.97E+05	6.97E+05	4.13E+05	3.94E+02	1.94E+06	2.02E+04	2.20E+06	1.10E+04	4.52E+06
Co-60	7.53E+04	7.53E+04	6.97E+04	6.97E+04	4.13E+04	3.94E+01	2.26E+06	5.16E+03	5.34E+06	1.20E+03	9.32E+05
Fe-59	1.20E+04	1.20E+04	1.11E+04	1.11E+04	6.60E+03	6.30E+00	1.97E+04	2.18E+02	2.22E+04	1.72E+02	5.56E+04
Ni-65	4.52E+04	4.52E+04	4.18E+04	4.18E+04	2.48E+04	2.36E+01	1.55E+02	1.96E+00	1.96E+02	0.00E+00	6.86E+02
Zn-65	3.01E+02	3.01E+02	2.79E+02	2.79E+02	1.65E+02	1.58E+01	2.50E+03	1.55E+01	3.04E+03	4.80E+00	2.92E+03
Zn-69m	4.52E+03	4.52E+03	4.18E+03	4.18E+03	2.48E+03	2.36E+00	8.44E+01	1.07E+00	1.07E+02	1.94E+03	3.74E+02
I-131	1.81E+06	1.81E+06	1.67E+06	1.67E+06	9.90E+05	1.89E+04	4.78E+05	1.20E+05	6.00E+05	2.20E+04	2.10E+06
I-134	3.77E+07	3.77E+07	3.48E+07	3.48E+07	2.06E+07	3.94E+05	4.48E+04	1.14E+04	5.68E+04	0.00E+00	1.99E+05
Sr-89	4.07E+05	4.07E+05	3.76E+05	3.76E+05	2.23E+05	2.13E+02	7.52E+05	8.22E+03	8.50E+05	5.60E+03	2.02E+06
Cs-134	2.11E+04	2.11E+04	1.95E+04	1.95E+04	1.16E+04	1.10E+01	4.26E+05	1.35E+03	6.54E+05	3.40E+02	2.46E+05
Cs-136	1.39E+04	1.39E+04	1.28E+04	1.28E+04	7.59E+03	7.25E+00	6.12E+03	7.54E+01	7.54E+03	1.32E+02	2.56E+04
W-187	4.52E+05	4.52E+05	4.18E+05	4.18E+05	2.48E+05	2.36E+02	1.46E+04	1.86E+02	1.86E+04	1.32E+01	6.50E+04
Sr-90	3.01E+04	3.01E+04	2.79E+04	2.79E+04	1.65E+04	1.58E+01	1.16E+06	2.14E+03	4.34E+06	4.80E+02	3.86E+05
Y-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.15E+06	2.10E+03	4.34E+06	4.80E+02	3.74E+05
Sr-92	1.66E+07	1.66E+07	1.53E+07	1.53E+07	9.08E+06	8.67E+03	6.10E+04	7.74E+02	7.74E+04	0.00E+00	2.70E+05
Y-92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.10E+04	7.74E+02	7.74E+04	0.00E+00	2.70E+05
Mo-99	3.01E+06	3.01E+06	2.79E+06	2.79E+06	1.65E+06	1.58E+03	2.70E+05	3.42E+03	3.42E+05	4.40E+03	1.20E+06
Tc-99m	1.14E+07	1.14E+07	1.06E+07	1.06E+07	6.27E+06	5.99E+03	3.28E+05	4.22E+03	4.22E+05	3.00E-05	1.48E+06
Ru-103	2.56E+03	2.56E+03	2.37E+03	2.37E+03	1.40E+03	1.34E+00	3.68E+03	4.12E+01	4.16E+03	3.40E+01	1.09E+04
Rh-103m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.68E+03	4.12E+01	4.16E+03	3.40E+01	1.09E+04
Ag-110m	9.04E+03	9.04E+03	8.36E+03	8.36E+03	4.95E+03	4.73E+00	7.70E+04	4.68E+02	9.34E+04	1.46E+02	8.80E+04
Te-132	1.81E+06	1.81E+06	1.67E+06	1.67E+06	9.90E+05	9.46E+02	1.92E+05	2.44E+03	2.44E+05	4.00E+03	8.52E+05
I-132	1.81E+07	1.81E+07	1.67E+07	1.67E+07	9.90E+06	1.89E+05	2.48E+05	1.68E+04	3.16E+05	4.20E+03	1.10E+06
I-135	1.81E+07	1.81E+07	1.67E+07	1.67E+07	9.90E+06	1.89E+05	1.62E+05	4.12E+04	2.06E+05	6.80E-04	7.20E+05
Cs-137	3.16E+04	3.16E+04	2.93E+04	2.93E+04	1.73E+04	1.66E+01	1.22E+06	2.24E+03	4.60E+06	5.20E+02	4.06E+05
Ba-140	1.16E+06	1.16E+06	1.07E+06	1.07E+06	6.35E+05	6.07E+02	5.02E+05	6.14E+03	6.14E+05	1.10E+04	2.08E+06
La-140	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.06E+05	6.14E+03	6.14E+05	1.26E+04	2.08E+06
Ce-143	4.67E+03	4.67E+03	4.32E+03	4.32E+03	2.56E+03	2.44E+00	2.10E+02	2.66E+00	2.66E+02	7.20E-01	9.30E+02
Pr-143	4.97E+03	4.97E+03	4.60E+03	4.60E+03	2.72E+03	2.60E+00	2.52E+03	3.06E+01	3.06E+03	4.80E+01	1.03E+04

TABLE 15.7-12 (Continued)

Isotope	Waste Collector Tank	Floor Drain Collector Tank	Waste Sample WST	Floor Drains Storage Tank	Chemical Waste Tank	Chemical Waste Distillate Tank	RWCU Backwash Setting Tank[(1)]	Condensate Filter Backwash Receiving Tank[(1)]	Condensate Filter Backwash Setting Tank[(1)(2)]	Fuel Pool Filter/Demin Backwash Receiving Tank[(1)]	Spent Resin Tank[(1)]
Ce-144	4.52E+03	4.52E+03	4.18E+03	4.18E+03	2.48E+03	2.36E+00	4.32E+04	2.42E+02	5.32E+04	7.20E+01	4.54E+04
Nd-147	1.81E+03	1.81E+03	1.67E+03	1.67E+03	9.90E+02	9.46E-01	6.64E+02	8.20E+00	8.20E+02	1.62E+01	2.82E+03
Pr-147	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.67E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Np-239	3.16E+07	3.16E+07	2.93E+07	2.93E+07	1.73E+07	1.66E+04	2.42E+06	3.08E+04	3.08E+06	3.20E+04	1.08E+07
Pu-239	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.42E+02	6.00E-01	1.49E+03	0.00E+00	1.06E+02
Sr-91	9.94E+06	9.94E+06	9.19E+06	9.19E+06	5.45E+06	5.20E+03	1.28E+05	1.63E+03	1.63E+05	7.60E-02	5.70E+05
Y-91m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.30E+04	1.54E+03	0.00E+00	7.60E-02	0.00E+00
Y-91	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.44E+05	0.00E+00	1.63E+05	0.00E+00	3.62E+05
Zr-95	5.27E+03	5.27E+03	4.88E+03	4.88E+03	2.89E+03	2.76E+00	1.23E+04	1.30E+02	1.40E+04	7.60E+01	3.00E+04
Nb-95	5.42E+03	5.42E+03	5.01E+03	5.01E+03	2.97E+03	2.84E+00	1.92E+04	1.98E+02	2.18E+04	2.40E+02	4.16E+04
Zr-97	4.37E+03	4.37E+03	4.04E+03	4.04E+03	2.39E+03	2.29E+00	1.00E+02	1.27E+00	1.27E+02	1.12E-02	4.46E+02
Nb-97	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E+02	1.27E+00	1.27E+02	0.00E+00	4.46E+02
Te-129m	4.52E+04	4.52E+04	4.18E+04	4.18E+04	2.48E+04	2.36E+01	5.52E+04	6.24E+02	6.28E+04	5.40E+02	1.74E+05
Te-129	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.48E+04	3.92E+02	3.94E+04	0.00E+00	1.09E+05
I-133	1.22E+07	1.22E+07	1.13E+07	1.13E+07	6.68E+06	1.28E+05	3.46E+05	8.74E+04	4.38E+05	2.60E+02	1.53E+06
Ce-141	5.12E+03	5.12E+03	4.74E+03	4.74E+03	2.81E+03	2.68E+00	2.00E+04	2.26E+02	2.26E+04	6.80E+01	6.36E+04
Ru-106	3.46E+02	3.46E+02	3.20E+02	3.20E+02	1.90E+02	1.81E-01	4.16E+03	1.98E+01	5.28E+03	5.40E+00	3.66E+03
Br-83	2.11E+06	2.11E+06	1.95E+06	1.95E+06	1.16E+06	2.21E+04	6.84E+03	1.74E+03	8.68E+03	0.00E+00	3.04E+04
Br-84	4.37E+06	4.37E+06	4.04E+06	4.04E+06	2.39E+06	4.57E+04	3.14E+03	7.98E+02	3.98E+03	0.00E+00	1.40E+04
Br-85	2.71E+06	2.71E+06	2.51E+06	2.51E+06	1.49E+06	2.84E+04	1.76E+02	4.46E+01	2.24E+02	0.00E+00	7.82E+02
Tc-101	2.41E+07	2.41E+07	2.23E+07	2.23E+07	1.32E+07	1.26E+04	7.80E-03	9.82E+01	9.82E+03	0.00E+00	3.44E+04
Cs-138	3.16E+07	3.16E+07	2.93E+07	2.93E+07	1.73E+07	1.66E+04	2.30E+04	2.92E+02	2.92E+04	0.00E+00	1.02E+05
Ba-139	2.56E+07	2.56E+07	2.37E+07	2.37E+07	1.40E+07	1.34E+04	4.80E+04	6.12E+02	6.12E+04	0.00E+00	2.14E+05
Rh-106	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.16E+03	1.98E+01	5.28E+03	5.52E+00	3.66E+03
Ba-137m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.15E+06	2.12E+03	4.36E+06	5.20E+02	3.84E+05
Ba-141	3.01E+07	3.01E+07	2.79E+07	2.79E+07	1.65E+07	1.58E+04	1.25E+04	1.58E+02	1.58E+04	0.00E+00	5.54E+04
Ba-142	2.86E+07	2.86E+07	2.65E+07	2.65E+07	1.57E+07	1.50E+04	6.90E+03	8.80E+01	8.80E+03	0.00E+00	3.08E+04
La-142	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.90E+03	8.80E+01	8.80E+03	0.00E+00	3.08E+04
Pr-144	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.52E+04	2.42E+02	5.32E+04	0.00E+00	4.54E+04
Pm-147	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.76E+02	1.28E+00	8.10E+02	0.00E+00	2.18E+02

NOTES:

(1) This activity shown for these tanks is for the liquid portion of the tank contents only. Activity bound up in solids residing in these tanks will not travel through the porous concrete mat into the underdrain system/volume and therefore is not considered part of the Liquid Radwaste rupture source term.

(2) These values are also applicable to the Fuel Pool F/D Backwash Settling Tank as the tanks can be utilized interchangeably.

<TABLE 15.7-13>

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TABLE 15.7-14

ANNUAL INTEGRATED
RADIOLOGICAL DOSES FROM INGESTION OF
WATER FOR FAILURES IN THE QUALITY GROUP D
PORTION OF THE LIQUID RADWASTE SYSTEM[1]

	Component	Whole Body Exposure (mrem)	Thyroid Exposure (mrem)
1.	Condensate Filter Backwash Settling Tank[2]	1.4E+0 7.5E+0	5.5E-1 2.9E+0
2.	Condensate Filter Backwash Receiving Tank	5.6E-3	1.5E-2
3.	Chemical Waste Distillate Tank	1.1E-4	2.0E-3
4.	Concentrated Waste Tank[3]		Deleted
5.	RWCU Backwash Settling Tank	2.4E+0	1.1E+0
6.	Fuel Pool F/D Backwash Receiving Tank	1.4E-3	2.9E-3
7.	Waste Collector Tank	9.0E-2	2.3E-1
8.	Waste Sample Tank	8.4E-2	2.1E-1
9.	Floor Drains Collector Tank	9.0E-2	2.3E-1
10.	Floor Drain Sample Tank	8.4E-2	2.1E-1
11.	Spent Resin Tank	9.7E-1	6.4E-1
12.	Chemical Waste Tank	5.0E-2	1.3E-1

NOTES:

(1) Doses are based on the isotopic concentrations calculated to be present at the nearest drinking water intake.

(2) Radiological doses from rupture of condensate filter backwash settling tank are calculated assuming two different sources of activity. The first set of values is based on sludge activities from the condensate filters via the condensate filter backwash receiving tank. The second set of values is based on sludge activities from the Waste Collector/Floor Drain filters via the dewatering tank. These are also applicable to the fuel pool filter/demin backwash settling tank.

(3) This tank is no longer used as originally designed. Further evaluation would be needed prior to use.

TABLE 15.7-15a

RADIONUCLIDE CONCENTRATIONS AT NEAREST DRINKING WATER INTAKE FOR FAILURE OF CONDENSATE FILTER BACKWASH SETTLING TANK[(1)]

Isotope	Instantaneous Radionuclide Concentration (micro Ci/cc)	<10 CFR 20, Appendix B> Effluent Concentration	Fraction of Effluent Water Concentrations
F-18	9.78E-35	7.00E-04	1.40E-31
Na-24	2.93E-13	5.00E-05	5.86E-09
P-32	5.06E-11	9.00E-06	5.63E-06
Cr-51	2.84E-09	5.00E-04	5.68E-06
Mn-54	2.96E-09	3.00E-05	9.87E-05
MN-56	5.57E-27	7.00E-05	7.95E-23
Co-58	7.97E-08	2.00E-05	3.98E-03
Co-60	2.05E-07	3.00E-06	6.84E-02
Fe-59	7.76E-10	1.00E-05	7.76E-05
Ni-65	6.61E-30	1.00E-04	6.61E-26
Zn-65	1.15E-10	5.00E-06	2.30E-05
Zn-69m	2.09E-15	6.00E-05	3.49E-11
I-131	7.60E-09	1.00E-06	7.60E-03
I-134	3.93E-61	4.00E-04	9.83E-58
Sr-89	1.69E-08	8.00E-06	2.12E-03
Cs-134	1.41E-08	9.00E-07	1.57E-02
Cs-136	1.18E-10	6.00E-06	1.96E-05
W-187	9.06E-12	3.00E-05	3.02E-07
Sr-90	9.42E-08	5.00E-07	1.88E-01
Y-90	1.86E-08	7.00E-06	2.66E-03
Sr-92	3.20E-26	4.00E-05	8.00E-22
Y-92	2.12E-22	4.00E-05	5.31E-18
Mo-99	1.54E-09	2.00E-05	7.69E-05
Tc-99m	2.90E-16	1.00E-03	2.90E-13
Ru-103	8.09E-11	3.00E-05	2.70E-06
Rh-103m	4.70E-59	6.00E-03	7.84E-57
Ag-110m	3.53E-09	6.00E-06	5.89E-04
Te-132	1.40E-09	9.00E-06	1.56E-04
I-132	1.61E-28	1.00E-04	1.61E-24
I-135	6.62E-16	3.00E-05	2.21E-11
Cs-137	9.99E-08	1.00E-06	9.99E-02
Ba-140	9.51E-09	8.00E-06	1.19E-03
La-140	1.00E-09	9.00E-06	1.12E-04
Ce-143	2.48E-13	2.00E-05	1.24E-08
Pr-143	4.83E-11	2.00E-05	2.41E-06
Ce-144	1.14E-09	3.00E-06	3.79E-04
Nd-147	1.20E-11	2.00E-05	6.00E-07
Pr-147	0.00E+00	1.00E-03	0.00E+00
Np-239	1.06E-08	2.00E-05	5.30E-04
Pu-239	3.23E-11	2.00E-08	1.61E-03
Sr-91	6.26E-14	2.00E-05	3.13E-09

TABLE 15.7-15a

RADIONUCLIDE CONCENTRATIONS AT NEAREST DRINKING WATER INTAKE FOR FAILURE OF CONDENSATE FILTER BACKWASH SETTLING TANK[1]

Isotope	Instantaneous Radionuclide Concentration (micro Ci/cc)	<10 CFR 20, Appendix B> Effluent Concentration	Fraction of Effluent Water Concentrations
Y-91m	0.00E+00	2.00E-03	0.00E+00
Y-91	3.28E-09	8.00E-06	4.10E-04
Zr-95	2.83E-10	2.00E-05	1.42E-05
Nb-95	4.18E-10	3.00E-05	1.39E-05
Zr-97	5.89E-15	9.00E-06	6.54E-10
Nb-97	7.46E-50	3.00E-04	2.49E-46
Te-129m	1.20E-09	7.00E-06	1.71E-04
Te-129	1.03E-48	4.00E-04	2.58E-45
I-133	6.42E-11	7.00E-06	9.18E-06
Ce-141	4.30E-10	3.00E-05	1.43E-05
Ru-106	1.13E-10	1.00E-08	1.13E-02
Br-83	2.43E-29	9.00E-04	2.70E-26
Br-84	5.72E-96	4.00E-04	1.43E-92
Br-85	0.00E+00	2.00E-09	0.00E+00
Tc-101	3.75E-201	2.00E-03	1.88E-198
Cs-138	4.79E-94	4.00E-04	1.20E-90
Ba-139	3.37E-42	2.00E-04	1.69E-38
Rh-106	0.00E+00	2.00E-09	0.00E+00
Ba-137m	0.00E+00	2.00E-09	0.00E+00
Ba-141	1.89E-158	3.00E-04	6.28E-155
Ba-142	1.35E-263	7.00E-04	1.93E-260
La-142	7.73E-39	1.00E-04	7.73E-35
Pr-144	2.03E-166	6.00E-04	3.39E-163
Pm-147	1.75E-11	7.00E-05	2.50E-07
Total			4.06E-01

NOTE:

[1] These values are also applicable to the Fuel Pool F/D Backwash settling tank.

<TABLE 15.7-15b>

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TABLE 15.7-16

SUMMARY OF TOTAL CONCENTRATION AND TOTAL FEWC AT NEAREST DRINKING WATER INTAKE FOR FAILURES OF QUALITY GROUP D EQUIPMENT

	Component	Instantaneous Radionuclide Concentration (μCi/cc)	Fraction of <10 CFR 20, Appendix B> Effluent Water Concentration
1.	Waste Collector Tank	2.2E-07	4.0E-02
2.	Waste Sample Tank	2.1E-07	3.7E-02
3.	Floor Drains Collector Tank	2.2E-07	4.0E-02
4.	Floor Drains Sample Tank	2.1E-07	3.7E-02
5.	Chemical Waste Distillate Tank	4.1E-09	2.6E-04
6.	Condensate Filter Backwash Receiving Tank	3.1E-09	1.9E-03
7.	Condensate Filter Backwash Settling Tank[1]	8.9E-08 5.8E-07	8.4E-02 4.1E-01
8.	Chemical Waste Tank	1.9E-06	2.7E-02
9.	Concentrated Waste Tank[2]		Deleted
10.	RWCU Backwash Settling Tank	2.8E-07	1.4E-01
11.	Fuel Pool F/D Backwash Receiving Tank	1.3E-09	4.2E-04
12.	Spent Resin Tank	4.1E-07	1.1E-01

NOTES:

[1] Total concentration and Fraction of Effluent Water Concentration for the Condensate Filter Backwash Settling Tank are calculated assuming two different sources of activity. The first set of values is based on sludge activities from the condensate filters via the condensate backwash receiving tank. The second set of values is based on sludge activities from the waste collector/floor drain filters via the dewatering tank. These values also apply to the Fuel Pool F/D Backwash Settling Tank.

[2] This tank has been abandoned and cannot be used without an evaluation to ensure limits are not exceeded.

<TABLE 15.7-17>
<TABLE 15.7-18>
<TABLE 15.7-19>
<TABLE 15.7-20>
<TABLE 15.7-21>
<TABLE 15.7-22>
<TABLE 15.7-23>
<TABLE 15.7-24>
<TABLE 15.7-25>
<TABLE 15.7-26>
<TABLE 15.7-27>
<TABLE 15.7-28>
<TABLE 15.7-29>
<TABLE 15.7-30>

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TABLE 15.7-31

SEQUENCE OF EVENTS FOR FAILURE OF MAIN STEAM AIR EJECTOR LINES

Time, sec	Events
0	Air ejector lines fail
0+	Air begins to be accumulated in condenser. Condenser pressure rises
3	High condenser pressure trips turbine (plant scrams)
3+	Bypass opens, further increasing condenser pressure
8	High condenser pressure trips MSIV's

TABLE 15.7-32

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS
(ASSUMING 24 HOUR RADIOACTIVE DECAY)

	Design Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents	
A. Power level	3,833 MWt
B. Burn-up	- Peak rod exposure is less than 62,000 MWD/MT - Between 54,000 and 62,000 MWD/MT, the maximum linear heat generation rate does not exceed 6.3 KW/Ft peak rod average exposure.
C. Fuel damage (GNF2)	150 rods
D. Release of activity to containment pool by nuclide, per failed rod	<Section 15.7.6.4.1>
E. Iodine fractions - organic - elemental	.0015 .9985
F. Radial peaking factor	2.0
II. Data and assumptions used to estimate activity released	
A. Primary containment leak rate	Instantaneous total release of all activity leaving pool to environment
B. Secondary containment leak rate	N/A
C. Isolation valve closure times	N/A
D. Filtration efficiencies	N/A

TABLE 15.7-32 (Continued)

		Design Basis Assumptions
E.	Recirculation systems parameters (flow rates vs. time, mixing factor, etc.)	N/A
F.	Containment spray parameters parameters (flow rate, drop size, etc.)	N/A
G.	Containment volumes	N/A
H.	Pool removal (overall effective decontamination factor)	200
I.	All other pertinent data and assumptions	<Section 15.7.6>
J.	Activity released to environment	<Table 15.7-34>
K.	Control room parameters	See below
	1. Volume (ft^3)	371,760 ft^3

2.

Design Flow (cfm)	Intake	Exhaust (Air Purge)	Emergency Recirculation
Case 1	6600 (Normal +10%)	5400 (Normal -10%)	0
Case 2	6600 Isolated after activity introduced into control room	5400 Started after 2 hrs	0
Case 3	6600 Isolated after activity introduced into control room	0	27,000(Normal -10%) Started after 2 hrs Filter Efficiency = 80%

TABLE 15.7-32 (Continued)

	Design Basis Assumptions
III. Dispersion Data	
A. Boundary and LPZ distances (m)	863/4002
B. Offsite X/Q's (Corresponding to 7 year meteorological data for 0-2 hr for SB/0-8 hr for LPZ)	4.3E-4/4.8E-5 s/m^3
C. Control Room X/Q's (0-8 hr)	3.5E-4 s/m^3
IV. Dose Data	
A. Method of dose calculation	<Section 15.0.3.5>
B. Dose conversion assumptions	
1. Dose conversion assumptions (Offsite)	<Table 15.0-4>
2. Dose Conversion Assumptions (Control Room)	<Table 15.0-4>
C. Peak activity concentrations in containment	N/A
D. Doses	<Table 15.7-35>

<TABLE 15.7-33>

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TABLE 15.7-34

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
ASSUMING 24 HOUR RADIOLOGICAL DECAY OF THE FUEL
DESIGN BASIS ANALYSIS
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)[1]

Isotope	Design Basis Source Terms Activity
I-129	5.64E-6
I-130	1.21E+0
I-131	1.81E+2
I-132	1.44E+2
I-133	1.14E+2
I-134	6.00E-6
I-135	1.87E+1
Kr-79	1.36E-8
Kr-81	1.62E-8
Kr-83m	1.15E+1
Kr-85	6.81E+2
Kr-85m	1.49E+2
Kr-87	2.43E-2
Kr-88	4.63E+1
Xe-127	1.62E-3
Xe-129m	1.66E-1
Xe-131m	2.73E+2
Xe-133m	1.40E+3
Xe-133	4.60E+4
Xe-135m	5.99E+2
Xe-135	1.27E+4
Br-80	1.72E-6
Br-80m	1.60E-6
Rb-82	4.96E-1
Rb-83	1.48E-2
Rb-87	0[2]
Rb-88	0[2]
Cs-135	0[2]

NOTES:

[1] GNF2 Fuel @ 3,833 MWt
[2] Retained completely in the pool

TABLE 15.7-35

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
ASSUMING 24 HOUR RADIOLOGICAL DECAY OF THE FUEL
(DESIGN BASIS ANALYSIS)

RADIOLOGICAL EFFECTS

	TEDE Dose (rem)	Licensing Limit (rem)
Exclusion Area (863 Meters)	1.43	6.3
Low Population Zone (4002 Meters)	0.16	6.3
Control Room:		
Case 1	1.02	5
Case 2	2.77	5
Case 3	2.78	5

15.8 OTHER EVENTS

15.8.1 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

This transient was performed as part of analyses supporting PNPP operation in various operating modes and/or with equipment out of service results of which are presented in the following Chapter 15 appendices:

- <Appendix 15D> Partial Feedwater Heating Operation Analysis
- <Appendix 15E> Maximum Extended Operating Domain Analysis
- <Appendix 15F> Recirculation System Single-Loop Operation Analysis

The Perry design utilizes diverse, highly redundant and very reliable equipment to effect a scram. This equipment is frequently tested and would insert the control rods even if multiple component failures should occur, thus making the possibility of an anticipated transient without scram (ATWS) event extremely remote.

The NRC has established requirements to further reduce the risk to the public resulting from such an event. These requirements are specified in <10 CFR 50.62>, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants." For the BWR, <10 CFR 50.62> requires an alternate rod insertion (ARI) system, a manual standby liquid control scram (SLCS) and an automatic recirculation pump trip (RPT). The Perry design satisfies the ATWS design requirements of <10 CFR 50.62>.

The Perry design meets the general design criteria of Title 10 of the Code of Federal Regulations <Section 3.1>. ATWS requirements are "normal scram" failure additional to the "single failure criteria" incorporated in the general design criteria. This philosophy of more than one “single failure criteria” is clearly beyond the design basis events discussed in <Chapter 15>. Consequently, the discussion of ATWS is discussed in <Appendix 15C>.

15.8.2 STATION BLACKOUT (SBO)

The Station Blackout (SBO) event is defined as a complete loss of AC power to the essential and nonessential switchgear buses. This involves a loss of offsite power concurrent with a turbine trip and failure of the onsite emergency AC power system. The SBO event does not include the loss of AC power to buses fed from station batteries through inverters or the loss of AC power from alternate AC power sources. It also does not assume any other concurrent single failure or design basis accident. The loss of offsite power followed by the loss of more than one standby electrical division involves assumptions which are beyond the “single failure criteria” incorporated in the general design criteria of <10 CFR 50, Appendix A> <Section 3.1>. This philosophy of more than one “single failure” is clearly a beyond design basis event. However, the NRC has established requirements to further reduce the risk to the public resulting from an SBO event. These requirements are specified in <10 CFR 50.63>, “Loss of All Alternating Current Power.” Additional guidance is contained in <Regulatory Guide 1.155>, Revision 0 Dated August 1988, “Station Blackout.”

The SBO rulemaking specifies preventive and mitigative measures for this event, i.e., maintaining a highly reliable AC power system and having procedures in place and operators trained to these procedures to enable the plant to restore from the loss of offsite and onsite AC power. It also requires analysis to show that the plant is able to “cope” with a SBO for a specified time period. The SBO “coping” time is a function of the utility’s grid reliability, weather conditions associated with the location of the plant and actual station design. PNPP is a “4-hour” coping plant.

Based upon NRC evaluation of PNPP, the NRC stated that the staff agrees with the position that PNPP is a "4-hour" coping station. Furthermore, the NRC stated that PNPP conforms to the SBO rule, <10 CFR 50.63>.

A discussion of the SBO event is contained in <Appendix 15H>.

<APPENDIX 15A>

PLANT NUCLEAR SAFETY OPERATIONAL ANALYSIS (NSOA)

APPENDIX 15A

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

APPENDIX 15A

LIST OF TABLES

APPENDIX 15A

PLANT NUCLEAR SAFETY OPERATIONAL ANALYSIS (NSOA)

(A System-Level/Qualitative Type Plant FMEA)

15A.1 OBJECTIVES

The objectives of the Nuclear Safety Operational Analysis (NSOA) are cited below.

The NSOA was developed and included in the PNPP Unit 1 Safety Analysis Report in accordance with <Regulatory Guide 1.70> Revision 3, 11/78. The NSOA represented a systematic methodology which was developed to identify a minimum bounding set of design basis events, and the safety structures that mitigate each of these events. Safety-related functions, structures and systems are a subset of the safety functions, structures and systems identified in the NSOA. The NSOA captured generic developments including the design, testing and operating experience of the early BWR product lines. The repair time rule discussion in <Appendix 15A.5.3> was used as an assumption in early reliability analyses of BWR systems, and has been superceded by subsequent licensing basis documents which have undergone revision to reflect individual plant operating practices (e.g., the PNPP Technical Specifications and implementation of the Maintenance Rule). The NSOA identifies on a generic system level basis, those systems which were originally identified for inclusion in the technical specifications, and the safety systems utilized during the different modes of plant operation.

15A.1.1 ESSENTIAL PROTECTIVE SEQUENCES

Identify and demonstrate that essential protection sequences needed to accommodate the plant normal operations, anticipated and abnormal operation transients and design basis accidents are available and adequate. In addition, each event considered in the plant safety analysis <Chapter 15> is further examined and analyzed. Specific essential protective sequences are identified. The appropriate sequence is discussed for all BWR operating modes.

15A.1.2 DESIGN BASIS ADEQUACY

Identify and demonstrate that the safety design basis of the various structures, systems or components, needed to satisfy the plant essential protection sequences are appropriate, available and adequate. Each protective sequence identifies the specific structures, systems or components performing safety or power generation functions. The interrelationships between primary systems and secondary (or auxiliary equipment) in providing these functions are shown. The individual design bases (identified throughout the FSAR for each structure, system or component) are brought together by the analysis in this section. In addition to the individual equipment design basis analysis, the plant-wide design bases are examined and presented here.

15A.1.3 SYSTEM-LEVEL/QUALITATIVE TYPE FMEA

Identify 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. Each protective sequence entry is evaluated relative to SACF or SOE criteria. Safety classification aspects and interrelationships between systems are also considered.

15A.1.4 NSOA CRITERIA RELATIVE TO PLANT SAFETY ANALYSIS

Identify the systems, equipment or components' operational conditions and requirements essential to satisfy the nuclear safety operational criteria utilized in the <Chapter 15> plant events.

15A.1.5 TECHNICAL SPECIFICATION OPERATIONAL BASIS

Establish limiting operating conditions, testing and surveillance bases relative to plant technical specifications.

15A.2 APPROACH TO OPERATIONAL NUCLEAR SAFETY

15A.2.1 GENERAL PHILOSOPHY

The specified measures of safety used in this analysis are referred to as "unacceptable consequences." 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. Refer to <Figure 15A.2-1> for a description of the systematic process by which these unacceptable results are converted into safety requirements.

15A.2.2 SPECIFIC PHILOSOPHY

The following guidelines are utilized to develop the NSOA.

a. Scope and Classification Of Plant Events

1. Normal (Planned) Operations

   Normal operations which are under planned conditions in the absence of significant abnormalities. Operations subsequent to an incident (transient, accident or special event) are not considered planned operations until the procedures being followed or equipment being used are identical to those used during any one of the defined planned operations. Specific events are described further in <Table 15A.6-1>.

2. Anticipated (Expected) Operational Transients

   Anticipated Operational Transients are deviations from normal conditions which are expected to occur at a moderate frequency, and as such the design should include capability to withstand the conditions without operational impairment.

Included are incidents that result from a single operator error, control malfunction and others as described in <Table 15A.6-2>.

3. Abnormal (Unexpected) Operational Transients

Abnormal Operational Transients are deviations from normal conditions which occur infrequently. The design should include a capability to withstand these conditions without operational impairment. Refer to <Table 15A.6-3> for description of events included within this classification.

4. Design Basis (Postulated) Accidents

Design Basis Accident (DBA) is a hypothesized accident, the characteristics and consequences of which are utilized in the design of those systems and components pertinent to the preservation of radioactive material barriers and the restriction of radioactive material release from the barriers. The potential radiation exposures resulting from a design basis accident are greater than for any similar accident postulated from the same general accident assumptions. Specific events are described in <Table 15A.6-4>.

5. Special (Hypothetical) Events

Special Events are postulated to demonstrate some special capability of the plant in accordance with NRC requirements. For analyzed events within this classification see <Table 15A.6-5>.

b. Safety and Power Generation Aspects

Matters identified with “safety” classification are governed by regulatory requirements. Safety functions include:

1. The accommodation of abnormal operational transients and postulated design basis accidents.

2. The maintenance of containment integrity.

3. The assurance of ECCS.

4. The continuance of reactor coolant pressure boundary (RCPB) integrity.

Safety classified aspects are related to <10 CFR 100> or <10 CFR 50.67> (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <10 CFR 50.67>) 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 nonmechanistic) plant and environmental situations.

Power generation classified considerations are related to continued plant power generation operation, equipment operational matters, component availability aspects, and to long term offsite 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, 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, normal 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 (e.g., such as further failures or operator errors) certainly influence the classification grouping. The events in this appendix are initially grouped per initiating frequency occurrence. The imposition of further failures necessitates further classification to a lower frequency category.

The introduction of SACF or SOE into the examination of planned operation, anticipated operational transients or abnormal operational transient evaluations has not been previously considered a design basis or evaluation prerequisite. It is provided and included here to demonstrate the plant’s capability to accommodate the new requirement.

d. Conservative Analysis - Margins

The unacceptable consequences established in this appendix relative to the public health and safety aspects are in themselves in strict and conservative conformance to regulatory requirements.

Restrictive Operations on hypothetical limits established by further operational limits (e.g., setpoint margins) leads to disrespect for true safety aspects.

e. Safety Function Definition

First, the essential protective sequences shown for an event in this appendix list the minimum structures and systems required to be available to satisfy the SACF or SOE evaluation aspects of the event. Other protective "success paths" exist in some cases than are shown with the event.

Second, not all the events involve the same natural, environmental or plant conditional assumptions. For example LOCA and SSE are associated with Event 39. In Event 35, Control Rod Drop Accident CRDA is not assumed to be associated with any SSE or OBE occurrence. Therefore, seismic safety function requirements are not considered for Event 35. Some of the safety function equipment associated with the Event 35 protective sequence are also capable of handling more limiting events, such as Event 39.

Third, containment may be a safety function for some event (when uncontained radiological release would be unacceptable) but for other events it may not be applicable (e.g., during refueling). The requirement to maintain the containment in postaccident recovery is only needed to limit doses to less than <10 CFR 100> or <10 CFR 50.67> (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <10 CFR 50.67>).

After radiological sources are depleted with time, further containment is unnecessary. Thus, the “time domain” and “need for” aspects of a function should be and are taken into account and considered when evaluating the events in this appendix.

Fourth, the operation of Engineered Safety Feature (ESF) equipment, for normal operational events should not be misunderstood to mean that ESF equipment requirements apply to this event category.

Likewise the interpretation of the use of ESF-SACF capable systems for anticipated operational transient protective sequences should not imply that these equipment requirements (seismic, redundancy, diversity, testable, IEEE, etc.) are appropriately required for anticipated operational transients.

f. Envelope and Actual Event Analysis

The event analyses presented in <Chapter 15> do not include event frequency considerations. It does present an “envelope analysis” evaluation based on expected situations. Study of the actual plant occurrences, their frequency, their actual impact are reflected in their categorization in this appendix. This places the plant safety evaluations and impressions into a better perspectus by focusing attention on the “envelope analysis” with more appropriate understanding.

15A.2.2.1 Consistency of the Analysis

<Figure 15.A.2-2> 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 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.2-2> are avoided in the NSOA by directing the analysis to "event-consequences" oriented aspects. Analytical inconsistencies are avoided by treating all the events of a category under the same set of functional rules. Applying another set of functional rules to another category and by having a consistent set of rules between categories. Thus, it is valid to compare the results of the analyses of the events in any one category and invalid to compare events of different category, and thus different rules, to the other category. An example of this is the different rules (limits, assumptions, etc.) of accidents compared to anticipated transients.

15A.2.3 COMPREHENSIVENESS OF THE ANALYSIS

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.2.4 SYSTEMATIC APPROACH TO THE ANALYSIS

In summary, the systematic method utilized in this analysis contributes to both the consistency and comprehensiveness of the analysis mentioned above. The desired characteristics representative of a systematic approach to selecting BWR operational requirements are listed as follows:

a. Specify measures of safety-unacceptable consequences.

b. Consider all normal operations.

c. Systematic event selection.

d. Common treatment analysis of all events of any one type.

e. Systematic identification of plant actions and systems essential to avoiding unacceptable consequences.

f. Emergence of operational requirements and limits from system analysis.

<Figure 15A.2-1> illustrates the systematic process by which the operational and design basis nuclear safety requirements and technical specifications are derived. The process involves the evaluation of carefully selected plant events relative to the unacceptable

consequences (specified measures of safety). Those limits, actions, systems, and component level found to be essential to achieving acceptable consequences are the subjects of operational requirements.

15A.2.5 RELATIONSHIP OF NUCLEAR SAFETY OPERATIONAL ANALYSIS TO SAFETY ANALYSES OF CHAPTER 15

One of the main objectives of the operational analysis 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 but in light of frequency of occurrence, unacceptable consequences, assumption categories, and etc.

The detailed discussion relative to each of the events covered in <Chapter 15> will not be repeated in this appendix. Refer to the specific section in <Chapter 15> as cross-correlated in <Table 15A.6-1>, <Table 15A.6-2>, <Table 15A.6-3>, <Table 15A.6-4>, and <Table 15A.6-5>.

<Table 15A.6-1>, <Table 15A.6-2>, <Table 15A.6-3>, <Table 15A.6-4>, and <Table 15A.6-5> provides cross-correlation between the NSOA event, its protection sequence diagram and its safety evaluation in <Chapter 15>, or other USAR sections.

15A.2.6 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 consequences are mitigated following specified plant events by examining and challenging the system design.

b. To assure the consequences of a transient or accident is acceptable with the existence of a SACF or SOE criteria.

The individual structures and systems 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.2.7 UNACCEPTABLE CONSEQUENCES CRITERIA

<Table 15A.2-1>, <Table 15A.2-2>, <Table 15A.2-3>, <Table 15A.2-4>, and <Table 15A.2-5> identify the unacceptable consequences 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.2.8 GENERAL NUCLEAR SAFETY OPERATIONAL CRITERIA

The following general nuclear safety operational criteria are used to select operational requirements:

Applicability	Nuclear Safety Operational Criteria
Planned operation anticipated, abnormal operational transients, design basis accidents, and additional special plant capability events	The plant shall be operated so as to avoid unacceptable consequences.
Anticipated and abnormal operational transients and design basis 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 consequences associated with anticipated or abnormal operational transients or design basis accidents. However, this requirement is not applicable during structure, system or component repair if the availability of the safety action is maintained either by restricting the allowable repair time or by more frequently testing a redundant structure, system or component.

The unacceptable consequences associated with the different categories of plant operation and events are dictated by:

a. Probability of occurrence.

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

UNACCEPTABLE CONSEQUENCES CRITERIA

PLANT EVENT CATEGORY: NORMAL OPERATION

Unacceptable Consequences

1-1 Release of radioactive material to the environs that exceed the limits of either <10 CFR 20> or <10 CFR 50>.

1-2 Fuel failure to such an extent that, if the freed fission products were 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.2-2

UNACCEPTABLE CONSEQUENCES CRITERIA

PLANT EVENT CATEGORY: ANTICIPATED OPERATIONAL TRANSIENTS

Unacceptable Consequences

2-1	Release of radioactive material to the environs that exceeds the limits of <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 stress exceeding that allowed for transients by applicable industry codes when containment is required.

TABLE 15A.2-3

UNACCEPTABLE CONSEQUENCES CRITERIA

PLANT EVENT CATEGORY: ABNORMAL OPERATIONAL TRANSIENTS

	Unacceptable Consequences
3-1	Radioactive material release exceeding the guideline values of a small fraction of <10 CFR 100>.
3-2[1]	Failure of the fuel barrier as a result of exceeding mechanical or thermal limits.
3-3	Nuclear system stress exceeding that allowed for transients by applicable industry codes.
3-4	Containment stresses exceeding that allowed for transients by applicable industry codes when containment is required.

NOTE:

[1] Failure of the fuel barrier means gross core-wide fuel cladding perforations.

TABLE 15A.2-4

UNACCEPTABLE CONSEQUENCES CRITERIA

PLANT EVENT CATEGORY: DESIGN BASIS ACCIDENTS

Unacceptable Consequences

4-1	Radioactive material release exceeding the guideline values of <10 CFR 100> or <10 CFR 50.67> (future revisions to design basis analyses that compare consequences to 10 CFR 100 will be updated to <10 CFR 50.67>).
4-2[1]	Failure of the fuel barrier as a result of exceeding mechanical or thermal limits.
4-3	Nuclear system stresses exceeding that allowed for accidents by applicable industry codes.
4-4	Containment stresses exceeding that allowed for accidents by applicable industry codes when containment is required.
4-5	Overexposure to radiation of plant main control room personnel.

NOTE:

[1] Failure of the fuel barrier includes fuel cladding fragmentation (loss-of-coolant accident) and excessive fuel enthalpy (control rod drop accident).

TABLE 15A.2-5

UNACCEPTABLE CONSEQUENCES CONSIDERATIONS

PLANT EVENT CATEGORY: SPECIAL EVENTS

Special Events Considered

A. Reactor shutdown from outside the main control room

B. Reactor shutdown without control rods

C. Reactor shutdown with anticipated transient without scram (ATWS)

D. Spent Fuel Cask Drop

Capability Demonstration

5-1 Ability to shutdown reactor by manipulating controls and equipment outside the main control room.

5-2 Ability to bring the reactor to the cold shutdown condition from outside the main control room.

5-3 Ability to shutdown the reactor independent of control rods.

5-4 Ability to contain radiological contamination.

5-5 Ability to limit radiological exposure.

15A.3 METHOD OF ANALYSIS

15A.3.1 GENERAL APPROACH

The NSOA is performed on the plant as designed. 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.2-1> shows the process used in the analysis. The following inputs are required for the analysis of specific plant events:

a. Unacceptable Consequences Criteria <Appendix 15A.2.7>.

b. General Nuclear Safety Operational Criteria <Appendix 15A.2.8>.

c. Definition of BWR Operating States <Appendix 15A.3.2>.

d. Selection of Events for Analysis <Appendix 15A.3.3>.

e. Rules for Event Analysis <Appendix 15A.3.5>.

With this information, each selected event can be evaluated to determine systematically, the actions, the systems and the limits essential to avoiding the defined unacceptable consequences. The essential plant components and limits so identified are then considered to be in agreement with and subject to nuclear operational, design basis requirements and technical specification restrictions.

15A.3.2 BWR OPERATING STATES

Four BWR operating states in which the reactor can exist are defined in <Appendix 15A.6.2.4> and summarized 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 sections of the USAR 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 Normal Operation

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, and refueling outage.

The normal 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 when 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 when heatup ends and includes continued plant operation at power levels in excess of heatup power.

e. Achieving shutdown: Begins when 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 when achieving nuclear shutdown ends and includes all plant actions normal to the continued removal of decay heat and the reduction of RPV temperature and pressure.

The exact point at which some of the planned operations end and others begin cannot be precisely determined. It will be shown later that such precision is not required, for 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 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. Reactor pressure vessel pressure increase.

b. Reactor pressure vessel water (moderator) temperature decrease.

c. Control rod withdrawal.

d. Reactor pressure 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 threatens the integrity of the fuel as the coolant becomes unable to adequately remove the heat generated in the core. An increase in coolant flow through the core reduces the void content of the moderator, and results in an insertion of positive reactivity. Core coolant temperature increase 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 expected increase in nuclear system pressure and power.

Anticipated operational transients are defined as transients resulting from a single active component failure, SACF, or single operator error, SOE, 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 single operational failures or operator errors in this range of probability 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 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 various types of a single operator error or a single active component failure are applied to various plant systems with a consideration for a variety of plant conditions to discover events directly resulting in an undesired parameter variation. Once discovered, each event is evaluated for the threat it poses to the integrity of the radioactive material barriers.

15A.3.3.3 Abnormal Operational Transients

To select abnormal operational transients, eight nuclear system parameter variations are considered as potential initiating causes of gross core-wide fuel failures and threats of the reactor coolant pressure boundary. The parameter variations are as follows:

a. Reactor pressure vessel pressure increase.

b. Reactor pressure vessel water (moderator) temperature decrease.

c. Control rod withdrawal.

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.

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.

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. Failure of major power generation equipment components.

b. Multiple electrical failures.

c. Multiple operator errors.

d. Combinations of equipment failure and an operator error.

Operator error is defined as an active deviation from nuclear plant standard operating practices. A multiple operator error is the set of actions that is a direct consequence of several unexpected erroneous decisions.

Examples of multiple operator errors are as follows:

a. Inadvertent loading and operating a fuel assembly in an improper position.

b. Unauthorized 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 an undesired parameter variation. Once discovered, each event is evaluated for the threat it poses to the integrity of the various radioactive material barriers.

15A.3.3.4 Design Basis Accidents

Accidents are defined as hypothesized events that affect the radioactive material barriers and are not expected during plant operations. These are plant events, equipment failures, 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, reactor building and auxiliary building initially intact. (Event 35)

b. Directly to the containment. (Event 37)

c. Directly to the reactor, auxiliary or turbine buildings with the containment initially intact. (Events 35, 38, 39, 40)

d. Directly to the reactor or auxiliary buildings with the containment not intact. (Events 36, 45)

e. Directly to the spent fuel containing facilities. (Event 36)

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 consideration for the full spectrum of plant conditions, to examine events that result in the release of radioactive material.

15A.3.3.5 Special Events

A number of additional events are evaluated to demonstrate plant capabilities relative to special arbitrary nuclear safety criteria. These special events involve extremely low probability occurrence situations. As an example, the adequacy of the redundant reactivity control system is demonstrated by evaluating the special event: "reactor shutdown without control rods." Another similar example, the capability to perform a safe shutdown from outside the main control room is demonstrated by evaluating the special event "reactor shutdown from outside the main control room."

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 "normal operations" to the operating states follows from the definitions of planned operations. A planned operation is considered applicable within an operating state if the planned operation can be conducted when the reactor exists under the physical conditions defining the operating state.

15A.3.5 GUIDELINES FOR EVENT ANALYSIS

The following functional guidelines are followed in performing SACF, operational and design basis analyses for the various plant events:

a. An action, system or limit shall be 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 (c) below) shall be considered for each event analyzed so that all essential protection sequences are identified. Consideration is not limited to "worst cases" because lesser cases sometimes may require more restrictive actions or systems different from the "worst cases."

c. The initial conditions for transients, accidents and additional plant capability events shall be limited to conditions that would exist during planned operations in the applicable operating state.

d. For normal operations, consideration shall be made only for actions, limits and systems essential to avoiding the unacceptable consequences during operation in that state (as opposed to transients, accidents and additional plant capability events, which are followed through to completion). Normal operations are treated differently from other events because the transfer from one state to another during planned operations is deliberate. For events other than normal operations, the transfer from one state to another may be unavoidable.

e. Limits shall be derived only for those essential parameters that are continuously monitored by the operator. Parameter limits associated with the required performance of an essential system are considered to be included in the requirement for the operability of the system. Limits on frequently monitored process parameters are called “envelope limits,” and limits on parameters associated with the operability of a safety system are called “operability limits.” Systems associated with the control of the envelope parameters are considered nonessential if it is possible to place the plant in a safe condition without using the system in question.

f. For transients, accidents and special events, consideration shall be made for the entire duration of the event and aftermath until some planned operation is resumed. Normal operation is considered resumed when the procedures being followed or equipment being used are identical to those used during any one of the defined planned operations. Where “Extended Core Cooling” is an immediate integral part of the event, it will be included in the protection sequence. Where it may be an eventual part of the event it will not be directly added but of course can be implied to be available.

g. Credit for operator action shall be taken on a case-by-case basis depending on the conditions that would exist at the time operator action would be required. Because transients, accidents and special events are considered through the entire duration of the event until normal operation is resumed, manual operation of certain systems is sometimes required following the more rapid or automatic portions of the event. Credit for operator action is taken only when the operator can reasonably be expected to accomplish the required action under the existing conditions.

h. For transients, accidents and special events, only those actions, limits and systems shall be 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 shall identify all the support or auxiliary systems essential to the functioning of the front-line safety systems. Safety system auxiliaries whose failure results in safe failure of the front-line safety systems shall be considered nonessential.

j. A system or action that plays a unique role in the response to a transient, accident or special event shall be considered essential unless the effects of the system or action are not included in the detailed analysis of the event.

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.2-1>. 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 as indicated in <Appendix 15A.6>.

To derive the operational requirements and technical specifications for the individual components of a system included in any essential protection sequence, the following steps are taken:

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 consequences.

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 <Appendix 15A.5>.

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[1]

Conditions	States			
	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.

NOTE:

(1) Further discussion is provided in <Appendix 15A.6.2.4>.

15A.4 DISPLAY OF OPERATIONAL ANALYSIS RESULTS

15A.4.1 GENERAL

To fully identify and establish the requirements, restrictions and limitations that must be observed during plant operation, plant systems and components must be related to the needs for their actions in satisfying the nuclear safety operational criteria. This appendix displays these relationships in a series of block diagrams. System level activation as defined in the figures are not sequential order, but only show participation in the event.

<Table 15A.3-1>, <Table 15A.6-1>, <Table 15A.6-2>, <Table 15A.6-3>, <Table 15A.6-4>, and <Table 15A.6-5> indicates 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. 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 consequence. 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 consequences 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 needed to satisfy the nuclear safety operational criteria. Essentiality is determined through an analysis in which the safety action, system or limit being considered is completely disregarded in the analyses of the applicable operations or events. If the nuclear safety operational criteria are satisfied without the safety action, system or limit, then the safety action, system or limit is not essential, and no operational nuclear safety requirement would be indicated. When disregarding a safety action, system or limit results in violating one or more nuclear safety operational criteria, the safety action, system or limit is considered essential, and the resulting operational nuclear safety requirements can be related to specific criteria and unacceptable consequences.

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>.

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 (front-line 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, 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.

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.

15A.5.2 ALLOWABLE REPAIR TIMES

Allowable repair times are selected by computation using appropriate availability analysis methods 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 should only be used as needed to restore failed equipment to operation, not for routine maintenance. Using this time should be an event as rare as failure of the equipment itself. Routine maintenance should be scheduled when the equipment is not needed.

b. At the conclusion of the repair, the repaired component must be retested and placed in service.

c. Once the need for repair of a failed component is discovered, repairs should proceed as quickly as possible consistent with good craftmanship.

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), <Appendix 15A.9>.

15A.6 OPERATIONAL ANALYSES

Results of the operational analyses are discussed in the following paragraphs and displayed on <Figure 15A.6-1>, <Figure 15A.6-2>, <Figure 15A.6-3>, <Figure 15A.6-4>, <Figure 15A.6-5>, <Figure 15A.6-6>, <Figure 15A.6-7>, <Figure 15A.6-8>, <Figure 15A.6-9>, <Figure 15A.6-10>, <Figure 15A.6-11>, <Figure 15A.6-12>, <Figure 15A.6-13>, <Figure 15A.6-14>, <Figure 15A.6-15>, <Figure 15A.6-16>, <Figure 15A.6-17>, <Figure 15A.6-18>, <Figure 15A.6-19>, <Figure 15A.6-20>, <Figure 15A.6-21>, <Figure 15A.6-22>, <Figure 15A.6-23>, <Figure 15A.6-24>, <Figure 15A.6-25>, <Figure 15A.6-26>, <Figure 15A.6-27>, <Figure 15A.6-28>, <Figure 15A.6-29>, <Figure 15A.6-30>, <Figure 15A.6-31>, <Figure 15A.6-32>, <Figure 15A.6-33>, <Figure 15A.6-34>, <Figure 15A.6-35>, <Figure 15A.6-36>, <Figure 15A.6-37>, <Figure 15A.6-38>, <Figure 15A.6-39>, <Figure 15A.6-40>, <Figure 15A.6-41>, <Figure 15A.6-42>, <Figure 15A.6-43>, <Figure 15A.6-44>, <Figure 15A.6-45>, <Figure 15A.6-46>, <Figure 15A.6-47>, <Figure 15A.6-48>, <Figure 15A.6-49>, <Figure 15A.6-50>, and <Figure 15A.6-51> and in <Table 15A.6-1>, <Table 15A.6-2>, <Table 15A.6-3>, <Table 15A.6-4>, and <Table 15A.6-5>.

15A.6.1 SAFETY SYSTEM AUXILIARIES

<Figure 15A.6-1> and <Figure 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 <Figure 15A.6-47>, <Figure 15A.6-48>, <Figure 15A.6-49>, <Figure 15A.6-50>, and <Figure 15A.6-51>.

15A.6.2 NORMAL OPERATIONS

15A.6.2.1 General

Requirements for the normal or planned operations normally involve limits (L) on certain key process variables and restrictions (R) on certain plant equipment. The control block diagrams for each operating state <Figure 15A.6-3>, <Figure 15A.6-4>, <Figure 15A.6-5>, and <Figure 15A.6-6> show only those controls necessary to avoid unacceptable safety consequences, 1-1 through 1-4 of <Table 15A.2-1>. <Table 15A.6-1> summarizes additional information for Normal Operation.

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 consequences.

15A.6.2.2 Event Definitions

a. 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:

   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. 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.

c. 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.

d. Event 4 - Power Operation - Electric Generation

Power operation begins where heatup ends and includes continued plant operation at power levels in excess of heatup power or steady-state operation. It also includes plant maneuvers such as:

1. Daily electrical load reduction and recoveries.

2. Electrical grid frequency control adjustment.

3. Control rod movements.

4. Power generation surveillance testing involving:

    (a) Turbine stop valve closing.

(b) Turbine control valve adjustments.

(c) MSIV exercising.

e. 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.

f. 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 Consequences

The following paragraphs describe the safety actions for planned operations. Each description includes a selection of the operating states that apply to the safety action, the plant system affected by limits or restrictions and the unacceptable consequence that is avoided. The four operating states are defined in <Table 15A.3-1>. The unacceptable consequences criteria are tabulated in <Table 15A.2-1>.

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 main vent. Gaseous releases through other vents are monitored by the ventilation monitoring system. The process liquid radiation monitors are not required, because all liquid wastes are monitored by batch sampling before a 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 l-1).

15A.6.2.3.2 Core Coolant Flow Rate Control

In State D, when above approximately 10 percent Nuclear Boiler (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.6.2.3.5 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 135 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 overtension 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 shutdown and is generating less than 20 percent power (State 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 suppression pool temperature to maintain containment 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 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).

15A.6.2.4 <u>Operational Safety Evaluations</u>

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 shutdown. Applicable planned operations are achieving criticality and cooldown (Events 2 and 6, respectively).

Sequence diagrams relating safety actions for planned operations, plant systems and applicable events are shown 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 building pressure and temperature control
Spent fuel storage 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 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 building pressure and temperature control
- Spent fuel storage shielding, cooling and reactivity control

15A.6.3 ANTICIPATED 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 <Figure 15A.6-7>, <Figure 15A.6-8>, <Figure 15A.6-9>, <Figure 15A.6-10>, <Figure 15A.6-11>, <Figure 15A.6-12>, <Figure 15A.6-13>, <Figure 15A.6-14>, <Figure 15A.6-15>, <Figure 15A.6-16>, <Figure 15A.6-17>, <Figure 15A.6-18>, <Figure 15A.6-19>, <Figure 15A.6-20>, <Figure 15A.6-21>, <Figure 15A.6-22>, <Figure 15A.6-23>, <Figure 15A.6-24>, <Figure 15A.6-25>, <Figure 15A.6-26>, <Figure 15A.6-27>, <Figure 15A.6-28>, and <Figure 15A.6-29.). The auxiliaries for the front-line safety systems are indicated in the auxiliary diagrams <Figure 15A.6-1> and <Figure 15A.6-2> and the commonality of auxiliary diagrams <Figure 15A.6-47>, <Figure 15A.6-48>, <Figure 15A.6-49>, <Figure 15A.6-50>, and <Figure 15A.6-51>.

15A.6.3.2 Required Safety Actions/Related Unacceptable Consequences

The following list relates that safety actions for anticipated operational transients to mitigate or prevent the unacceptable safety consequences. Refer to <Table 15A.2-2> for the unacceptable consequences criteria.

Safety Action	Related Unacceptable Consequences Criteria	Reason Action Required
Scram and/or RPT	2-2 2-3	To prevent fuel damage and to limit RPV system pressure rise.
Pressure Relief	2-3	To prevent excessive RPV system pressure rise.
Core and Containment Cooling	2-1, 2-2 2-4	To prevent fuel and containment damage in the event that normal cooling is interrupted.
Reactor Vessel Isolation	2-2	To prevent fuel damage by reducing the outflow of steam and water from the reactor vessel, thereby limiting the decrease in reactor vessel water level.
Restore ac Power	2-2	To prevent fuel damage by restoring ac power to systems essential to other safety actions.
Prohibit Rod Motion	2-2	To prevent exceeding fuel limits during transients.
Containment Isolation	2-1, 2-4	To minimize radiological effects.

15A.6.3.3 Event Definitions and Operational Safety Evaluations

a. Event 7 - Manual or 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 purpose, 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 basis design of the plant systems. <Figure 15A.6-7> identifies the protection sequences for this event.

b. Event 8 - Loss-of-Plant Instrument or Service Air System

Loss of all plant instrument or service air system causes the closure of isolation valves and reactor shutdown. Although these actions occur, they are not a requirement to prevent unacceptable consequences 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 resulting 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> and <Figure 15A.6-14> show 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 boiler 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 in accumulator tanks for the continual operation of the ADS safety/relief valves until reactor shutdown is accomplished.

c. Event 9 - Inadvertent HPCS 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 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.

d. Event 10 – Inadvertent 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.

e. Event 11 - Recirculation Flow Control Failure (Increasing Flow)

A recirculation flow control failure causing increased flow is applicable in States C and D. In State D, the resulting increase in power level is limited by a reactor 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.

f. Event 12 - Recirculation Flow Control Failure (Decreasing Flow)

This recirculation flow control malfunction causes a decrease in core coolant flow. This event is not applicable to States A and B because the reactor vessel head is off and the recirculation pumps normally would not be in use.

The number and type of flow controller failure modes determine the protection sequence for the event. For flow control valve control systems, the fast closure of one or two control valves results in the protective sequence of <Figure 15A.6-12>.

g. 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 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 thermohydraulic 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 steamline isolation soon occurs after the initiation of the RCIC/HPCS systems on water Level 2. Relief valve actuation will follow.

h. 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.6-14 (1)>, 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 and high-pressure core and RCIC cooling systems supply 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.6-14 (2)>, the scram safety action is accomplished through the combined actions of the neutron monitoring, reactor protection and control rod drive systems.

i. 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 regulator closes the main turbine control valves enough to stabilize pressure at a level slightly below the initial value. There are no unique safety system requirements for this event.

If the event occurs when the feedwater system is not active in State D, a loss in the coolant inventory results in a reactor vessel isolation. The low water level signal initiates reactor vessel isolation. The nuclear boiler pressure relief system provides pressure relief.

Core cooling is accomplished by the RCIC and HPCS systems which are 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 and HPCS, which are initiated on low water levels by the IDC system or are manually operated. Containment-suppression pool cooling is manually initiated.

j. Event 16 - Control Rod Withdrawal Error During Refueling and Startup Operations

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 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 <Figure 15A.6-16>.

k. 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 <Figure 15A.6-17>.

Systems in the power range (0 to 100 percent NBR) prevent the selection of an out-of-sequenced rod movement by using the Rod Pattern Control System (RPCS) which uses either Banked Position or Grouped Notch Withdrawal sequences. In addition, the movement of the rod is monitored and limited within acceptable intervals either by neutronic effects or actual rod motion. The RC&IS provides movement surveillance. Beyond these rod motion control limits are the fuel/core SCRAM protection systems. In State C, no protective actions are described above.

l. Event 18 - Loss of Shutdown Cooling

The loss of RHRS 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 RHRS-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 RHRS suppression pool cooling mode (both manually operated) can be used to maintain water level and remove decay heat. Suppression pool cooling is actuated to remove heat energy from the suppression pool system.

m. Event 19 - RHR Shutdown Cooling - Increased Cooling

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 RPV system pressure is too high to permit operation of the shutdown cooling (RHRS) <Figure 15A.6-19>. No unique safety actions are required to avoid the unacceptable safety consequences for transients as a result of a reactor coolant temperature decrease induced by misoperation of the 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.

n. 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 restoration of RPV water level by RCIC and HPCS. In State C the reactor water low (Level 3) scram can be bypassed when the plant mode switch is in the ‘SHUTDOWN’ position.

As shown in <Figure 15A.6-20>, the reactor protection and control rod drive systems effect a scram on water Level 3. The primary containment and reactor vessel isolation control system (PCRVICS) and the main steam line isolation valves act to isolate the reactor vessel on water Level 1. 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 RPV pressure relief system. Either the RCIC or HPCS system can maintain adequate water level for initial core cooling and to restore and maintain water level. For long term shutdown and extended core coolings, containment/suppression pool cooling systems are manually initiated.

The requirements for operating State C is the same as for State D except that the scram action is not required in State C.

o. 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 reduction 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 setpoint. 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.

p. Event 22 - Feedwater Controller Failure - Maximum Demand

A feedwater controller failure, causing an excess of coolant inventory in the reactor vessel, is possible in all operating states. Feedwater controller failures considered are those that would give failures of automatic flow control, manual flow control or feedwater bypass valve control. In operating States A and B, no safety actions are required since the vessel head is removed and the moderator temperature is low. In operating State D, any positive reactivity effects 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 RPV pressure relief system. Initial restoration of the core water level is by the RCIC and HPCS systems. Prolonged isolation may require extended core cooling and containment/suppression pool cooling.

q. Event 23 - Pressure Regulator Failure (Open Direction)

A pressure regulator failure in the open direction, causing the opening of the turbine control valves and opening of all of the turbine bypass valves, applies only in operating States C and D, because in other states the pressure regulator is not in operation. From the viewpoint of rapid depressurization, a pressure regulator failure is most severe and rapid in operating State D at low power. From the viewpoint of fuel and post-isolation conditions, it is most limiting at full 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 high reactor water level (L8) or on closure of the MSIV's due to low pressure in the steam lines. 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 the low pressure setpoint. Under other conditions, isolation if necessary can be manually initiated. After isolation is completed, decay heat will cause reactor vessel pressure to increase until limited by the operation of the safety/relief valves. Core cooling following isolation can be provided by 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.

r. Event 24 - Pressure Regulator Failure - Closed

A pressure regulator failure in the closed direction (or downscale), causing the closing of turbine control valves, applies only in operating States C and D, because in other states the pressure regulator is not in operation.

A single pressure regulator failure downscale would result in little or no effect on the plant operation. The second pressure regulator would provide turbine-reactor control. If the second unit failed this would result in the worst situation, yet it is much less severe than Events 25, 27, 30, and 31. The dual pressure regulator failures are most severe and rapid in operating State D at high power.

The various protection sequences giving the safety actions are shown in <Figure 15A.6-24>. Upon failure of one pressure regulator downscale, normally a backup regulator will maintain the plant in the present status upon the initial regulator downscale failure.

An additional single active component failure of the backup regulator will result in a high flux or pressure SCRAM, system isolation and subsequent extended isolation core cooling system actuations.

s. Event 25 - Main Turbine Trips (With Bypass System Operation)

A main turbine trip can occur only in operating State D (during heatup or power operation). A turbine trip during heatup is not as severe as a trip at full power because the initial power level is less than 38 percent, thus minimizing the effects of the transient and enabling return to planned operations via the bypass system operation. For a turbine trip above 38 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 will result at water Level 1 and RCIC and HPCS system initiation will result from water Level 2. <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.

t. 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 38 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) and will also initiate isolation, pressure relief valve actuation, RCIC, and HPCS initial core cooling. A scram can also be 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 38 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 suppression pool cooling. When the RPV depressurizes sufficiently, the low pressure core cooling systems provide core cooling until a planned operation via RHRS shutdown cooling is achieved.

u. Event 27 - Main Generator Trip (With Bypass System Operation)

A main generator trip with bypass system operation can occur only in operating State D (during heatup or power operation). Fast closure of the main turbine control valves is initiated whenever an electrical grid disturbance occurs 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 38 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 can result, pressure relief and initial core cooling by RCIC and HPCS will take place. Prolonged shutdown of the turbine generator unit will necessitate extended core and containment cooling. A generator trip during heatup (<38 percent) is not severe because the turbine bypass system can accommodate the decoupling of the reactor and the turbine generator unit, thus minimizing the effects of the transient and enabling return to planned operations. <Figure 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.

v. Event 28 - Auxiliary Transformer Failure

There is a variety of possible plant electrical component failures which could affect the reactor system. The loss of auxiliary power 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 the safety actions required to accommodate a loss of 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 primary containment and reactor vessel isolation control system (PCRVICS) 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 RPV pressure relief system. With continued isolation, decay heat may cause increased RPV pressure, and periodically lift relief valves which will cause reactor vessel water level to decrease. The core and containment cooling sequences shown in <Figure 15A.6-28> denote the short and long term actions for achieving adequate cooling.

w. Event 29 - Loss of Offsite Power - Grid Loss

There is a variety of plant-grid electrical component failures which can affect reactor operation. The loss of both onsite 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 effect a scram from main turbine trip or loss of reactor protection system power sources. The turbine trip will initiate recirculation pump trip (RPT). The primary containment and reactor vessel isolation control system (PCRVICS) 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 continues to increase RPV pressure, periodically lifting relief valves and causing 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 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 <Figure 15A.6-30>, <Figure 15A.6-31>, <Figure 15A.6-32>, <Figure 15A.6-33>, and <Figure 15A.6-34>. The auxiliaries for the front-line safety systems are indicated in the auxiliary diagrams <Figure 15A.6-1> and <Figure 15A.6-2> and the commonality of auxiliary diagrams <Figure 15A.6-47>, <Figure 15A.6-48>, <Figure 15A.6-49>, <Figure 15A.6-50>, and <Figure 15A.6-51>.

15A.6.4.2 Required Safety Actions/Related Unacceptable Consequences

The following list relates the safety actions for abnormal operational transients to mitigate or prevent the unacceptable safety consequences cited in <Table 15A.2-3>.

Safety Action	Related Unacceptable Consequences Criteria	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, Suppression Pool and Containment Cooling	3-2 3-4	To limit further fuel and containment damage in the event that normal cooling is interrupted.

Safety Action	Related Unacceptable Consequences Criteria	Reason Action Required
Reactor Vessel Isolation	3-2	To limit further fuel damage by reducing the outflow of steam and water from the reactor vessel, thereby limiting the decrease in reactor vessel water level.
Restore ac Power	3-2	To limit initial fuel damage by restoring ac power to systems essential to other safety actions.
Containment Isolation	3-1	To limit radiological effects.

15A.6.4.3 Event Definition and Operational Safety Evaluation

a. 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 bypass operation results in the same situation as the 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 38 percent. A scram, RPT, isolation, relief valve, and RCIC and HPCS operation will immediately result in prolonged shutdown, which will follow the same pattern as Event 27.

The thermohydraulic and thermodynamic effects on the core, of course, are more severe than with the bypass operating. Since the event is of lower probability than Event 27, the unacceptable consequences are less limiting.

The load rejection and turbine trip are similar abnormal operational transients and, although main generator trip is a less severe transient than a turbine trip due to the rapid closure of the turbine stop valves, the required safety actions are the same.

b. 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. Main turbine trip at plant operation above or below 38 percent power, with bypass system failure, will result in the same transient effects: a scram, a RPT, an isolation, subsequent relief valve actuation, and immediate RCIC and HPCS actuation. After initial shutdown, extended core and containment cooling will be required as noted previously in Event 27.

Turbine trips without bypass system operations results in very severe thermohydraulic impacts on the reactor core. The allowable limit or acceptable calculational techniques for this event is less restrictive since the event is of lower probability of occurrence than the turbine trip with a bypass operation event.

c. Event 32 - Inadvertent Loading and Operation with Fuel Assembly in Improper Position

Operation with a fuel assembly in the improper position is shown in <Figure 15A.6-32> and can occur in all operating states. No protection sequences are necessary relative to this event. Calculated results of worst fuel handling loading error will not cause fuel cladding integrity damage. It requires three independent equipment/operator errors to allow this situation to develop.

d. 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 involves operation at design power in State D. A main turbine trip will occur as vessel water 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 and temperature. 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>.

e. 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 involves operation at design power in State D. A main turbine trip will occur as vessel water 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 and temperature. 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 ACCIDENTS

15A.6.5.1 General

The safety requirements and protection sequences for accidents are described in the following paragraphs for Events 35 through 45. The protection sequence block diagrams show the safety actions and the sequence of front-line safety systems used for the accidents (refer to <Figure 15A.6-35>, <Figure 15A.6-36>, <Figure 15A.6-37>, <Figure 15A.6-38>, <Figure 15A.6-39>, <Figure 15A.6-40>, <Figure 15A.6-41>, <Figure 15A.6-42>, and <Figure 15A.6-43>). The auxiliaries for the front-line safety systems are indicated in the auxiliary diagrams <Figure 15A.6-1> and <Figure 15A.6-2> and the commonality of auxiliary diagrams <Figure 15A.6-47>, <Figure 15A.6-48>, <Figure 15A.6-49>, <Figure 15A.6-50>, and <Figure 15A.6-51>.

15A.6.5.2 Required Safety Actions/Unacceptable Consequences

The following list relates the safety actions for design basis accident to mitigate or prevent the unacceptable consequences cited in <Table 15A.2-4>.

Safety Action	Related Unacceptable Consequences Criteria	Reason Action Required
Scram	4-2 4-3	To prevent fuel cladding failure[1] and to prevent excessive nuclear system pressures.
Pressure Relief	4-3	To prevent excessive nuclear system pressure.
Core Cooling	4-2	To prevent fuel cladding failure.
Reactor Vessel Isolation	4-1	To limit radiological effect to not exceed the guideline values of <10 CFR 100>.

Safety Action	Related Unacceptable Consequences Criteria	Reason Action Required
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 Environmental Control	4-5	To prevent overexposure to radiation of plant personnel in the control room.
Limit Reactivity Insertion Rate (passive)	4-2 4-3	To prevent fuel cladding failure and to prevent excessive nuclear system pressure.

NOTE:

(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

a. 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 in a shutdown state by more than the reactivity worth of 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. After the reactor has been scrammed, core cooling is accomplished by either the RCIC or the HPCS or the normal feedwater system.

b. Event 36 - Fuel Handling Accident Outside Containment

Because a fuel-handling accident can potentially occur any time when fuel assemblies are being manipulated in the fuel handling building, 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>.

c. Event 37 - Loss-of-Coolant Accidents Resulting from Postulated Piping Breaks Within RCPB Inside Containment (DBA-LOCA)

Pipe breaks inside the containment are considered only when the nuclear system is significantly pressurized (States C and D). The result is a release of steam and water into the containment. Consistent with NSOA criteria, the protection requirements consider all size line breaks 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.6-37>, 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, ADS, LPCI, and LPCS), containment and reactor vessel isolation control system, reactor/shield/auxiliary buildings, annulus exhaust gas treatment system, control room heating, cooling and ventilation system, emergency service water systems, hydrogen control 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 RHRS (suppression pool cooling mode) and ADS or relief valves (controlled depressurization) is required to maintain containment pressure and fuel cladding temperature within limits during extended core cooling.

d. Events 38, 39, 40 - Loss-of-Coolant Accidents (LOCA) Resulting from Postulated Pipe Breaks - Outside Containment

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 subset of the State D sequence.

The protection sequences for the various possible pipe breaks outside the containment are shown in <Figure 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 the LPCS or LPCI. These systems provide parallel paths to effect initial core cooling, thereby satisfying the single-failure criterion. Extended core cooling is accomplished by the single-failure proof, parallel combination of LPCS, HPCS and LPCI systems. The ADS or relief valve system operation and the RHRS suppression pool cooling mode (both manually operated) are required to maintain containment pressure and fuel cladding temperature within limits during extended core cooling.

e. 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, and is shown in <Figure 15A.6-39>.

The reactor operator initiates 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 <Figure 15A.6-26>.

f. 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. Protective sequences for the event are shown in <Figure 15A.6-40>.

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 <Figure 15A.6-26>.

g. Event 43 - Radioactive Liquid Waste System Failure (Release to Atmosphere)

Releases which could occur inside and outside of the containment, not covered by Events 35, 36, 37, 38, 39, 40, 42, and 50 will include equipment leaks and failures inside structures housing the subject process equipment. Conservative values for releases have been assumed and evaluated.

The protective sequences for this event are provided in <Figure 15A.6-41>.

h. Event 44 - Postulated Radioactive Releases Due to Liquid Containing Tank Failures

The postulated events that could cause release of the radioactive inventory of a waste tank include a tank failure and/or an operator error. The possibility of a tank failure and consequent release rates receives primary consideration in system and component design. The tanks and piping to the first isolation valve are Safety Class 3, Quality Group C components and are designed to Seismic Category I. The concentrator waste tanks are 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 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>.

i. Event 45 - Fuel Handling Accident Inside Containment

A fuel-handling accident inside containment can only occur when fuel assemblies are being manipulated over the reactor core. Therefore, this accident is only considered in operating State A. 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.6-43>.

15A.6.6 SPECIAL EVENTS

15A.6.6.1 General

Additional special events are postulated to demonstrate that the plant is capable of accommodating off-design occurrences. (Refer to Events 46 through 49.) As such, these events are beyond the safety requirements of the other event categories. The safety actions shown on the sequence diagrams <Figure 15A.6-44>, <Figure 15A.6-45>, and <Figure 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 <Figure 15A.6-1>, <Figure 15A.6-2>, <Figure 15A.6-47>, <Figure 15A.6-48>, <Figure 15A.6-49>, <Figure 15A.6-50>, and <Figure 15A.6-51>.

15A.6.6.2 Required Safety Action/Unacceptable Consequences

The following list relates the safety actions for special events to prevent the unacceptable consequences cited in <Table 15A.2-5>:

Safety Action	Related Unacceptable Consequences Criteria	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.
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

a. Event 46 - Spent Fuel Cask Drop

As discussed in <Section 9.1.4.2.2.2>, Fuel Handling Area Crane, the hoist for this crane’s main hook that handles spent fuel casks has been upgraded to single-failure-proof in accordance with applicable guidelines of NRC <NUREG-0554> (Single-Failure-Proof Cranes for Nuclear Power Plants, May 1979) and NRC <NUREG-0612> (Control of Heavy Loads at Nuclear Power Plants, July 1980) to support spent fuel dry storage cask handling activities. With the hoist for the crane’s main hook qualified as single-failure-proof, a cask drop accident is not a credible event and need not be postulated.

b. Event 47 - Reactor Shutdown - ATWS

Reactor shutdown from a plant transient occurrence (e.g., turbine trip) without the use of mechanical control rods is an event which has been evaluated to determine the capability of the plant to be safely shutdown. The event is applicable in any operating state. <Figure 15A.6-44> 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 and RCIC 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 RPV pressure falls to 100 to 200 psig level, the RHRS 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.

c. Event 48 - Reactor Shutdown From Outside the Control Room

Reactor shutdown from outside the 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.6-45> 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 and 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 below 135 psig level, the RHRS shutdown cooling mode is started.

d. Event 49 - 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 consequence 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.6-46>, the standby liquid control system is manually initiated and controlled in States B and D.

TABLE 15A.6-1

NORMAL OPERATION

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.	BWR Operating State A	B	C	D
1	Refueling	<Figure 15A.6-3>	-	X			
	- Initial	<Figure 15A.6-4>		X			
	- Reload	<Figure 15A.6-5>		X			
		<Figure 15A.6-6>		X			
2	Achieving Criticality	<Figure 15A.6-3>	-	X	X	X	X
		<Figure 15A.6-4>		X	X	X	X
		<Figure 15A.6-5>		X	X	X	X
		<Figure 15A.6-6>		X	X	X	X
3	Heatup	<Figure 15A.6-6>					X
4	Power Operation - Generation - Steady-State - Daily Load Reduction and 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	<Figure 15A.6-6>	-				X

TABLE 15A.6-1 (Continued)

NORMAL OPERATION

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.	BWR Operating State A	B	C	D
5	Achieving Shutdown	<Figure 15A.6-4>	-		X		X
		<Figure 15A.6-6>			X		X
6	Cooldown	<Figure 15A.6-3>	-	X		X	
		<Figure 15A.6-5>		X		X	

TABLE 15A.6-2

ANTICIPATED OPERATIONAL TRANSIENTS

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.	BWR Operating State A	B	C	D
7	Manual or Inadvertent SCRAM	<Figure 15A.6-7>	<Section 7.2>	X	X	X	X
8	Loss of Plant Instrument or Service Air Systems	<Figure 15A.6-8>	<Section 9.3.1>	X	X	X	X
9	Inadvertent Startup of HPCS Pump	<Figure 15A.6-9>	<Section 15.5.1>	X	X	X	X
10	Inadvertent Startup of Idle Recirculation Loop Pump	<Figure 15A.6-10>	<Section 15.4.4>	X	X	X	X
11	Recirculation Loop Flow Control Failure with Increasing Flow	<Figure 15A.6-11>	<Section 15.4.5>			X	X
12	Recirculation Loop Flow Control Failure with Decreasing Flow	<Figure 15A.6-12>	<Section 15.3.2>			X	X
13	Recirculation Loop Pump Trip - With One Pump - With Two Pumps	<Figure 15A.6-13>	<Section 15.3.1>			X	X
14	Inadvertent MSIV Closure		<Section 15.2.4>				
	- With Four Valves	<Figure 15A.6-14 (1)>				X	X
	- With One Valve	<Figure 15A.6-14 (2)>				X	X

TABLE 15A.6-2 (Continued)

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.	BWR Operating State A	B	C	D
15	Inadvertent Operation of One Safety/Relief Valve	<Figure 15A.6-15>	<Section 15.6.1>	X	X	X	X
	- Opening/Closing						
	- Stuck Open						
16	Continuous Control Rod Withdrawal Error	<Figure 15A.6-16>	<Section 15.4.1>				
	- During Startup				X		
	- During Refueling			X			
17	Continuous Control Rod Withdrawal Rod Error at Power	<Figure 15A.6-17>	<Section 15.4.2>			X	X
18	RHRS - Shutdown Cooling Failure Loss of Cooling	<Figure 15A.6-18>	<Section 15.2.9>	X	X	X	X
19	RHRS - Shutdown Cooling Failure Increased Cooling	<Figure 15A.6-19>	<Section 15.1.6>	X	X	X	X
20	Loss of All Feedwater Flow	<Figure 15A.6-20>	<Section 15.2.7>			X	X
21	Loss of Feedwater Heater	<Figure 15A.6-21>	<Section 15.1.1>				X
22	Feedwater Controller Failure Maximum Demand - Low Power	<Figure 15A.6-22>	<Section 15.1.2>	X	X	X	X
23	Pressure Regulator Failure - Open	<Figure 15A.6-23>	<Section 15.1.3>			X	X

TABLE 15A.6-2 (Continued)

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.	BWR Operating State A	B	C	D
24	Pressure Regulator Failure - Closed	<Figure 15A.6-24>	<Section 15.2.1>			X	X
25	Main Turbine Trip with Bypass System Operational	<Figure 15A.6-25>	<Section 15.2.3>				X
26	Loss of Main Condenser Vacuum	<Figure 15A.6-26>	<Section 15.2.5>			X	X
27	Main Generator Trip (Load Rejection with Bypass System Operational)	<Figure 15A.6-27>	<Section 15.2.2>				X
28	Loss of Plant Normal Onsite AC Power - Auxiliary Transformer Failure	<Figure 15A.6-28>	<Section 15.2.6>	X	X	X	X
29	Loss of Plant Normal Offsite AC Power - Grid Connection Failure	<Figure 15A.6-29>	<Section 15.2.6>	X	X	X	X

TABLE 15A.6-3

ABNORMAL OPERATIONAL TRANSIENTS

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.	BWR Operating State A	B	C	D
30	Main Generator Trip (Load Rejection with Bypass System Failure)	<Figure 15A.6-30>	<Section 15.2.2>				X
31	Main Turbine Trip with Bypass System Failure	<Figure 15A.6-31>	<Section 15.2.3>				X
32	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	<Figure 15A.6-32>	<Section 15.4.7>	X	X	X	X
33	Recirculation Loop Pump Seizure for One Loop	<Figure 15A.6-33>	<Section 15.3.3>				X
34	Recirculation Loop Pump Shaft Break	<Figure 15A.6-34>	<Section 15.3.4>				X

TABLE 15A.6-4

DESIGN BASIS ACCIDENTS

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.	BWR Operating State A	B	C	D
35	Control Rod Drop Accident	<Figure 15A.6-35>	<Section 15.4.9>				X
36	Fuel Handling Accident Outside Containment	<Figure 15A.6-36>	<Section 15.7.4>	X	X	X	X
37	Loss-of-Coolant Accident[1] Resulting from Spectrum of Postulated Piping Breaks Within the RCPB Inside Containment	<Figure 15A.6-37>	<Section 15.6.5>			X	X
38	Small, Large, Steam, and Liquid Piping Breaks Outside Containment	<Figure 15A.6-38>	<Section 15.6.4>			X	X
39	Instrument Line Break Outside Drywell	<Figure 15A.6-38>	<Section 15.6.2>			X	X
40	Feedwater Line Break Outside Containment	<Figure 15A.6-38>	<Section 15.6.6>			X	X
41	Gaseous Radwaste System Leak or Failure	<Figure 15A.6-39>	<Section 15.7.1>	X	X	X	X

TABLE 15A.6-4 (Continued)

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.	BWR Operating State A	B	C	D
42	Augmented Offgas Treatment System Failure	<Figure 15A.6-40>	<Section 15.7.1>	X	X	X	X
43	Radioactive Liquid Waste System Failures (Release to Atmosphere)	<Figure 15A.6-41>	<Section 15.7.2>	X	X	X	X
44	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	<Figure 15A.6-42>	<Section 15.7.3>	X	X	X	X
45	Fuel Handling Accident Inside Containment	<Figure 15A.6-43>	<Section 15.7.6>	X			

NOTE:

(1) small, intermediate and large

TABLE 15A.6-5

SPECIAL EVENTS

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.	BWR Operating State A	B	C	D
46	Shipping Cask Drop - Spent Fuel	N/A	<Section 15.7.5>	X	X	X	X
47	Reactor Shutdown from Anticipated Transient Without SCRAM (ATWS)	<Figure 15A.6-44>	<Section 15.8>	X	X	X	X
48	Reactor Shutdown from Outside the Control Room	<Figure 15A.6-45>	<Section 7.4>	X	X	X	X
49	Reactor Shutdown Without Control Rods	<Figure 15A.6-46>	<Section 9.3.5>	X	X	X	X

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 that can be 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>

RELOAD SAFETY ANALYSIS

CYCLE 19 (RELOAD 18)

APPENDIX 15B

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

<u>TABLE OF CONTENTS</u> (Continued)

TABLE OF CONTENTS (Continued)

APPENDIX 15B

LIST OF TABLES

APPENDIX 15B

LIST OF FIGURES

15B.0 INTRODUCTION

The main text of the Updated Safety Analysis Report (USAR) describes in detail the original design and analysis of Unit 1 of the Perry Nuclear Power Plant including subsequent plant changes. This Appendix describes the reload safety analysis performed for subsequent cycles to supplement the original first cycle safety analysis.

The original design of Perry plant described in the main text is intended to be valid for the licensed life of the plant except that the safety analysis was performed using the initial cycle GE fuel design and plant parameters. The reload analysis is performed to establish or re-verify that the plant will continue to meet the criteria that determine the safety analysis licensing basis as new fuel designs are introduced, or plant modifications are made which affect the transient or accident analysis performance for subsequent cycles.

Each Core Operating Limits Report (COLR) submittal (or, when required, a reload license amendment) is supported by a Supplemental Reload Licensing Report that provides the results of the reload safety analysis and the technical basis for this appendix.

The following sections describe the impact of the reload on each chapter of the USAR. The reload affects the fuel related design description and analyses, plant transient responses and accident analyses for the current reload cycle. In general, the reload safety analysis affects mostly <Chapter 4>, <Chapter 5>, <Chapter 6> and <Chapter 15>. Therefore, a description of the effects on these chapters represents the main content of this appendix. Only those chapters/sections that are directly affected are reviewed for impact. Section and chapter numbering corresponds to the numbering used in the main text (e.g., the ECCS performance evaluation is described in <Appendix 15B.6.3.3>, in the main text it is described in <Section 6.3.3>. <Section 6.1> or <Section 6.2> do not appear in this appendix, since they do not involve the reload). This chapter uses designation for references as "Reference 15B-XX" and "Reference XX". Both formats refer to references in this chapter.

15B.0.1 SUMMARY

During the refueling outage approximately one-third of the core is removed and replaced with an equal number of fresh or reinserted bundles. The current cycle fuel bundle information and reference loading pattern are presented in <Table 15B.4.3-1> and <Figure 15B.4.3-1>. This fuel and loading configuration allows the plant to operate at full power for the number of effective full power days (EFPDs) specified in <Table 15B.4.3-1> before the next required fuel reload.

In order to evaluate the safety impact of the fuel reload, fuel lattice physics calculations, 3-D simulation, transient and accident evaluations were performed. The approved methodologies described in the General Electric Standard Application for Reactor Fuel: GESTAR II (hereafter referred to as GESTAR - Reference 15B-4) were used in the evaluations. Equipment out-of-service and expanded operating modes described in Appendices 15D, 15E, and 15F are also included. Specifically, the evaluations utilize the following methodologies and some key input parameters that are different from the initial cycle analysis:

a. (Deleted)

b. GEMINI system of analysis methods.

c. GEXL-PLUS thermal correlation.

d. Two out-of-service Safety/Relief valves were assumed for the transient analysis (six out-of-service Safety/Relief valves were assumed for the overpressure protection analysis - corresponding to Technical Specification 3.4.4 Safety/Relief Valves).

Analysis for the following extended operating domains/modes of operation was performed to re-verify or determine operating limits for this cycle of operation with:

a. Feedwater heater(s) out-of-service during the cycle, corresponding to a reduction of up to 100°F in feedwater temperature at rated power <Appendix 15D>.

b. Planned operation with feedwater heater(s) out-of-service to extend an operating cycle (beyond the end of the normal fuel cycle), corresponding to a reduction of up to 170°F in feedwater temperature at rated power <Appendix 15D>.

   The Oscillation Power Range Monitor setpoint methods have been changed to NEDE-33766P-A, GEH Simplified Stability Solution (GS3) (Reference 15B-51). The GS3 methods are a generic approach to establishing the OPRM Period Based Detection Algorithm Setpoints. This generic approach assumes feedwater temperature reductions are limited to 120°F anytime during the cycle.

   Reload core designs will continue to be developed assuming feedwater temperature reductions of 100°F during the cycle and 170°F beyond the end of the normal fuel cycle. Plant operations will be limited to 100°F during the cycle to account for the assumptions of the core design and 120°F beyond the end of the normal fuel cycle to account for the GS3 limitation.

c. Operation with one recirculation loop out-of-service <Appendix 15F>.

d. Extension of the power/flow operating region as defined by the Maximum Extended Operating Domain (MEOD) boundary <Appendix 15E>.

e. Operation with one pressure regulator out of service.

f. Operation with power load unbalance out of service.

Four main safety areas are re-evaluated for each reload analysis. They are:

a. Shutdown margin demonstration,

b. Transient Minimum Critical Power Ratio (MCPR) responses,

c. Overpressure protection, and

d. Emergency core cooling system performance (with the new fuel bundle types) during the postulated design basis loss-of-coolant accident.

A shutdown margin (SDM) calculation is performed to demonstrate that the core is capable of being made subcritical with sufficient margin in the most reactive condition throughout the cycle with the strongest control rod withdrawn. In addition, a SDM calculation is performed to demonstrate that the core can be brought to and maintained in a subcritical condition through boron injection from the Standby Liquid Control System (SLC). These two calculations demonstrate that the shutdown margin requirement is met for this reload cycle.

The analysis of the anticipated operational occurrences (transients) was conducted according to the methodologies described in GESTAR (Reference 15B-4). Limiting events that are analyzed for each reload are determined by a sensitivity study described in GESTAR that examined the impact on the Minimum Critical Power Ratio (MCPR) due to the changes in fuel design. Based on results of that study, several limiting events have been identified and are analyzed using the appropriate input parameters and boundary conditions for each cycle. The MCPR results of these limiting events (from the transient analyses) form the basis of the MCPR operating limits for each reload cycle. Implementation of these limits in the PNPP Technical Specifications, Specification 3.2.2 Minimum Critical Power Ratio, ensures that the MCPR Safety Limit will not be exceeded during the most severe anticipated operational occurrences.

The overpressure protection evaluation demonstrated that the ASME code limits and requirements are met for this reload cycle.

A review with respect to the design basis Loss-of-Coolant Accident (LOCA) demonstrates that the peak cladding temperature and maximum oxidation fraction for the reload fuel complies with the <10 CFR 50.46> ECCS performance acceptance criteria. The LOCA results form the basis of the average planar linear heat generation rates for each reload cycle. The applicable limits on average planar linear heat generation rates are identified in PNPP Technical Specifications, Specification 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR).

The analysis of the anticipated operational occurrences (transients) was conducted according to the methodologies described in GESTAR (Reference 15B-4). Limiting events that are analyzed for each reload are determined by a sensitivity study described in GESTAR that examined the impact on the fuel thermal mechanical limits due to the changes in fuel design. Based on results of that study, several limiting events have been identified and are analyzed using the appropriate input parameters and boundary conditions for each cycle. The fuel thermal mechanical results of these limiting events (from the transient analyses) form the basis of the LHGR operating limits for each reload cycle. The applicable limits on linear heat generation rates are identified in the PNPP Technical Specifications, Specification 3.2.3, Linear Heat Generation Rate (LHGR).

15B.4 CHAPTER 4 - REACTOR

This chapter provides information on the design of the reactor including the fuel, materials, and reactivity controls. The fuel design description is mostly contained in GESTAR (Reference 15B-4). Cycle specific information for the current reload cycle is provided below.

15B.4.3 NUCLEAR DESIGN

The fuel bundle types installed in the core for this cycle are of the same basic design as other approved designs in GESTAR (Reference 15B-4). <Table 15B.4.3-1> describes the important fuel bundle information for these fuel types and provides the enrichment, number of bundles and cycle loaded for each of the fuel types. This basic information comes from the Supplemental Reload Licensing Report for the current cycle. Additional information on NRC approved GE fuel designs is presented in a companion report to GESTAR, entitled GE Fuel Bundle Designs (Reference 15B-7). For new fuel types not yet included in this report a proprietary supplement to the current cycle Supplemental Reload Licensing Report provides the new fuel bundle design information.

The safety and plant performance design bases for the nuclear design are described in GESTAR (Reference 15B-4). The current cycle core loading is composed of several fuel types each of a unique rod by rod enrichment distribution. The nuclear and thermal hydraulics characteristics of the fuel bundles are simulated in a GE lattice computer model and 3-D simulator (Reference 15B-5) for the development of the current cycle loading pattern. The loading pattern considers the integrated effect of mixing the new bundles with the irradiated fuel bundles in the core. The objective of this loading pattern is to optimize the fuel burnup efficiency. This consideration includes meeting predetermined target radial and axial power distribution, thermal limits and fuel cycle exposures. The unique bundle types are distributed in the core based on the principle of minimizing radial power peaking and maximizing core reactivity. The reference loading pattern and its target cycle exposure are provided in the Supplemental Reload Licensing Report for the current cycle. The reference loading pattern is also shown in <Figure 15B.4.3-1>.

Because the reload licensing process requires an assumption as to the condition of the core at the end of the previous cycle, it is possible that the as-loaded core may not be identical to the reference core. The as-loaded core is compared to the reference core to assure that licensing calculations performed on the reference core are applicable. Certain key parameters, which affect the licensing calculations, are reviewed to assure that there is no adverse impact.

The safety aspect of the thermal-mechanical performance for the fuel during normal steady-state operation has been evaluated and documented in depth in Section 2.0, Fuel Mechanical Design of GESTAR (Reference 15B-4). Other nuclear and thermal hydraulic parameters that directly affect the safety performance of the plant for steady-state, transient and accident condition have been evaluated and documented in Section 3.0, Nuclear Design of GESTAR (Reference 15B-4). These evaluations are directly applicable to the reload fuel types and core loading patterns for this reload cycle.

A shutdown margin calculation is performed using the 3-D simulator (Reference 15B-5) to demonstrate that the core is capable of being made subcritical with sufficient margin in the most reactive condition throughout the cycle with the strongest control rod withdrawn. The calculated core effective neutron multiplication factor (k_{eff}) and the variation throughout the cycle (Δk) information for the current reload reference core loading pattern including the resultant shutdown margin at the beginning of the cycle and the minimum margin mid-cycle condition are provided in the Supplemental Reload Licensing Report and are presented in <Table 15B.4.3-2>. These results represent significant margin to the minimum 0.38% $\Delta k/k$ Technical Specification shutdown margin requirement.

TABLE 15B.4.3-1

CURRENT CYCLE
FUEL BUNDLE INFORMATION
BASED ON THE REFERENCE CORE LOADING PATTERN

Fuel Type	Cycle Loaded	Number
Irradiated:		
GNF2-P10SG2B394-8G7.0/5G6.0-120T2-150-T6-4467	17	76
GNF2-P10SG2B397-14GZ-120T2-150-T6-4468	17	12
GNF2-P10SG2B397-14GZ-120T2-150-T6-4469	17	16
GNF2-P10SG2B418-13GZ-120T2-150-T6-4470	17	40
GNF2-P10SG2B420-4G7.0/8G6.0-120T2-150-T6-4471	17	16
GNF2-P10SG2B402-14GZ-120T2-150-T6-4619	18	96
GNF2-P10SG2B418-12G6.0-120T2-150-T6-4620	18	48
GNF2-P10SG2B402-6G7.0/7G6.0-120T2-150-T6-4621	18	88
GNF2-P10SG2B418-6G7.0/6G6.0-120T2-150-T6-4622	18	36
GNF2-P10SG2B405-10G7.0/2G6.0-120T2-150-T6-4623	18	32
New		
GNF2-P10SG2B401-14GZ-120T2-150-T6-4774	19	80
GNF2-P10SG2B400-14GZ-120T2-150-T6-4775	19	32
GNF2-P10SG2B401-2G7.0/10G6.0-120T2-150-T6-4776	19	56
GNF2-P10SG2B402-13GZ-120T2-150-T6-4777	19	24
GNF2-P10SG2B416-13GZ-120T2-150-T6-4778	19	64
GNF2-P10SG2B416-12GZ-120T2-150-T6-4779	19	32
Total:		**748**

All bundles are of GE, S-lattice (BWR/6 lattice type), barrier design.

The results presented in this appendix are based on analysis with the reload core reference loading pattern. As described in <Appendix 15B.4.3> variations may be made for the actual core loading as long as certain licensing parameters still meet requirements.

TABLE 15B.4.3-2

CURRENT CYCLE CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL SYSTEM WORTH - NO VOIDS, 20°C BASED ON THE REFERENCE CORE LOADING PATTERN[2]

Beginning of Cycle, K-effective	
Uncontrolled	1.131
Fully Controlled	0.946
Strongest Control Rod Out[3]	0.988
R, Maximum Increase in Cold Core Reactivity with Exposure into Cycle	0.001
Cycle exposure at which R occurs	14,600 MWD/ST
Resultant Shutdown Margin[1]	1.07%

NOTES:

(1) Shutdown margin is relative to the Cycle 19 beginning of cycle cold critical target K-effective of 1.000.
(2) Assumes Cycle 18 obtains a minimum core average exposure of 32,172 MWD/ST.
(3) Most reactive condition: 100°C.

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	35	37	39	41	43	45	47	49	51	53	55	57	59
60										35	38	39	38	38	37	37	38	38	39	38	35									
58								35	38	35	35	35	38	11	13	13	11	38	35	35	35	38	35							
56							35	35	35	12	10	10	12	12	13	13	12	12	10	10	12	35	35	35						
54						35	37	39	10	12	45	44	45	44	44	44	44	45	44	45	12	10	39	37	35					
52					36	36	13	10	44	44	44	41	41	44	9	9	44	41	41	44	44	44	10	12	36	36				
50				35	36	35	10	45	45	13	42	9	41	9	41	41	9	41	9	42	13	45	45	10	35	36	35			
48			35	37	12	10	11	44	10	40	9	40	9	40	9	9	40	9	40	9	40	10	44	11	10	9	37	35		
46		35	35	39	10	45	44	13	40	9	42	11	40	11	42	42	11	40	11	42	9	40	13	44	45	10	39	35	35	
44		38	35	10	44	45	10	40	13	42	11	40	11	40	11	11	40	11	40	11	42	13	40	10	45	44	10	35	38	
42	35	35	12	12	44	13	40	9	42	11	42	9	42	9	40	40	9	42	9	42	11	42	9	40	13	44	12	12	35	35
40	38	35	10	45	44	42	9	42	11	42	11	43	11	43	11	11	43	11	43	11	42	11	42	9	42	44	45	10	35	38
38	39	35	10	44	41	9	40	11	40	9	43	9	42	9	40	40	9	42	9	43	9	40	11	40	9	41	44	10	35	39
36	38	38	12	45	41	41	9	40	11	42	11	42	9	40	11	11	40	9	42	11	42	11	40	9	41	41	45	12	38	38
34	38	11	12	44	44	9	40	11	40	9	43	9	40	11	43	43	11	40	9	43	9	40	11	40	9	44	44	12	11	38
32	37	13	13	44	9	41	9	42	11	40	11	40	11	43	9	9	43	11	40	11	40	11	42	9	41	9	44	13	13	37
30	37	9	13	44	9	41	9	42	11	40	11	40	11	43	9	9	43	11	40	11	40	11	42	9	41	9	44	13	13	37
28	38	11	12	44	44	9	40	11	40	9	43	9	40	11	43	43	11	40	9	43	9	40	11	40	9	44	44	12	11	38
26	38	38	12	45	41	41	9	40	11	42	11	42	9	40	11	11	40	9	42	11	42	11	40	9	41	41	45	12	38	38
24	39	35	10	44	41	9	40	11	40	9	43	9	42	9	40	40	9	42	9	43	9	40	11	40	9	41	44	10	35	39
22	38	35	10	45	44	42	9	42	11	42	11	43	11	43	11	11	43	11	43	11	42	11	42	9	42	44	45	10	35	38
20	35	35	12	12	44	13	40	9	42	11	42	9	42	9	40	40	9	42	9	42	11	42	9	40	13	44	12	12	35	35
18		38	35	10	44	45	10	40	13	42	11	40	11	40	11	11	40	11	40	11	42	13	40	10	45	44	10	35	38	
16		35	35	39	10	45	44	13	40	9	42	11	40	11	42	42	11	40	11	42	9	40	13	44	45	10	39	35	35	
14			35	37	12	10	11	44	10	40	9	40	9	40	9	9	40	9	40	9	40	10	44	11	10	12	37	35		
12				35	36	35	10	45	45	13	42	9	41	9	41	41	9	41	9	42	13	45	45	10	35	36	35			
10					36	36	9	10	44	44	44	41	41	44	9	9	44	41	41	44	44	44	10	9	36	36				
8						35	37	39	10	12	45	44	45	44	44	44	44	45	44	45	12	10	39	37	35					
6							35	35	35	12	10	10	12	12	13	13	12	12	10	10	12	35	35	35						
4								35	38	35	35	35	38	11	13	13	11	38	35	35	35	38	35							
2										35	38	39	38	38	37	37	38	38	39	38	35									

Fuel Type

9 = GNF2-P10SG2B402-14GZ-120T2-150-T6-4619 (GNF2)	(C18)	38 = GNF2-P10SG2B418-13GZ-120T2-150-T6-4470 (GNF2)	(C17)
10 = GNF2-P10SG2B418-12G6.0-120T2-150-T6-4620 (GNF2)	(C18)	39 = GNF2-P10SG2B420-4G7.0/8G6.0-120T2-150-T6-4471 (GNF2)	(C17)
11 = GNF2-P10SG2B402-6G7.0/7G6.0-120T2-150-T6-4621 (GNF2)	(C18)	40 = GNF2-P10SG2B401-14GZ-120T2-150-T6-4774 (GNF2)	(C19)
12 = GNF2-P10SG2B418-6G7.0/6G6.0-120T2-150-T6-4622 (GNF2)	(C18)	41 = GNF2-P10SG2B400-14GZ-120T2-150-T6-4775 (GNF2)	(C19)
13 = GNF2-P10SG2B405-10G7.0/2G6.0-120T2-150-T6-4623 (GNF2)	(C18)	42 = GNF2-P10SG2B401-2G7.0/10G6.0-120T2-150-T6-4776 (GNF2)	(C19)
35 = GNF2-P10SG2B394-8G7.0/5G6.0-120T2-150-T6-4467 (GNF2)	(C17)	43 = GNF2-P10SG2B402-13GZ-120T2-150-T6-4777 (GNF2)	(C19)
36 = GNF2-P10SG2B397-14GZ-120T2-150-T6-4468 (GNF2)	(C17)	44 = GNF2-P10SG2B416-13GZ-120T2-150-T6-4778 (GNF2)	(C19)
37 = GNF2-P10SG2B397-14GZ-120T2-150-T6-4469 (GNF2)	(C17)	45 = GNF2-P10SG2B416-12GZ-120T2-150-T6-4779 (GNF2)	(C19)

Reference
Core Loading Pattern

Figure 15B.4.3-1

15B.4.4 THERMAL AND HYDRAULIC DESIGN

The design basis for the thermal-hydraulic design is described in GESTAR (Reference 15B-4). The detailed input of the thermal-hydraulic design parameters used in the current cycle reload safety analysis is described in <Appendix 15B.15> (Accident Analysis). The current cycle reload analysis was performed consistent with the operating region referred to as the Maximum Extended Operating Domain (MEOD) described in <Appendix 15E>. Operation at rated 100% power (consistent with GEMINI methods) has been analyzed at 81% core flow and 105% core flow. Operation above the rated power rod line is analyzed up to the MEOD Boundary Line as defined in <USAR 15E.2>.

The stability compliance of GE fuel designs contained in GESTAR (Reference 15B-4) has been demonstrated on a generic basis independent of plant and cycle parameters. The NRC has reviewed and approved the methodology. Therefore, no cycle specific decay ratio calculations are required. NRC approval also includes those expanded operating modes analyzed for the reload cycle.

The conditions for thermal hydraulic stability acceptance by the NRC includes that the plant has the Oscillation Power Range Monitor (OPRM) system, procedures and Technical Specifications. The OPRM system is provided to detect the evidence of thermal-hydraulic instability and respond appropriately (e.g., by scramming of control rods if specified conditions are met). The OPRM system is described in <Section 7.2.1> of the USAR.

OPRM's generate a scram signal in an instability event in order to preclude the fuel from exceeding the Safety Limit MCPR.

Cycle specific setpoints for the OPRM's were determined for Cycle 19 (Reference 15B-31). The Period Based Detection Algorithm Amplitude Setpoint, Sp, and the Confirmation Count Setpoint, N2, are documented in the Core Operating Limits Report. The Period Based Detection Algorithm Amplitude Setpoint, Sp, is set such that the fuel SL MCPR is not exceeded during an oscillation event. The development of the amplitude setpoint, SP, and the confirmation count setpoint, N2, is described in NEDE-33766P-A, Revision 1, "GEH Simplified Stability Solution (GS3)," March 2015 (Reference 15B-51).

Should the OPRM's become inoperable, Technical Specifications require that alternate methods be initiated to detect and suppress thermal hydraulic instability oscillations. For this alternate method of detect and suppress, the original guidance was provided in BWR Owner's Group Letter BWROG-94078, "Guidelines for Stability Interim Corrective Action", June 6, 1994 (ICA's – Reference 15B-35).

The ICA's at Perry have since been updated per the guidance contained in BWR Owner's Group Letter OG 02-0119-260; July 17, 2002; Subject: "Backup Stability Protection (BSP) for Inoperable Option III Solution" (Reference 15B-36). The BSP alternate method of detect and suppress establishes regions on the power to flow map for restricted plant operations and necessary procedural guidance for operations near or in those regions. In accordance with the restricted areas of the power to flow map and the procedural guidance, operator action is taken before thermal hydraulic instability oscillations can develop to the point where the Safety Limit MCPR is challenged.

As part of the ICA methodology, restricted areas of the power to flow map were developed; a Manual Scram Required Region and a Controlled Entry Region, . These regions were generic in nature (flow and rod line boundaries) and were applied across the BWR fleet. As such there was no specific defined margin of protection for individual reactor sites.

As fuel and core designs advanced (10 by 10 fuel matrixes, 2 year cycles, power uprates) the assumed level of protection provided by the ICA’s began to decrease (Reference 15B-36). The BWROG developed Backup Stability Protection, BSP’s, as a means to restore the level of protection provided by the ICA’s. The BSP method requires a plant specific/cycle specific stability analysis be performed to establish the areas of restricted plant operations on the plant’s power to flow map. This analysis was completed to support operations in Cycle 10 (Reference 15B-37) and (Reference 15B-38). As reload fuel and core designs vary from cycle to cycle, it may be necessary to either adjust the procedural guidance or adjust the restricted regions of the power to flow map. Reload fuel and core designs will be evaluated to verify applicability of the Cycle 10 analysis, or the analysis will be updated as required. This analysis used approved GESTAR II methods and analysis tools for stability/decay ratio calculations. The ICA exclusion regions are combined with the BSP exclusion regions to develop the bounding exclusion regions using a plant specific/cycle specific stability analysis, the level of protection is restored.

Two BSP analyses were provided – one for nominal feedwater temperatures (425.5°F) over the cycle and one for reduced feedwater temperatures (325.5°F) over the cycle. The BSP feedwater temperature reduction to 325.5°F is within the temperature reduction analyzed in <USAR Chapter 15, Appendix D, Partial Feedwater Heating Operation Analysis>. The USAR 15D analysis also allows for full power operations following the end of cycle (all rods out) with final feedwater temperatures down to 255.5°F. If end of cycle full power operations with final feedwater temperatures below 325.5°F were to occur, the OPRM's would have to be operable. If the OPRM's were inoperable, additional analysis would be required to allow full power operations below 325.5°F.

15B.5 CHAPTER 5 - REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

This chapter provides information on the reactor coolant system and pressure containing components that define the reactor coolant pressure boundary. Reload related changes include re-performing the overpressure protection analysis each cycle to verify adequate protection. Also, the associated safety/relief valve setpoint and valve capacity assumed in the analysis may be changed.

15B.5.2.2 Overpressurization Protection

The overpressure protection system must accommodate the most severe pressurization transients so that the ASME code limit of 1375 psig is not exceeded. The Main Steam Isolation Valve (MSIV) closure with secondary scram (flux scram) has been determined to be the most limiting event for overpressure protection. This transient is analyzed for each reload cycle using models described in GESTAR (Reference 15B-4). The analysis is performed at the 102% power and 105% core flow condition to bound the reload cycle operating conditions and to account for power level uncertainties as specified in <Regulatory Guide 1.49>, "Power Levels of Nuclear Power Plants." Normal feedwater heating is assumed in the analysis because partial feedwater heating results in lower peak pressures during the postulated disturbance due to the lower initial pressure.

The overpressurization analysis (MSIV Closure with Flux Scram Event) assumes six (6) safety/relief valves (SRV) are out-of-service out of the nineteen (19) available safety/relief valves. The thirteen (13) safety valves with the highest setpoints were assumed to be operational. Operating conditions for this cycle assumed for this transient are presented in <Table 15B.5.2-1>.

The sequence of events for this transient for the current cycle is presented in <Table 15B.5.2-2>. The results of the analysis are provided in the current cycle Supplemental Reload Licensing Report as well as <Table 15B.5.2-3> and show that the peak steamline and vessel bottom head pressures are well below the ASME Code limit of 1375 psig. The transient response is shown in <Figure 15B.5.2-1>.

TABLE 15B.5.2-1

KEY INPUT PARAMETERS AND INITIAL CONDITIONS FOR THE CURRENT RELOAD CYCLE OVERPRESSURE PROTECTION ANALYSIS[2]

1.	Thermal Power Level, MWt	3,833
2.	Steam Flow, lbs per hr	16.67×10^6
3.	Core Flow, lbs per hr	109.2×10^6
4.	Feedwater Flow Rate, lbs per hr	16.67×10^6
5.	Feedwater Temperature, °F	427.5
6.	Vessel Dome Pressure, psig	1045
7.	Vessel Core Pressure, psig	1054.3
8.	Turbine Inlet Pressure, psig	985.9
9.	Nuclear Characteristics	EOC
10.	Setpoints for Safety/Relief Valves Safety Function, psig[1]	1200, 1216, 1226

NOTES:

(1) The 13 SRVs with the highest safety function setpoints are assumed to be operational. Includes 3% setpoint drift.
(2) Analysis is performed at 102% of rated thermal power.

TABLE 15B.5.2-2

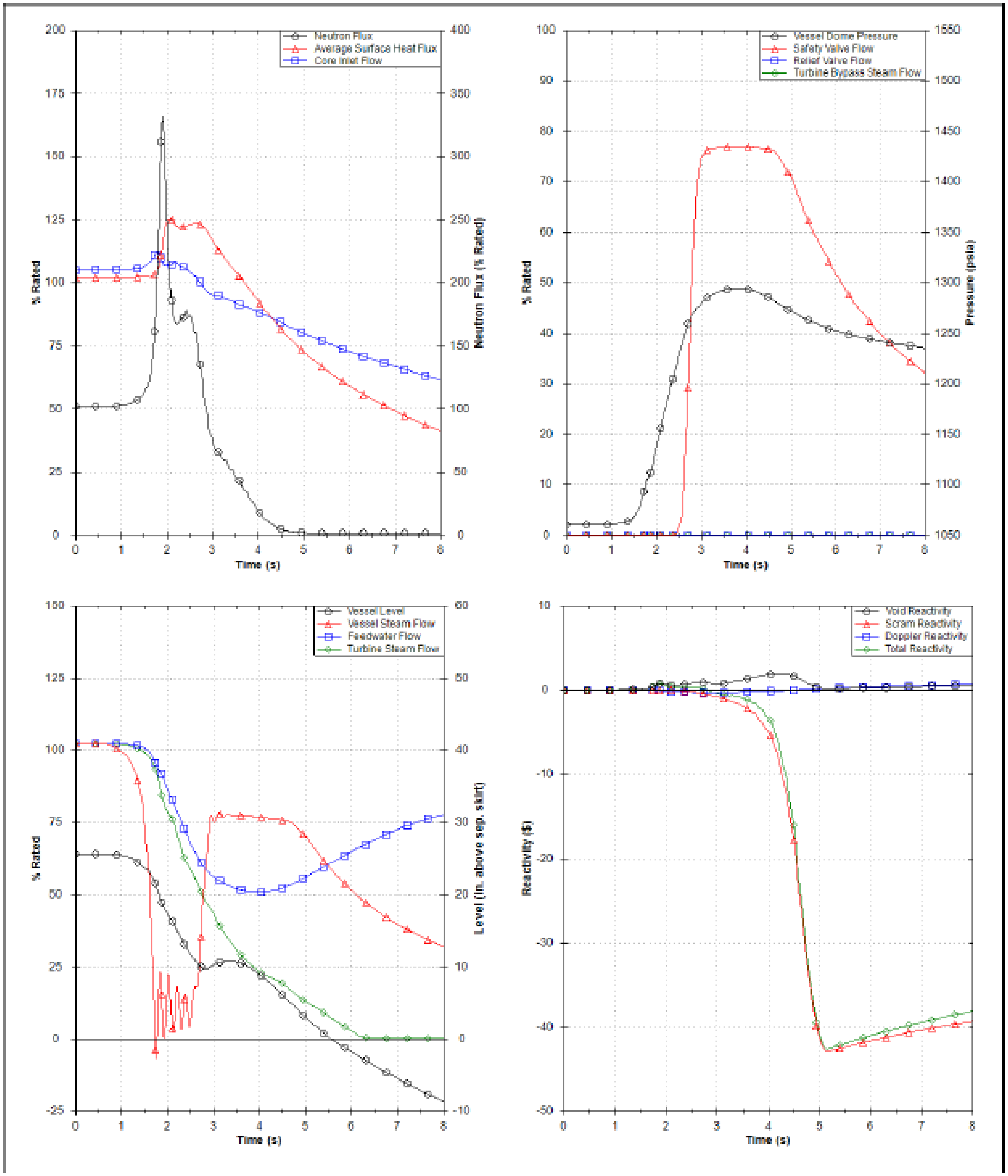
SEQUENCE OF EVENTS FOR THE MSIV CLOSURE WITH FLUX SCRAM (OVERPRESSURE PROTECTION) EVENT

Time-Sec	Events
0	Closure of all main steam isolation valves (MSIV) was initiated.
0.6	MSIVs start closing. Failure of direct position scram was assumed.
1.76	Neutron flux reached the high APRM flux scram setpoint and initiated reactor scram.
1.8	Reactor dome pressure reached the setpoint of recirculation pump trip.
2.1	Recirculation pump initiated to coastdown (includes 0.3 second RPT response time).
2.4	Steamline pressure reached the Group 1 safety/relief valves pressure setpoint (spring-action mode). Group 1 safety/relief valves open.
2.5	Group 2 safety/relief valves setpoint reached.
2.6	Group 3 safety/relief valves setpoint reached.
3.0	MSIVs completely closed.
3.7	Vessel bottom pressure reached its peak value.
>10 (est)	Safety/relief valves opened in their spring-action mode closed. (Not simulated in analysis)
>20 (est)	Wide-range sensed water level reached L2 setpoint. HPCS and RCIC flow entered reactor vessel. Safety valves close and reopen cyclicly. (Not simulated in analysis)

TABLE 15B.5.2-3

OVERPRESSURIZATION PROTECTION ANALYSIS RESULTS

Peak Steamline Pressure	1275 psig
Peak Vessel Bottom Pressure	1306 psig
ASME Code Limit	1375 psig



Plant Response to MSIV Closure (Flux Scram) - ICF (HBB)

Figure 15B.5.2-1

15B.6 CHAPTER 6 - ENGINEERED SAFETY FEATURES

15B.6.3.3 Performance Evaluation

This chapter provides information on the engineered safety feature (ESF) systems and components that are designed to ensure that the effects of a loss-of-coolant accident (LOCA) are mitigated and the radioactivity release from these accidents are limited. The emergency core cooling system (ECCS) performance analyses are re-examined for each reload to demonstrate conformance with the <10 CFR 50.46> ECCS performance criteria.

The SAFER/PRIME-LOCA methodology is used to demonstrate conformance with <10 CFR 50.46> <Section 6.3.3>.

With the introduction of GNF2 for Cycle 16, Perry’s LOCA analysis was updated. A specific GNF2 LOCA analysis was performed using the SAFER/PRIME methodologies (Reference 15B-48) and (Reference 15B-49). Additionally, in establishing the GNF2 analysis basis, various parameters were changed. It should be noted that Perry’s licensing basis are those parameters listed in <Chapter 6> and contained in Perry’s Technical Specifications. The GNF2 analysis basis is more conservative in that the calculated Peak Cladding Temperature is higher than what would have been calculated if the licensing basis parameters would have been used.

Based on the previous analysis work, only the limiting cases for GNF2 were analyzed. The most limiting case for GNF2 was the large break LOCA concurrent with High Pressure Core Spray Diesel Generator Failure at full power, maximum extended load line analysis condition of 81% core flow, and final feedwater temperature reduction. A range of small break LOCA’s were performed to verify that small break LOCA’s were not limiting for GNF2.

For Cycle 19, the reactor core consists of GNF2 fuel.

The analysis results for GNF2 (and for Cycle 19):

Parameter	Analysis Results (DBA Break)	Analysis Limit
1. Licensing PCT[3]	1610°F	</=2200°F[1]
2. Maximum Local Oxidation	< 1.0%	</= 17%[1]
3. Core-Wide Metal-Water Reaction	< 0.1%	</= 1.0%[1]
4. Upper Bound PCT	N/A	N/A[2]

<u>NOTES</u>:

(1) <10 CFR 50.46> ECCS-LOCA Analysis Acceptance Criteria

(2) Upper Bound PCT Limit not applicable for SAFER/PRIME methodology based on further analysis (Reference 47)

(3) Includes <10 CFR 50.46> methodology error and change reports

<10 CFR 50.46> methodology error or change reports are issued by the analysis supplier to identify errors or changes in the current record of analysis. These reports are reviewed for applicability and impact to Perry. The licensing basis and upper bound peak cladding temperatures are adjusted as necessary and reported in the table above. The applicable <10 CFR 50.46> methodology error or change reports are listed in (Reference 15B-25) and (Reference 15B-32).

The MAPLHGR limits as identified in GESTAR II are based on ECCS limits. MAPLHGR limits are recorded in the cycle specific Core Operating Limits

Report (COLR). The COLR is prepared and submitted to the NRC prior to the beginning of cycle startup in accordance with Technical Specification 5.6.

15B.9 CHAPTER 9 - AUXILIARY SYSTEM

This chapter provides information on the design and analysis of the plant auxiliary systems including the Fire Protection evaluation. The standby liquid control system (SLC) shutdown margin capability is re-examined for each cycle as part of the reload analysis.

The Standby Liquid Control (SLC) System is a reactivity control system to bring the reactor from rated power to a cold shutdown subcritical condition at any time when needed throughout the life of the core. The minimum average concentration of the natural boron required in the reactor core to accomplish meeting the minimum shutdown margin requirement has been determined to be 816 ppm (parts per million) at 68°F (20°C). A shutdown margin calculation (with boron injection) is performed for each reload cycle. For this cycle a 3.0% Δk SLCS shutdown margin exists at 181 degrees C which satisfies the Technical Specification 3.1.7 Bases requirement of providing the capability of bringing the reactor, at anytime in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon-free state without taking credit for control rod movement. Results of this analysis are provided in the Supplemental Reload Licensing Report for the current cycle. Therefore, the SLCS shutdown margin capability is adequate for the present cycle.

15B.15 CHAPTER 15 - ACCIDENT ANALYSIS

This chapter provides the effects of anticipated operational occurrences and accidents on Unit 1 of the Perry Nuclear Power Plant. This section describes the transient and accident analyses performed for the current reload cycle.

15B.15.0 GENERAL

In order to establish the safety analysis basis for each reload cycle, several limiting anticipated operating occurrences (transients) are re-analyzed. The limiting transient events and the methodology involved in their selection is described in GESTAR (Reference 15B-4) and summarized in <Section 15.0.6>, Reload Safety Analysis in <Chapter 15>.

The following are the limiting events that establish the licensing basis for the current cycle reload safety analysis.

1. Loss of Feedwater Heating (100°F).

2. Feedwater Controller Failure - Maximum Demand (143%).

3. Turbine Trip without Bypass.

4. Pressure Regulator Downscale Failure.

5. Rod Withdrawal Error at Power.

In addition, the mislocated bundle accident and the misoriented bundle accident are evaluated on a cycle specific basis.

The Fuel Handling Accident was re-verified to ensure the original analysis for 8x8 fuel bounds the 9x9 and 10x10 fuel.

The transient analyses utilizes the reload core reference loading pattern and covers the MEOD power-flow operating map as described in <Appendix 15B.4.4>. The analysis also covers all extended operating domains and modes of operation (i.e. partial feedwater heating, maximum extended operating domain and single-loop operation modes of operation as described in <Appendix 15D>, <Appendix 15E> and <Appendix 15F>).

The turbine trip, feedwater controller failure and pressure regulator downscale failure events were analyzed with the GE one-dimensional transient response ODYN model (Reference 15B-6) and the 100°F loss of feedwater heating and control rod withdrawal error event were analyzed using the 3-D BWR core simulator model (Reference 15B-5).

The transient analyses for the current reload cycle employ some methodologies and boundary conditions that are different from the original initial cycle design analysis. The most significant methodology differences are the application of the critical power correlation and the GEMINI method of analysis. Both of these methods have been approved for application in GESTAR (Reference 15B-4).

The critical power correlation is confirmed every cycle. When there is a change in fuel design parameters, such as fuel and water rod diameter, channel sizing and spacer design, the critical power correlation (GEXL) is also confirmed. If the GEXL correlation cannot be confirmed then a new critical power correlation is established. A description of the critical power correlation and its application is documented in GESTAR (Reference 15B-4).

The GEMINI methods replace the previous ODYN GENESIS set of methods in which a 1.044 adjustment factor was applied to the transient MCPR results. With the GEMINI methods, the MCPR response for each event is determined using statistically determined scram times and event-unique adders are applied to adjust for Technical Specification scram times and other uncertainties and conservatism to develop the operating limit MCPR values from the analytically determined MCPR responses. The GEMINI methods also only require analysis to be performed at 100% power since the event uncertainties are handled by the event-unique adders discussed above. A description of the GEMINI methods and their application are documented in GESTAR (Reference 15B-4).

In addition to the critical power correlation and GEMINI methods, other significant input parameters that are different than the original analysis include assuming the two (2) lowest setpoint safety/relief valves to be out-of-service in the transient analyses. The Safety Limit MCPR was increased from 1.06 for the initial cycle to 1.07 for reloads 1 through 4 (Reference 15B-3) for two recirculation loop operation and 1.08 for single recirculation loop operation.

For subsequent reloads, a cycle-specific Safety Limit MCPR calculation is performed. The cycle-specific SLMCPR was determined using the analysis basis documented in GESTAR with the following exceptions:

1. The reference core loading pattern in <Figure 15B.4.3-1> was analyzed.

2. The actual bundle parameters (e.g., local peaking) were used.

3. The full cycle exposure range was analyzed (Reference 15B-25).

The safety limit MCPR 99.9% for the current reload cycle is given in <Table 15B.15.0-1>.

The following sections provide a detailed description of the transients that were analyzed for this reload following the format i.e., <Regulatory Guide 1.70> in the main text of the USAR. The five (5) transients and two (2) accidents analyzed for this reload represent the most limiting bounding events covering all categories. For those transients that were analyzed in the initial cycle that have been determined to be less limiting and bounded by the re-analyzed transients, detailed descriptions are not provided.

The analytical objectives and criteria for these reload transient analyses remain the same as that described in <Section 15.0> of the main USAR text. The input parameters and initial conditions for the reload analysis are specified in <Table 15B.15.0-1>. Analyses that assume data inputs different from these values are designated accordingly in the appropriate event description. The analysis covers the MEOD power-flow operating map shown in <Figure 15E.2-1>. The analysis initial conditions and results presented in this section are taken from the Supplemental Reload Licensing Report (Reference 15B-25) for the current cycle. A summary of the reload transient results (peak neutron flux, heat flux and MCPR responses) are presented in <Table 15B.15.0-2>. The transient responses for these events are presented in the respective figures in the following sections. The operating limit MCPR values for the current reload cycle resulting from the analysis results are given in <Table 15B.15.0-3>.

TABLE 15B.15.0-1

INPUT PARAMETERS AND INITIAL CONDITIONS
FOR RELOAD TRANSIENT ANALYSIS
AT BOUNDING CONDITIONS FOR EACH TRANSIENT

		At 255.5°F[1] Feedwater Temperature	At 425.5°F[2][5] Feedwater Temperature
1.	Thermal Power Level, MWt[3]	3,758	SAME
2.	Steam Flow, lbs per hr[3]	13.30×10^6	16.31×10^6
3.	Core Flow, lbs per hr[3]	109.2×10^6	SAME
4.	Feedwater Flow Rate, lbs per hr[3]	13.29×10^6	16.30×10^6
5.	Feedwater Temperature, °F[3]	255.5	425.5
6.	Vessel Dome Pressure, psig[3]	994.3	1025.0
7.	Vessel Core Pressure, psig[3]	1008.8	1041.3
8.	Turbine Bypass Capacity, % NBR	23.63	SAME
9.	Core Coolant Inlet Enthalpy, Btu per lb[3]	506.9	528.5
10.	Turbine Inlet Pressure, psig	948.8	957.9
11.	Fuel Lattice[3]	<Table 15B.4.3-1>	SAME
12.	Core Leakage Flow, %[3]	14.38	15.47
13.	Required MCPR Operating Limit[3]	<Table 15B.15.0-3>	SAME
14.	Safety Limit MCPR 99.9%[7]	1.07	SAME
15.	Jet Pump Ratio, M	2.35	2.30
16.	Safety/Relief Valve Capacity, lb/hr 1080 psig	840,448	SAME
17.	Number of Valves Installed	19	SAME
18.	Number of Valves Assumed in Transient Analyses	17	SAME
19.	Relief Function Delay, seconds	0.4	SAME

TABLE 15B.15.0-1 (Continued)

		At 255.5°F[1] Feedwater Temperature	At 425.5°F[2][5] Feedwater Temperature
20.	Relief Function Stroke Time, seconds	0.15	SAME
21.	Relief Function Lowest Setpoint, psig[6] Analytical Limit	1133	SAME
22.	Safety Function Delay, seconds	0.0	SAME
23.	Safety Function Stroke Time, seconds	0.3	SAME
24.	Safety Function Lowest Setpoint, psig[4] Allowable Value	1200	SAME
25.	High Flux Trip, % NBR Analytical Limit	122	SAME
26.	High Pressure Scram Setpoint, psig Analytical Limit	1,095	SAME
27.	Vessel Level Trips, Above Vessel Zero		
	Level 8 - (L8), inches	585.35	SAME
	Level 4 - (L4), inches	560.60	SAME
	Level 3 - (L3), inches	539.10	SAME
	Level 2 - (L2), inches	485.40	SAME
28.	APRM Simulated Thermal Power Trip Scram, % NBR, Analytical Limit	115	SAME
29.	RPT Delay, seconds	0.14	SAME
30.	Total Steamline Volume, ft^3	4,681	SAME

NOTES:

(1) The Feedwater Controller Failure-Maximum Demand is most limiting at 255.5°F feedwater temperature.

(2) The Turbine Trip without Bypass is most limiting at 425.5°F feedwater temperature.

(3) The 3-D BWR Simulator was used for the Loss of Feedwater Heating transient. Only these parameters are used as inputs.

(4) Includes 3% setpoint drift.

TABLE 15B.15.0-1 (Continued)

NOTES: (continued)

(5) Transients were not run for the intermediate feedwater temperatures (320°F and 370°F) because the operating limits would not improve for those conditions.

(6) Assumes the 2 lowest safety/relief valves are out-of-service in the transient analysis. Therefore, 1143 psig is the lowest setpoint used in the analysis (vice 1133 psig). For MSIV Closure (Flux Scram) overpressurization analysis, 6 safety valves are assumed out-of-service.

(7) Single loop safety limit MCPR is 1.13.

TABLE 15B.15.0-2

SUMMARY OF THE RELOAD TRANSIENT ANALYSIS

Paragraph	Figure	Transient Event	Cycle Core Average Exposure (MWD/ST)	Power/ Flow (% NBR)	Feed-Water Temp-erature (°F)	Maximum Neutron Flux (% NBR)	Maximum Core Average Heat Flux (% NBR)	Uncorrected Delta CPR[1][4] GNF2
<Section 15B.15.1.1.3.3>	-	100°F Loss of Feedwater Heating	See Note[2]	100/81	425.5	See Note[3]		-
<Section 15B.15.1.2.3.3>	<Figure 15B.15.1-1>	Feedwater Con-troller Failure-Maximum Demand (open to 143%)	See Note[2]	100/105	255.5	213.4	117.2	0.22
<Section 15B.15.2.3.3.3>	<Figure 15B.15.2-2>	Turbine Trip without Bypass	See Note[2]	100/105	425.5	232.2	107.8	0.24
<Section 15B.15.4.2>	-	Control Rod Withdrawal Error	See Note[2]	-	-	-	-	-
<Section 15B.15.4.7.1>[5]	-	Mislocated Bundle Accident	See Note[2]	-	-	-	-	-
<Section 15B.15.4.7.2>		Misoriented Bundle	See Note[2]	-	-	-	-	

NOTES:

(1) The analysis utilizes the GEMINI method of calculating delta CPR. The listed values are "uncorrected" since additional uncertainties (both event and fuel type) are applied to develop the required MCPR operating limits in <Table 15B.15.0-3>.
(2) The various transient events are evaluated throughout the entire cycle.
(3) Data is no longer available since ODYN evaluations are not performed for this transient. PANACEA simulations are used to evaluate MCPR response.
(4) Uncorrected delta CPR's are not reported for some transients.
(5) Mislocated bundle accident was not reanalyzed for Cycle 19 because Cycle 19 OLMCPR exceeded limits established in GESTAR II.

TABLE 15B.15.0-3

RELOAD TRANSIENT ANALYSES OPERATING LIMIT MCPR VALUES

Transient Event	Applicable Core Average Cycle Exposure Range (MWD/ST)	Applicable Flow at 100% Power %	Applicable Feedwater Temperature at 100% Power (°F)	OLMCPR[1][6] GNF2
100°F Loss of Feedwater Heating	BOC19 to EOC19	81-105	425.5-325.5[2]-255.5[3]	1.20
Feedwater Controller Failure - Maximum Demand (open to 143%)	BOC19 to EOC19	81-105	425.5-325.5[2]-255.5[3]	1.34
Turbine Trip Without Bypass	BOC19 to EOC19	81-105	425.5-325.5[2]-255.5[3]	1.37
Control Rod Withdrawal Error	BOC19 to EOC19	81-105	425.5-325.5[2]-255.5[3]	1.18
Mislocated Bundle Accident[5]	BOC19 to EOC19	81-105	425.5-325.5[2]-255.5[3]	Non-Limiting
Misoriented Bundle Accident	BOC19 to EOC19	81-105	425.5-325.5[2]-255.5[3]	1.24

NOTES:

(1) The required OLMCPR values for each event are generated utilizing the GE GEMINI method. Event unique adders are used to generate the required OLMCPR for each event. The highest event OLMCPR (at each feedwater temperature condition) determines the MCPR Operating Limit at that temperature and flow and/or the overall cycle MCPR Operating Limit.

(2) Lower feedwater temperature limit for operation during the operating cycle with feedwater heater(s) intentionally out-of-service.

(3) Lower feedwater temperature limit for operation with cycle extension with reduced feedwater heating.

(4) For single loop operations, the Single Loop Operating Limit MCPR is set equal to Two Loop Operating Limit MCPR plus 0.02. The 0.02 value bounds both the change in the Safety Limit MCPR 99.9% values from two Loop Operating to Single Loop Operating and the Cycle 19 GNF2 Single Loop Operating Recirc Pump Seizure Event.

(5) Mislocated bundle-accident was not recalculated for Cycle 19 because the Cycle 19 OLMCPR exceeded limits established in GESTAR II. The value listed in the table is the limit for which if the cycle specific Operating Limit MCPR were to decrease below a cycle specific mislocated bundle analysis would be required.

(6) For Cycle 19, the Operating Limit MCPR’s were determined based on fuel type and cycle exposure. The values limited above are for the most limiting point in the cycle. The following table identifies the appropriate Operating Limit MCPR between beginning of cycle through middle of the cycle and then from middle of cycle through to the end of cycle.

Two Loop Operating Limit MCPR is defined as

Two Loop OL – MCPR		GNF2
BOC to < MOC		1.31
>/= MOC to EOC		1.37

15B.15.1 DECREASE IN REACTOR COOLANT TEMPERATURE

Six transients were evaluated under the original design that are categorized under the decrease in reactor coolant temperature transient category. Two of these transients are re-analyzed to represent the licensing basis events under this transient category for the reload cycle. These are the Loss of Feedwater Heating and the Feedwater Controller Failure - Maximum Demand transients.

15B.15.1.1 Loss of Feedwater Heating

The loss of feedwater heating transient is identified in GESTAR to be one of the events most likely to limit operation because of MCPR considerations. As a result, this event is re-analyzed as a licensing basis transient for this reload cycle.

15B.15.1.1.1 Identification of Causes and Frequency Classification

15B.15.1.1.1.1 Identification of Causes

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

a. Steam extraction line to heater is closed

b. Feedwater or condensate is bypassed around heater.

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

Steady-state operation with partial feedwater heating is an operating mode that has been analyzed for PNPP in <Appendix 15D> and <Appendix 15E>. This operating mode could occur after a loss of feedwater heating that does not result in a scram after stabilizing the thermal power and restoring thermal limits to within the required Technical Specification limits for reduced temperature operation. Under this operating condition, another 100°F loss of feedwater heating (from this reduced feedwater temperature operating condition) is postulated to justify that this mode of operation is acceptable.

15B.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 at full power.

15B.15.1.1.2 Sequence of Events and Systems Operation

15B.15.1.1.2.1 Sequence of Events

<Table 15B.15.1-1> lists the sequence of events for the analyzed transient.

15B.15.1.1.2.1.1 Identification of Operator Actions

The transient analysis does not take credit for any operator action. If the transient occurs, the following operator actions will help reduce the severity and recover from the transient.

In the automatic flux/flow control mode, the reactor settles out at a lower recirculation flow with no change in steam output. The average power range monitor (APRM) neutron flux or thermal power alarm may sound to alert the operator that he should insert control rods to get back down to the rated flow control line, or that he should reduce flow if in the manual mode. Operating procedures describe turbine generator operation with feedwater heaters out-of-service. If reactor scram occurs, as it may in the manual flow control mode, the operator should monitor the reactor water level and pressure controls and the turbine generator auxiliaries during coastdown.

15B.15.1.1.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems. This event results in a slow prolonged power increase without flux spikes.

The APRM flow biased simulated thermal power monitor (TPM) is the primary protection system trip in mitigating the consequences of this event. For a 100°F design basis event, the thermal power increase is close to the high thermal power scram setpoint that may or may not result in a reactor scram. This scram is conservatively ignored in the licensing basis evaluation for this reload.

The TPM conservatively estimates thermal power by passing the APRM signal through a specified time constant. A scram is initiated when thermal power exceeds the flow-biased function shown in <Figure 15.1-5> of the main text. For a slow transient this limit will be reached before the APRM flux scram because of its 6 to 8 percent lower setpoint.

The specified time constant is a filtered signal representative of the fuel heat flux response. It is part of the instrumentation circuitry, not a core characteristic. The time constant approximates the fuel heat flux response. The adequacy of the time constant is demonstrated by the transient fuel margin and not by the precision with which it approximates the fuel dynamics. The specified time constant used in the reload safety analysis is 6±0.6 seconds.

The TPM is a safety grade system and is designed to be single failure proof. Surveillance testing of the TPM is included in the technical specifications.

Required operation of Engineered Safety Features (ESF) is not expected for either of these transients.

15B.15.1.1.2.3 The Effect of Single Failures and Operator Errors

These two events generally lead to an increase in reactor power level. The TPM mentioned in <Section 15B.15.1.1.2.2> is the mitigating system and is designed to be single failure proof. See <Appendix 15A> for additional discussion of this subject.

15B.15.1.1.3 Core and System Performance

15B.15.1.1.3.1 Mathematical Model

The quasi-steady-state nature of this transient enables this slow transient to be analyzed using the 3-dimensional, coupled nuclear-thermal hydraulics core simulator computer model. This mode is described in detail in (Reference 15B-5). This model calculates the changes in power level, power distribution, core flow, exposures, reactor thermal hydraulic characteristics and critical power ratio with spatially varying voids, control rods, burnable poisons and other variables under steady-state conditions. For this transient, the time for reactivity insertion is greater than the fuel thermal time constant and the core-hydraulic transport times. Therefore, the steady-state representation before and after the transient is adequate. This computer model has been qualified and approved by the NRC for application for this transient.

The 3-dimensional simulator model replaces the point kinetic model previously used to analyze this event for the initial cycle as described in <Section 15.1.1.3.1> of the main text.

15B.15.1.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in <Table 15B.15.0-1>. The 100% power, flow, normal feedwater heating and end-of-cycle analysis condition represents the bounding condition for this reload analysis. This condition also represents the lowest scram reactivity even though this analysis conservatively assumes no scram occurs during this 100°F loss. The transient is simulated by programming a change in feedwater enthalpy corresponding to a 100°F loss in feedwater heating. Only the manual modes of flow control are quantitatively analyzed.

15B.15.1.1.3.3 Results

In the automatic flux/flow control mode, the recirculation flow control system responds to the power increase by reducing core flow down the power flow control rod line thus causing a smaller power increase that causes this event to be less severe than the manual flow control case discussed below. Nuclear system pressure does not change; consequently, the reactor coolant pressure boundary is not threatened. If scram occurs, the results become very similar to the manual flow control case.

In the 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 thermal power may or may not occur for a 100°F loss event because it has been shown that the power increase for a 100°F loss event is slightly below the high thermal power scram setpoint. Vessel steam flow and the initial system pressure remains relatively constant or increases slightly. The MCPR remains at or above the Safety Limit. Therefore, the design basis is satisfied. Results of the analysis are provided in <Table 15B.15.0-2>. The operating limit MCPR for this transient is given in <Table 15B.15.0-3>. The sequence of events for this event is presented in <Table 15B.15.1-1>.

The analysis is performed at the 100% power, 105% core flow and normal feedwater heating condition and adequately bounds all power flow conditions <Appendix 15E> and the partial feedwater condition <Appendix 15D> because it has been shown that this is mainly a subcooling perturbation event that is not significantly affected by initial power, flow, and feedwater temperature conditions. It has been shown that lower initial feedwater temperature is bounded by the normal feedwater heating initial condition for this event. Furthermore, lower initial power levels will have initial CPR values that are greater than the analysis' assumed initial value. Therefore, this analysis is also applicable to single loop operating condition <Appendix 15F>.

15B.15.1.1.3.4 Considerations of Uncertainties

Important factors (such as reactivity and void coefficient exposure variation) have been accounted for in a conservative factor applied to the 3-D Simulator analysis results so that any deviations in actual plant operation are bounded by this analysis.

15B.15.1.1.4 Barrier Performance

As noted above the consequences of these events 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.

15B.15.1.1.5 Radiological Consequences

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

TABLE 15B.15.1-1

SEQUENCE OF EVENTS FOR THE LOSS OF
FEEDWATER HEATING EVENT
MANUAL FLOW CONTROL

Time-sec	Event
0	A 100°F temperature reduction in the feedwater system is initiated.
5 (est.)	Initial effect of colder feedwater starts to raise core power level gradually. (Not Simulated)
~200 (est.)	Core power settles into a higher level. The high APRM level may or may not initiate a reactor scram on high thermal power while core flow remains relatively constant. (Not Simulated)

15B.15.1.2 Feedwater Controller Failure - Maximum Demand

The feedwater controller failure - maximum demand transient is identified in GESTAR to be one of the events most likely to limit operation because of MCPR considerations. As a result, this event is re-analyzed as a licensing basis transient for this reload cycle.

15B.15.1.2.1 Identification of Causes and Frequency Classification

15B.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.

15B.15.1.2.1.2 Frequency Classification

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

15B.15.1.2.2 Sequence of Events and Systems Operation

15B.15.1.2.2.1 Sequence of Events

The increased feedwater flow adds additional subcooled water into the saturated coolant in the core. This increases the core power due to the additional reactivity from the reduction in void fraction. With excess feedwater flow the water level rises to the high level reference point at which time the feedwater pumps and the main turbine are tripped and a scram is initiated. <Table 15B.15.1-2> lists the sequence of events for <Figure 15.1-3>.

15B.15.1.2.2.1.1 Identification of Operator Actions

The transient analysis does not take credit for any operator action. If the transient occurs, the following actions will help to recover from the transient.

The operator should:

a. Observe that feedwater pump trip has terminated the failure event.

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

c. Identify causes of the failure and report all key plant parameters during the event.

15B.15.1.2.2.2 Systems Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. Important system operational actions for this event are high level tripping of the main turbine, turbine stop-valve scram trip initiation, feedwater pump trip, bypass valve opening, recirculation pump trip (RPT), 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 the feedwater pumps.

15B.15.1.2.2.3 The Effect of Single Failures and Operator Errors

In <Table 15B.15.1-2> 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.

15B.15.1.2.3 Core and System Performance

15B.15.1.2.3.1 Mathematical Model

The predicted dynamic behavior of this event has been determined by using a computer model of a generic direct-cycle BWR as described in (Reference 15B-6). This computer model has been improved and verified through extensive comparison of its predicted results with actual BWR test data.

This one-dimensional nonlinear computer 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 eight pressure nodes incorporating mass and momentum balances which will predict any wave phenomena present in the steam line during pressurization transients.

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, describes 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, pressure, and load demand are represented together with their dominant nonlinear characteristics.

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

15B.15.1.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in <Table 15B.15.0-1>.

The transient is simulated by programming an upper limit failure in the feedwater system so that 143 percent of nuclear boiler rated (NBR) feedwater flow occurs at the operating pressure of 1065 psig. An additional 5% higher flow rate conservatism is assumed in the analysis to allow for feedwater flow uncertainty. This value was determined during startup testing and replaces the 130% maximum flow used in the initial cycle analysis.

The analysis was performed at the 100% power, 105% core flow condition with a 250°F feedwater temperature partial feedwater heating condition (steady-state initial condition) with end-of-cycle exposure and all-rods-out scram characteristics. This initial condition represents the bounding condition for this reload analysis.

15B.15.1.2.3.3 Results

Sensitivity studies have been performed that provide the sensitivity for the licensing basis analysis for this event for this reload. The analysis was performed at the 100% power, 105% core flow and for the 250°F final feedwater temperature (partial feedwater heating) condition and represents the most limiting case for this cycle.

The simulated feedwater controller transient is shown in <Figure 15B.15.1-1>. The high water level turbine trip and feedwater pump trip are initiated. Scram occurs almost simultaneously, and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs.

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

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

The maximum neutron flux, maximum core average heat flux, and calculated delta CPR are provided in <Table 15B.15.0-2>. The operating limit MCPR for this transient is presented in <Table 15B.15.0-3>. MCPR remains above the safety limit for this event. Therefore, the design basis is satisfied.

The turbine bypass system opens to limit peak pressure. The peak pressure in the steamline near the safety/relief valves and the peak reactor coolant pressure at the bottom of the vessel are shown in <Table 15B.15.0-2> and are well below the ASME Code limit of 1375 psig.

This analysis adequately bounds all power/flow conditions <Appendix 15E> and partial feedwater heating conditions <Appendix 15D> because it has been shown that at the highest core flow (105%) and lowest feedwater temperature (250°F) represents the most limiting conditions for this transient event. The results of this analysis also ensure that the operating limits for single-loop operation <Appendix 15F> continue to be applicable.

15B.15.1.2.3.4 Consideration of Uncertainties

Important analytical factors (such as relief setpoints, scram stroke time and reactivity characteristics) have been adjusted statistically so that any deviation in the actual plant parameters will produce a less severe transient.

15B.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.

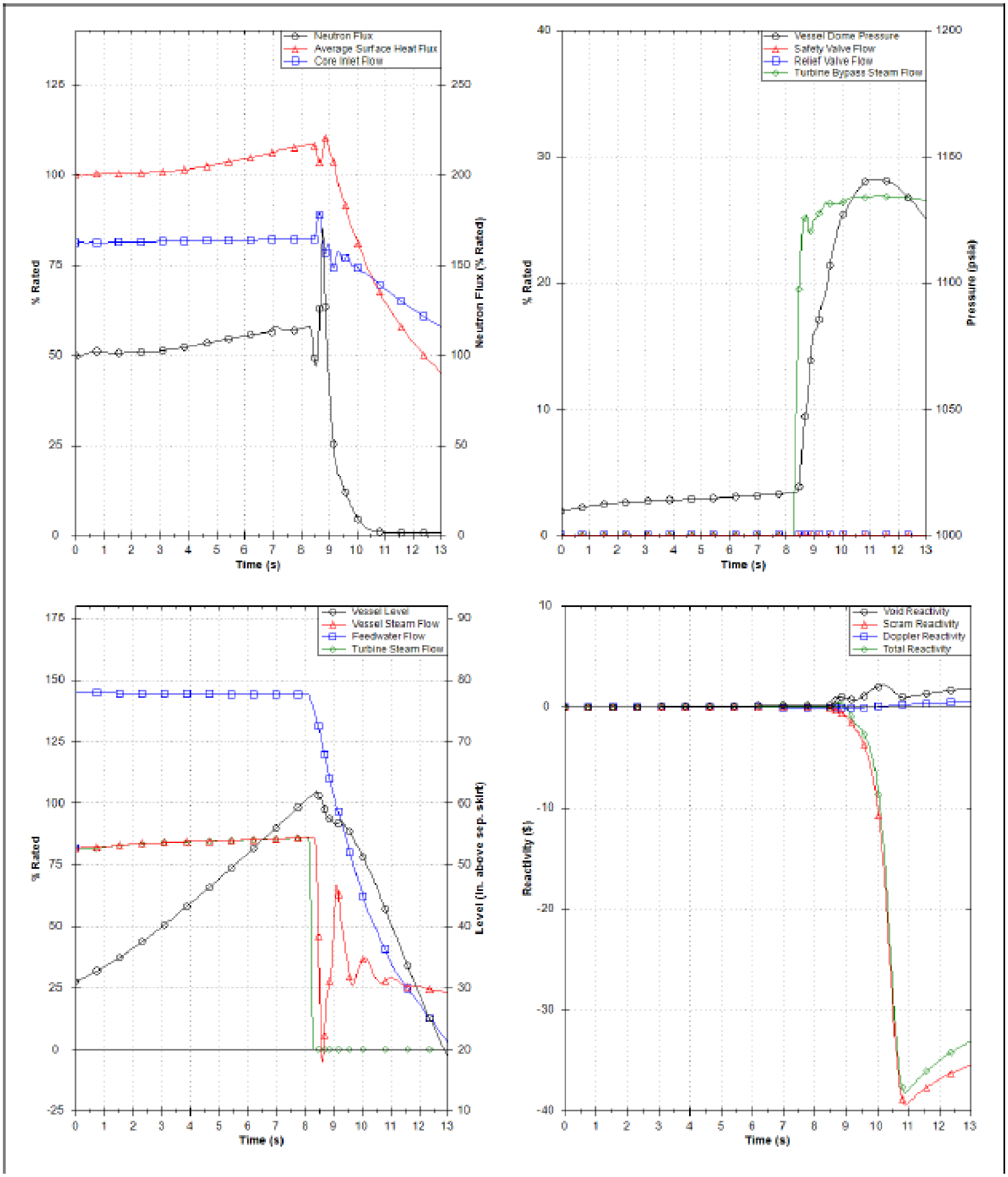
15B.15.1.2.5 Radiological Consequences

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 <Section 15.2.4.5>. Therefore, the radiological exposures noted in <Section 15.2.4.5> cover the consequences of this event.

TABLE 15B.15.1-2

SEQUENCE OF EVENTS FOR <FIGURE 15B.15.1-1>
FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND EVENT

Time-sec	Event
0	Initiate simulated failure to 153% upper limit at feedwater operating pressure of 996 psig on feedwater flow.
8	L8 vessel level set point initiates reactor scram and trips main turbine and feedwater pumps.
	Recirculation pump trip (RPT) initiated by stop valve position switches (does not include 0.14 second EOC-RPT response time).
	Main turbine bypass valves start to open due to turbine trip.
	No safety/relief valve actuation.
>50 (est.)	RCIC and HPCS flow into vessel (not simulated).



Plant Response to FW Controller Failure (EOC ICF & FWTR (UB))

Figure 15B.15.1-1

15B.15.2 INCREASE IN REACTOR PRESSURE

Ten transients were evaluated under the original design that are categorized under the increase in reactor pressure transient category. The two most limiting of these transients, the pressure regulator failure-closed and turbine trip (without bypass) events are re-analyzed to represent the licensing basis events under this transient category for the reload cycle.

15B.15.2.1 Pressure Regulator Failure - Closed

The pressure regulator downscale failure transient event is identified in GESTAR to be one of the events most likely to limit operation because of MCPR considerations for BWR/6 plants. This event is one of the scenarios considered under this transient.

For the initial core and power uprate, the Pressure Regulator Downscale Failure Event is calculated and the delta CPR results reported in <Chapter 15>. For reload cores, an evaluation is performed to determine if this event could potentially alter the cycle MCPR operating limit. If it does, the results are reported in the Supplemental Reload Licensing Report. For the current cycle, this evaluation concluded that this event is significantly less limiting than the other pressurization events in the reload analysis.

15B.15.2.2 Generator Load Rejection

The generator load rejection without bypass and turbine trip without bypass events are similar events categorized in this transient category. The more limiting of these two events has been identified in GESTAR to be one of the events most likely to limit operation because of MCPR considerations. Therefore, the limiting event is analyzed as a licensing basis transient for this reload cycle. It has been determined from both the original design <Appendix 15B.15.2.3> analysis and a sensitivity study of this reload analysis that the generator load rejection is not the most limiting event to be analyzed under this transient category.

15B.15.2.3 Turbine Trip

The generator load rejection without bypass and turbine trip without bypass events are similar events categorized in this transient category. The more limiting of these two events has been identified in GESTAR to be one of the events most likely to limit operation because of MCPR considerations. Therefore, the limiting event is analyzed as a licensing basis transient for this reload cycle. It has been determined from both the original design <Section 15.15.2.3> analysis and a sensitivity study of this reload analysis that the turbine trip without bypass is the most limiting event to be analyzed under this transient category.

15B.15.2.3.1 Identification of Causes and Frequency Classification

15B.15.2.3.1.1 Identification of Causes

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

15B.15.2.3.1.2 Frequency Classification

15B.15.2.3.1.2.1 Turbine Trip

This transient is categorized as an incident of moderate frequency. In defining the frequency of this event, turbine trips 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.

15B.15.2.3.1.2.2 Turbine Trip with Failure of the Bypass

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

Frequency:	0.0064/plant year
Mean time between events (MTBE):	156 years

Frequency Basis: As discussed in <Section 15.2.2.1.2.b>, the failure rate of the bypass is 0.0048. Combining this with the turbine trip frequency of 1.33 events/plant year yields the frequency of 0.0064/plant year.

However, PNPP is committed to compare the consequences of this transient with the allowable MCPR of a moderate frequency event.

15B.15.2.3.2 Sequence of Events and Systems Operation

15B.15.2.3.2.1 Sequence of Events

a. Turbine Trip

   Not analyzed for current cycle.

b. Turbine Trip with Failure of the Bypass

   Turbine trip at high power with bypass failure produces the sequence of events listed in <Table 15.B.15.2-2>.

15B.15.2.3.2.1.1 Identification of Operator Actions

The operator should:

a. Verify auto transfer of buses supplied by generator. If automatic transfer does not occur, manual transfer must be made.

b. Monitor and maintain reactor water level at required level.

c. Depending on conditions, initiate normal operating procedures for cooldown, or maintain pressure for restart purposes.

d. Put the mode switch in the shutdown position before the reactor pressure decays to <850 psig.

e. Secure the RCIC operation if reactor water level can be maintained above Level 2 without RCIC in operation.

f. Prevent reactor vessel water level from dropping to MSIV isolation signal (Level 1).

g. Monitor control rod drive positions and insert both the IRMs and SRMs.

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

15B.15.2.3.2.2 Systems Operation

a. Turbine Trip

Not analyzed for current cycle.

b. Turbine Trip with Failure of Bypass

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

Turbine Trip initiates closure of the turbine stop valves.

Turbine stop valve closure initiates a reactor scram trip via position signals to the protection system. Credit is taken for successful operation of the reactor protection system.

Turbine Bypass Valves fail to open (Failure of the Main Turbine Bypass System).

Turbine Trip initiates recirculation pump trip (RPT) thereby reducing core flow.

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

15B.15.2.3.2.3 The Effect of Single Failures and Operator Errors

a. Turbine Trips at Power Levels Greater Than 38 Percent NBR

Mitigation of pressure increase is accomplished by the reactor protection system functions. Main stop valve closure scram trip and RPT are designed to satisfy single failure criteria.

b. Turbine Trips at Power Levels Less Than 38 Percent NBR

Not analyzed for the current cycle.

15B.15.2.3.3 Core and System Performance

15B.15.2.3.3.1 Mathematical Model

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

15B.15.2.3.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in <Table 15B.15.0-1>.

Turbine stop valves full stroke closure time is 0.1 second. This is consistent with the design specification limit given in <Section 10.2>.

The current cycle was analyzed as if the Main Turbine operates in the Partial Arc Mode of operation. This matches the current operation of the main turbine system.

A reactor scram is initiated by position switches on the stop valves when the valves are less than 90 percent open. (This stop valve scram trip signal is automatically bypassed when the reactor is below 38 percent NBR.

Reduction in core recirculation flow is initiated by position switches on the main stop valves, which actuate trip circuitry which trips the recirculation pumps. (This trip signal is also bypassed below 38 percent NBR.

15B.15.2.3.3.3 Results

a. Turbine Trip

Not analyzed for the current cycle.

b. Turbine Trip with Failure of Bypass

Sensitivity studies have been performed that provide the sensitivity of the licensing basis analysis for the reload cycle. The analysis was performed at the 100% power and 105% core flow with 425.5° normal feedwater temperature representing the most limiting case for this cycle. The simulated turbine trip without bypass transient is shown in <Figure 15B.15.2-2>. The maximum neutron flux, maximum core average heat flux, and calculated delta CPR are provided in <Table 15B.15.0-2>. MCPR remains above the safety limit for this event. Therefore, the design basis is satisfied.

This analysis adequately bounds all MEOD power/flow <Appendix 15E> and partial feedwater heating conditions <Appendix 15D> because it has been shown that the highest core flow (105%) and normal feedwater temperature (425.5°F) initial conditions represent the most limiting conditions for this transient event. The results of this analysis also ensure that the operating limits for single-loop operation <Appendix 15F> continue to be applicable.

c. Turbine Trip with Bypass Failure, Low Power

Not analyzed for the current cycle.

15B.15.2.3.3.4 Consideration 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-rods-out end-of-equilibrium cycle conditions is assumed.

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

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

15B.15.2.3.4 Barrier Performance

15B.15.2.3.4.1 Turbine Trip

Not analyzed for the current cycle.

15B.15.2.3.4.2 Turbine Trip with Failure of Bypass

Safety/relief valve operation limits peak pressure. The peak pressure in the steamlines near the safety/relief valves and the peak reactor coolant pressure at the bottom of the vessel are shown in <Table 15B.15.0-2> and are well below the reactor coolant system transient pressure limit of 1375 psig.

15B.15.2.3.4.2.1 Turbine Trip with Failure of Bypass at Low Power

Not analyzed for the current cycle.

15B.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 <Section 15.2.4.5>. Therefore, the radiological exposures noted in <Section 15.2.4.5> cover the consequences of this event.

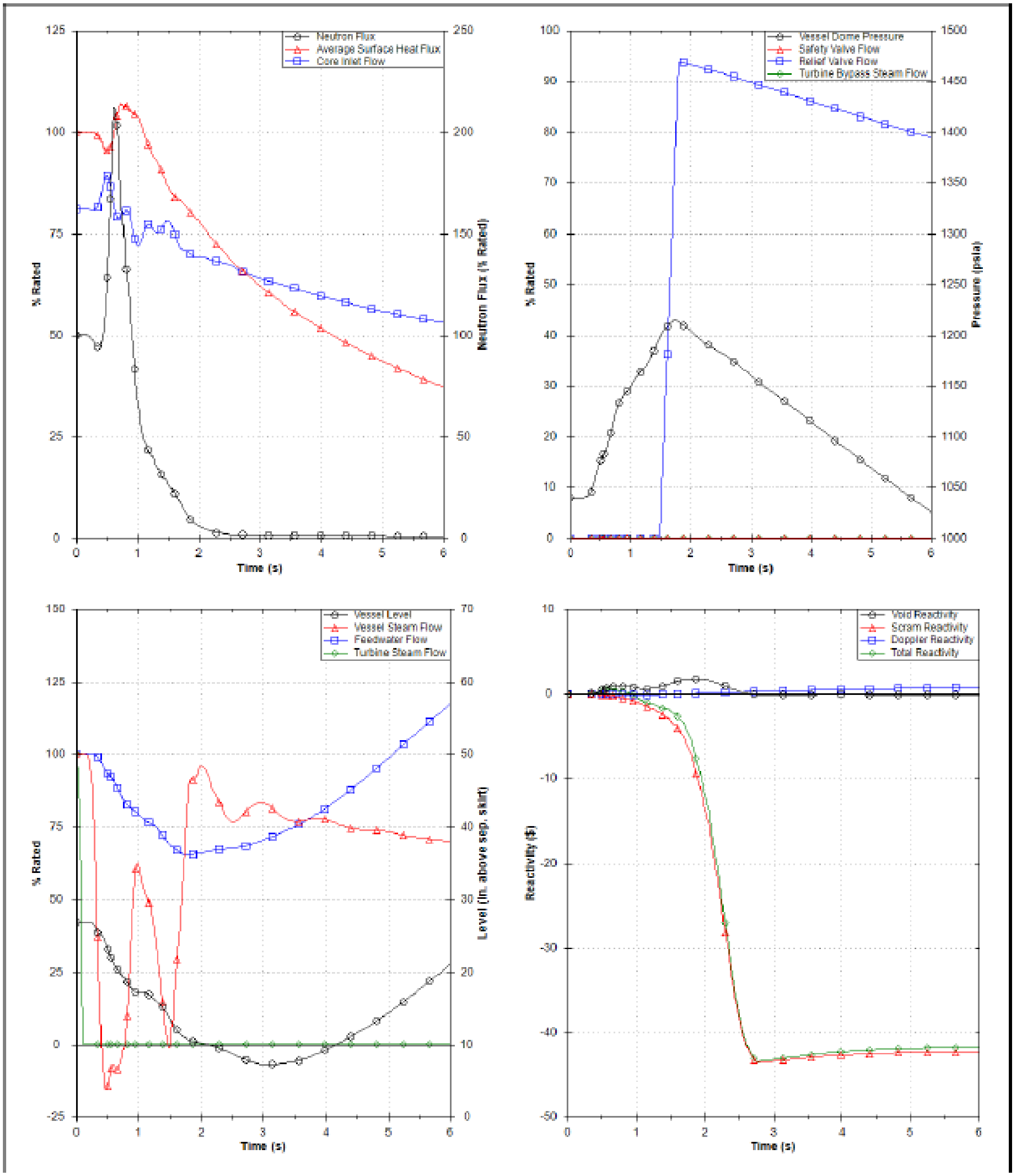
<TABLE 15B.15.2-1>

DELETED

TABLE 15B.15.2-5

SEQUENCE OF EVENTS FOR <FIGURE 15B.15.2-5>
TURBINE TRIP WITHOUT BYPASS

Time-sec(est.)	Event
0	Turbine trip initiates closure of main stop valves.
	Turbine bypass valves fail to operate.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip.
0.01	Main turbine stop valves reach 90% open position and initiate a recirculation pump (RPT) trip.
0.1	Turbine stop valves close.
1.5	Safety/relief valves open due to high pressure.
NA	Vessel water level (L8) trip initiates trip of the feedwater turbines.
>6	Group 1 safety/relief valves cycle to control pressure from decay heat.



Plant Response to Turbine Trip without Bypass (EOC ICF (HBB))

Figure 15B.15.2-2

15B.15.4.2 Rod Withdrawal Error at Power

The Rod Withdrawal Error at Power is identified in GESTAR (Reference 15B-4) to be one of the events most likely to limit operation because of MCPR considerations. This transient is analyzed generically or may be analyzed on a cycle-specific analysis. The applicability of the generic analysis is re-verified as new fuel designs, methodologies, or correlations are developed, e.g., GE11 fuel design, GEXL-PLUS, etc. If the generic analysis cannot be confirmed, then a cycle-specific evaluation is performed until an adequate data base exists to perform generic analysis using methods previously approved by the NRC.

The only difference between the generic analysis and the cycle-specific analysis is the Core and System Performance. For the current cycle, the generic analysis is used.

15B.15.4.2.3 Core and System Performance

15B.15.4.2.3.1 Mathematical Model

For the cycle-specific analysis, the reactor core behavior during the RWE transient is calculated by doing a series of steady-state calculations using a three-dimensional BWR simulator model. 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 time for reactivity insertion is greater than the fuel thermal time constant and core-hydraulic transport times, so that the steady-state assumption is adequate.

15B.15.4.2.3.2 Input Parameters and Initial Conditions

The cycle-specific analysis considers the continuous withdrawal of the maximum worth control rod at its maximum drive speed from the reactor, which is operating at rated power with a control rod pattern which results in the core being placed on thermal design limits. The reload core is assumed to be at its most reactive exposure with no xenon or samarium present. This condition is analyzed to ensure that the results obtained are conservative.

15B.15.4.2.3.3 Results

Results of analyzing the rod withdrawal error are provided in the current cycle Supplemental Reload Licensing Report as well as <Table 15B.15.0-3>. MCPR remains above the safety limit for this event. Therefore, the design basis is satisfied.

15B.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 gangs improperly selected and withdrawn. Because of the near-infinite combinations of control patterns and reactor states, all possible states cannot be analyzed. However, the rod pattern selected is such that the maximum worth control rod is fully inserted and the laterally adjacent or diagonally adjacent bundles are at their thermal operating limit. A series of steady-state calculations are performed for succeeding positions of the worst case control rod.

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.

15B.15.4.7 Misplaced Bundle Accident

Analysis of the misplaced bundle accident (fuel loading error) for the initial core considered only the mislocated bundle accident.

As new fuel types developed, fuel loading errors were re-evaluated for impact. The fuel loading error analysis evaluated as part of the reload safety analysis considers both the mislocated bundle accident <Appendix 15B.15.4.7.1> and the misoriented bundle accident <Appendix 15B.15.4.7.2>.

15B.15.4.7.1 Mislocated Bundle Accident

15B.15.4.7.1.1 Identification of Causes and Frequency Classification

15B.15.4.7.1.1.1 Identification of Causes

This misloading error involves the mislocation of at least two fuel bundles. One location is loaded with a bundle which would potentially operate at a lower critical power than it would otherwise. The other location would operate at a higher critical power. The low critical power location could have less margin to boiling transition than other bundles in the core; therefore, the MCPR operating limit is set to protect against this occurrence.

The mislocated bundle accident was not analyzed for the Reload 18/Cycle 19. Since cycle specific Operating Limit Minimum Critical Power Ratio is greater than 1.28 x (cycle specific safety limit/1.07) GESTAR methodologies do not require this transient to be analyzed.

15B.15.4.7.2 Misoriented Bundle Accident

15B.15.4.7.2.1 Identification of Causes and Frequency Classification

15B.15.4.7.2.1.1 Identification of Causes

The event discussed in this section is the improper orientation of a fuel bundle and subsequent operation of the core. The local and radial peaking factors increase sufficiently in the misoriented bundle to be of safety concern relative to fuel rod overheating (MCPR) and clad overstraining (1% plastic strain). Therefore, the MCPR operating limit is set to protect against this occurrence.

15B.15.4.7.2.1.2 Frequency of Occurrence

The misoriented bundle accident is an infrequent incident which affects only a limited number of fuel assemblies in the core.

Proper orientation of the fuel assembly in the reactor core is readily verified by visual observation and assured by core verification procedures following core loading. The core verification procedures are designed to minimize the possibility of the occurrence of the misoriented bundle accident.

15B.15.4.7.2.2 Sequence of Events and Systems Operation

Misoriented bundles, undetected by in-core instrumentation following refueling 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.

15B.15.4.7.2.2.1 Effects of Single Failures and Operator Errors

This analysis already represents the worst case, i.e., operation with a misoriented bundle, and there are no further operator errors which can make the event results any worse. This section is not applicable to this event; refer to <Appendix 15A> for further details.

15B.15.4.7.2.3 Core and System Performance

15B.15.4.7.2.3.1 Mathematical Model

A steady-state nuclear methods code is used to establish the local power peaking for both the 90 degree and 180 degree rotated assembly. Using the maximum bundle power increase for the entire cycle, a three-dimensional BWR simulator model is used to calculate the critical power ratio for the misoriented bundle.

15B.15.4.7.2.3.2 Input Parameters and Initial Conditions

The fuel loading error involves the misorientation of a fresh reload fuel bundle. The maximum bundle power increase is calculated due to both 90 and 180 degree rotation throughout the entire cycle. Additionally, the misoriented bundle is assumed to be tilted which affects the size of the water gap surrounding the bundle. This conservatively maximizes the bundle power increase for the misoriented bundle, since the rods with the highest enrichment are assumed adjacent to an unrealistically large water gap.

15B.15.4.7.2.3.3 Results

The misoriented bundle critical power ratio is computed. The results of the analysis are provided in the current cycle Supplemental Reload Licensing Report as well as <Table 15B.15.0-3>. The results include a 0.02 delta CPR penalty for a “tilted” misoriented bundle. The MCPR remains above the safety limit for this event. Therefore, the design basis is satisfied.

One possible state of operation for the fuel bundle is that it operates through the cycle close to or above the fuel thermal mechanical limit. The potential exists that if the fuel bundle operates above the thermal mechanical limit, one or more fuel rods may experience clad overstraining and failure. If this were to occur, the adverse consequences of operation are detectable and can be suppressed during operation. In this context, the adverse consequence is the perforation of a small number of fuel rods in the misoriented fuel assembly. Any perforations that may result would be localized, there would be only a few perforations, and the perforations would not propagate to other fuel rods or fuel assemblies.

15B.15.4.7.2.3.4 Consideration of Uncertainties

In order to ensure the conservatism of this analysis, a 0.02 delta CPR penalty is added for the tilted misoriented bundle. This ensures that the minimum CPR is conservatively bounded.

15B.15.4.7.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since it is a highly localized event with no increase in gross core power. No perceptible change in reactor pressure would occur. Thus, there is no concern relative to the barrier performance.

15B.15.4.7.2.5 Radiological Consequences

The perforation of a small number of fuel rods leads to the release of fission products to the reactor coolant, which are detected by the offgas system. An evaluation of radiological consequences is not required for this event due to the limited number of fuel rods affected (Reference 15B-4).

15B.16 ALTERNATE OPERATING MODE CONSIDERATIONS

The SAFER/PRIME (Reference 15B-45) analysis was applied to three alternate operating modes: feedwater temperature reduction, Maximum Extended Operating Domain (MEOD), and single loop operation. The following paragraphs describe these operating modes with respect to the SAFER/PRIME analysis.

15B.16.1 FEEDWATER TEMPERATURE REDUCTION

<Appendix 15B.15.1.1> discusses Loss of the Feedwater Heating (LOFH) capability of the plant up to 100°F. The impact on LOCA due to a Feedwater Temperature Reduction (FWTR) of up to 170°F was evaluated using SAFER/PRIME, (Reference 15B-45). This analysis was performed in the MEOD region at rated core power and low flow point (81% rated core flow) for the limiting case. The same ECCS parameters used for normal feedwater temperature are assumed for the FWTR LOCA analysis. Results show that the FWTR PCTs at 81% core flow for <10 CFR 50, Appendix K> cases are higher for both the first and second peaks compared to the corresponding PCTs for the rated core flow case with normal feedwater temperature. The Nominal FWTR cases are higher for the first peak and lower for the second peak compared to the corresponding PCTs for the rated core flow case with normal feedwater temperature (Reference 15B-45). Overall, there is minimum impact upon the licensing basis PCT.

15B.16.2 MAXIMUM EXTENDED OPERATING DOMAIN

The DBA suction break with HPCS DG failure case was analyzed for Perry at 81% rated core flow with both nominal and <10 CFR 50, Appendix K> assumptions using SAFER/PRIME. Results show that the PCT for a design basis LOCA at the MEOD low flow condition is approximately 97°F higher than the PCT associated with normal operations. This conclusion is valid for the Appendix K case, 3834 MWt core power level at 81% rated core flow conditions (Reference 15B-45). There is no required low flow MAPLHGR multiplier for ECCS considerations.

The high flow point (105% rated core flow) in the MEOD region was evaluated for the same case <Section 6.3.6>, (Reference 5). There is no adverse effect for the increased core flow. The nominal PCT is lower than the base case and the <10 CFR 50, Appendix K> PCT shows that there is a negligible effect compared to the base rated core flow case.

15B.16.3 SINGLE LOOP OPERATION

The ECCS analysis for Perry under Single Loop Operation (SLO) was evaluated using SAFER/PRIME methodology. The recirculation suction line break DBA with HPCS DG failure is analyzed at 66.6% rated power (2,502 MWt) and 53.2% rated core flow 55.3 Mlb/hr condition, with both nominal and <10 CFR 50, Appendix K> assumptions. This analysis assumes that there is essentially no period of recirculation pump coastdown (Reference 15B-45).

The approach used to assess SLO is to determine a MAPLHGR multiplier such that the PCT for SLO using nominal assumptions does not exceed the two-loop PCT at rated conditions. This results in a MAPLHGR multiplier of 0.80 for GNF2 SLO conditions. Using this MAPLHGR multiplier, the corresponding <10 CFR 50, Appendix K> SLO PCTs are calculated to confirm remaining below the <10 CFR 50, Appendix K> two-loop PCTs.

15B.16.4 PRESSURE REGULATOR OUT OF SERVICE OPTION

Pressure Regulator Out of Service (PROOS) option is an analysis using the Pressure Regulator Downscale Failure (PRDF) at off-rated conditions. At full power, the PRDF is bounded by other pressurization transients. However, as the reactor power at the beginning of the transient decreases, the impact of the PRDF to MCPR increases.

During a PRDF transient, the pressure regulator closes the turbine control valves. This increases pressure, which increases power in the reactor. When the reactor is at full power, the pressure and power increases quickly and causing a SCRAM. As the reactor power is decreased, the power is further from the SCRAM setpoint so it takes more time to SCRAM. This longer time to SCRAM increases the amount of specific heat in the fuel and impacts the CPR. There is a range of initial reactor power where the CPR is no longer bounded by the normal $MCPR_p$ limits.

There are two independent channels in the pressure regulating system and the PRDF transient is not applicable when both channels are operable.

The COLR identifies the range of the modified MCPR limits and the new limits. These limits may be incorporated by either a revision to the monitoring system or appropriate administrative limits.

15B.16.5 POWER LOAD UNBALANCE OUT OF SERVICE OPTION

Power Load Unbalance Out of Service (PLUOOS) option is an analysis which assumes that the Power to Load Unbalance Circuit will not cause a scram at off-rated conditions <Reference 15B-52>.

The Power Load Unbalance Circuit sense the difference between the generator load and the turbine load. If the difference exceeds the trip setpoint then the PLU cause a turbine control value fast closure. The turbine control valve fast closure then causes a reactor scram.

At full power, the Power to Load Unbalance (PLU) transient is bounded by other pressurization transients. However, if the initial reactor power at the beginning of the transient is reduced, the impact of the PLU transient to Minimum Critical Power Ratio (MCPR) and fuel Linear Heat Generation Rate (LHGR) increases. During a PLU transient, the reactor pressure increases, which increases power in the reactor. When the reactor is at full power, reactor pressure and power quickly increase to the pressure/power scram setpoints causing an automatic reactor shutdown (SCRAM). If the initial reactor power at the beginning of the transient is reduced, more time is required for the reactor to reach the pressure/power scram setpoints. This longer time to SCRAM increases the amount of specific heat in the fuel and thus impacts the Critical Power Ratio and Linear Heat Generation response of the fuel. With the Power Load Unbalance In Service, the normal Offrated Fuel Thermal Limits provide adequate protection to prevent the fuel from exceeding the Fuel Safety Limits.

With the Power Load Unbalance not in service, analysis has shown that over range of initial reactor powers, the normal Offrated Fuel Thermal Limits may not provide adequate protection for the fuel. The Power Load Unbalance Out Of Service (PLUOOS) analysis <Reference 15B-52> developed a new set of Offrated Thermal Limits (PLUOOS). If the fuel is operated within these limits (PLUOOS) then the Fuel Safety Limits will not be exceeded should a PLU transient were to occur from reduced power levels with the PLUOOS. This analysis did not evaluate the impact of PLUOOS on the main turbine and its associated systems.

The Core Operating Limits Report contains the additional set of Offrated Thermal Limits for use with the PLUOOS. These limits may be incorporated by either a revision to the core monitoring system or appropriate administrative limits.

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<APPENDIX 15C>

ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

APPENDIX 15C

TABLE OF CONTENTS

LIST OF TABLES

15C ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

A generic assessment of BWR mitigation of ATWS has been analyzed in depth in GE topical report NEDO-24222, "Assessment of BWR Mitigation of ATWS" (Reference 1). A Perry specific assessment follows.

The Perry ATWS analysis was re-performed for uprate to 3,758 MWt as discussed in the following sections.

15C.1 IDENTIFICATION OF CAUSES

The BWR scram systems are highly reliably and redundant. They have been constantly improved over the years to ensure that the control rods will be inserted upon demand under all conditions of operation. Therefore, the only postulated failure to scram would be an unforeseen, undetected simultaneous failure in the scram function resulting in a significant number of control rods failing to insert into the core upon demand. Furthermore, consequences of concern will only occur if this failure to scram is combined simultaneously with a few particular plant transients. These particular plant transients are also rare.

15C.2 FREQUENCY CLASSIFICATION

The probability of an ATWS event is extremely remote. An ATWS event has never happened.

15C.3 SAFETY CRITERIA

The equipment described in <Appendix 15C.5> was designed to meet the following requirements of <10 CFR 50.62>.

1. Installation of an Alternate Rod Insertion (ARI) system;

2. Boron injection equivalent to 86 gpm into 251 inch inside diameter vessel;

3. Installation of automatic Recirculation Pump Trip (RPT) logic (i.e., ATWS-RPT).

Plant specific analysis have been performed (Reference 4) to assure that the following acceptance criteria have been met.

1. Peak vessel bottom pressure less than ASME Service Level C limit (1,500 psig).

2. Peak cladding temperature does not exceed the limit of <10 CFR 50.46> (2,200°F).

3. Maximum cladding oxidation does not exceed the limit of <10 CFR 50.46> (17% of wall thickness).

4. Maximum suppression pool temperature does not exceed its design temperature (185°F).

5. Maximum containment pressure does not exceed its design pressure (15 psig).

15C.4 INITIAL CONDITIONS COMMON TO ALL EXAMINED CASES

Initial operating conditions used in the ATWS analyses for the Perry Nuclear Power Plant are contained in (Reference 4).

In all the transients analyzed and discussed hereafter, the scram system is postulated to have failed and ARI capability is not taken into account. The operators in the control room are assumed to activate the

SLCS promptly following the postulated event based on information available from the APRM's, ATWS signals of the RRCS logic and the Emergency Operating Procedures (EOPs).

15C.5 DESCRIPTION OF SYSTEMS AND EQUIPMENT DESIGNED EXCLUSIVELY FOR ATWS PREVENTION AND MITIGATION

The following is a summary of plant system and equipment design added or modified exclusively for the mitigation of an ATWS event. The scram discharge volume design has also been modified to minimize ATWS probability <Appendix 15C.6>.

The systems and features described below are tested to assure the functions required by <10 CFR 50.62> will be accomplished in a reliable manner. These design functions, in conjunction with established emergency instructions, provide assurance that the Perry Nuclear Power Plant is capable of responding appropriately to any postulated ATWS event.

The Redundant Reactivity Control System (RRCS)

The Redundant Reactivity Control System (RRCS) is the system which controls ATWS mitigation functions. This system consists of associated ATWS detection sensors (4 RPV dome high pressure sensors and 4 low vessel water level sensors) and the actuation logic to automatically initiate Alternate Rod Insertion (ARI), Recirculation Pump Trip (RPT) and Feedwater Runback. The ATWS sensors and logic also provide indication and alarm in the control room.

The RRCS is activated by either of the two divisions of the ATWS detection sensors and detection logic. The RRCS logic uses APRM signals (with appropriate time delay) to confirm the ATWS event. The RRCS can also be manually initiated.

The RRCS exceeds the requirements of <10 CFR 50.62> by providing specific ATWS signals and alarms to facilitate prompt operator response, and by automating runback of the feedwater. The RRCS is a highly reliable IE system which is electrically diverse from the RPS and built to IEEE-279, IEEE-323 (1974) and IEEE-344 (1975).

Alternate Rod Insertion (ARI)

The function of the alternate rod insertion (ARI) system is to provide an electrically diverse scram logic independent of the RPS, to blow down the scram discharge air header through valves separate from the reactor protection system (RPS) scram valves, thereby providing a parallel path for control rod insertion. ARI consists of redundant scram pilot air header exhaust valves which are actuated by the ATWS detection sensors of the RRCS logic.

The ARI system is Class 1E dc, built to IEEE-279 standards. (See <Section 7.6.1.12> for more details.)

Recirculation Pump Trip (RPT)

The recirculation pump motors are tripped by ATWS signals from the RRCS logic. The purpose of the RPT design is to reduce thermal power level and limit pressure rise in the reactor vessel.

The RPT design shall meet the following requirements (see <Section 7.6.1.12> for more details):

a. Meet IEEE 323-1974 and 344-1975, or be consistent with existing plant design requirements;

b. Meet IEEE 279, 379 and 384 (except for the Low Frequency Motor/Generator breakers);

c. Provide for inservice testability (except for action of final breakers).

Feedwater Runback

Feedwater flow is limited upon receipt of a high pressure signal and confirmed failures to scram from the RRCS logic, thereby reducing power and steam discharge to the supression pool without operator action. The feedwater runback function uses control-grade equipment and allows manual override to allow an increase in feedwater flow as appropriate.

Standby Liquid Control System

The Perry Standby Liquid Control (SLC) system has been evaluated to assure that it is capable of delivering the minimum flow rates required by the ATWS Rule <10 CFR 50.62>. This will control the nuclear fission chain reaction and thereby maintain suppression pool temperatures within specified limits. The boron solution is injected through the high pressure core spray piping to provide good mixing with core cooling water. Simultaneous operation of both loops of the SLC system is initiated manually from the control room in accordance with emergency instructions. The indicators and alarms provided by the RRCS and Rod Control and Information System assist the operator in determining that SLC initiation is appropriate. (See <Section 9.3.5.2> for more details.)

The SLC system design shall:

a. Provide a manual boron solution injection function for both loops simultaneously operated only from the Control Room;

b. Provide for replenishment capability of the SLC tank with a mixed boron solution from outside the containment;

c. Provide capability for periodic functional tests;

d. Assure that no single active logic component failure can prevent its function; and

e. Meet IEEE 323-1974 and 344-1975 or be consistent with existing plant design requirements.

15C.6 SCRAM DISCHARGE VOLUME MODIFICATIONS

Additionally, the control rod drive system scram discharge volume has been modified to minimize the potential for failure of the scram function from unavailability of this volume. The design modification will consist of the addition of redundant instrument volume water level sensors to the control rod drive hydraulic system and instrument line piping modifications. The design provides redundant 1E sensors, and redundant vent and drain valves.

15C.7 ATWS EVENT AND RESULTS

In order to study reactor responses with the injection of boron solution, the Alternate Rod Insertion (ARI) is deliberately ignored in this study, because with ARI there is no need for boron injection. As a result of the introduction of GNF2 fuel, two events were evaluated:

1. Main Steam Isolation Valve Closure,

2. Pressure Regulator Failed – Open (Maximum Demand).

The Main Steam Isolation Valve Closure and Pressure Regulator Failed Open are the limiting events for the ATWS acceptance criteria.

The key input parameters for the ATWS analysis are as follows:

1. Reactor power at 3,758 MWt,

2. Reactor dome pressure at 1,040 psia,

3. SRV opening setpoints,

4. ATWS high reactor pressure setpoint, and

5. Assumption of two SRVs out of service.

Details of these parameters and other inputs are documented in (Reference 4).

For the limiting cases, SLCS manual initiation is assumed to occur at 120 seconds.

Each event was analyzed at beginning of cycle and end of cycle at the minimum core flow condition. <Table 15C-8> provides the maximum values of the key parameters from the limiting cases as compared to the ATWS acceptance criteria.

15C.8 CONCLUSIONS

The Perry unique study presented here, and in (Reference 4) have shown that, with the ATWS equipment described, PNPP can withstand the consequence of an ATWS and still meet the safety criteria in <Appendix 15C.3> even with the ARI ignored. The summary of the ATWS results is given in <Table 15C-8>.

15C.9 REFERENCES

1. NEDE-24222, "Assessment of BWR Mitigation of ATWS," December 1979.

2. GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991; NEDO-31984, Class I (Non-proprietary), March 1992; and Supplements 1 & 2.

3. GE Nuclear Energy, "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors (Supplement 1 – Volume 4)," Licensing Topical Report NEDC-24154P Supplement 1, Class III, December 1997.

4. GE Hitachi Nuclear Energy, "GNF2 Fuel Design Cycle-Independent Analyses for Perry Nuclear Power Plant," 000N0605 R2, Revision 1, November 2014.

5. NEDC-33270P, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," Revision 5, May 2013.

<TABLE 15C-1>

<TABLE 15C-2>

<TABLE 15C-3>

<TABLE 15C-4>

<TABLE 15C-5>

<TABLE 15C-6>

<TABLE 15C-7>

(DELETED)

TABLE 15C-8

KEY ATWS RESULTS VS. ACCEPTANCE CRITERIA

Parameter	Acceptance Criteria	Limiting Uprated Power Result
Peak Vessel Pressure (psig)	1,500	1,285
Peak Cladding Temperature (°F)	2,200	not calculated[1]
Peak Local Oxidation (%)	17	not calculated[1]
Peak Suppression Pool Temperature (°F)	185	178
Peak Containment Pressure (psig)	15	9.3

NOTE:

[1] The Peak Cladding Temperature and Peak Local Cladding Oxidation values are dispositioned generically in (Reference 5).

<APPENDIX 15D>

PARTIAL FEEDWATER HEATING OPERATION ANALYSIS

APPENDIX 15D

TABLE OF CONTENTS

APPENDIX 15D

LIST OF TABLES

15D PARTIAL FEEDWATER HEATING (PFH) OPERATION

15D.1 INTRODUCTION AND SUMMARY

This section presents the results of a safety and impact evaluation for the operation of the Perry Nuclear Power Plant (PNPP) with partial feedwater heating at a steady-state condition during the operating cycle and beyond the end of cycle conditions. This evaluation is performed on an equilibrium cycle basis and is applicable to its initial core and its subsequent reload cycles. The results of this evaluation justify PNPP operation at 100% thermal power steady-state conditions with rated feedwater temperature ranging from 425.5°F to 325.5°F, and also beyond the end of cycle with rated feedwater temperature ranging from 425.5°F to 255.5°F.

Operation with partial feedwater heating was also evaluated for the Maximum Extended Operating Domain (MEOD) operating region <Appendix 15E>. The acceptability of partial feedwater heating for each cycle is reverified for the limiting transients as part of the reload safety analysis <Appendix 15B>.

The Oscillation Power Range Monitor setpoint methods have been changed to NEDE-33766P-A, GEH Simplified Stability Solution (GS3) (Reference 15B-51). The GS3 methods are a generic approach to establishing the OPRM Period Based Detection Algorithm Setpoints. This generic approach assumes feedwater temperature reductions are limited to 120°F anytime during the cycle.

Reload core designs will continue to be developed assuming feedwater temperature reductions of 100°F during the cycle and 170°F beyond the end of the normal fuel cycle. Plant operations will be limited to 100°F during the cycle to account for the assumptions of the core design and 120°F beyond the end of the normal fuel cycle to account for the GS3 limitation.

Operation with partial feedwater heating (PFH) occurs in the event that (1) certain stage(s) or string(s) or individual heater becomes inoperable, or (2) intentionally valving out the extraction steam to the feedwater heaters at the end of an operating cycle. <Chapter 15> has already evaluated the consequence of the transient with a sudden feedwater temperature loss of 100°F when initiated from 425.5°F rated feedwater temperature. This appendix will justify the continued operation of PNPP at the steady-state condition ranging from rated feedwater temperature of 425.5°F to 325.5°F during the operation cycle and, as low as 255.5°F beyond the normal operating cycle.

Initial evaluations required to justify PFH operation included abnormal operating transients, thermal hydraulic stability, the critical feedwater nozzle and sparger fatigue usage conditions, and the worst loss of coolant and containment response conditions. The results are summarized below:

a. The abnormal operating transients in <Chapter 15> were re-evaluated to determine the required operating MCPR limits for PFH operation. According to the worst limiting transient, the operating limit MCPR for the initial cycle needs to be increased by 0.01, that is 1.19 for the initial core and 1.20 for the reload core (based on extrapolating initial cycle conditions) during operating when the rated feedwater temperature is between 370°F and 320°F. For operation beyond the end of normal cycle ranging from 320°F to 250°F rated feedwater temperature, the operating limit MCPR for the initial cycle needs to be increased by 0.03, that is, 1.21 for the initial core and 1.22 for the reload core (based on extrapolating initial cycle conditions). The actual required operating limit MCPRs for the current reload cycle, for the various temperature regimes is presented in <Appendix 15B>, Reload Safety Analysis.

b. The loss-of-coolant accident (LOCA) and containment response as described in <Chapter 6> were re-evaluated for PFH operating condition. It is found that the conditions with normal feedwater temperature at 420°F bound those at PFH conditions.

c. Fuel integrity was evaluated with respect to General Design Criterion 12 <10 CFR 50, Appendix A>. It is shown that PFH operation satisfies the stability criteria and fuel integrity is not compromised.

d. The effect of acoustic and flow induced loads on the reactor shroud, shroud support and jet pumps were re-investigated to assure that design limits are not exceeded. The effect of PFH on feedwater nozzle and sparger fatigue usage factor was examined. It was found that the increased fatigue usage in 40 years still meets the acceptance criteria.

A re-analysis of the partial feedwater heating transient was performed at 3,758 MWt core power conditions summarized in <Table 15.0-1>. This re-analysis considered the same Load Rejection and Feedwater Controller Failure transients presented here except at a feedwater temperature of 225.5°F. The results of the re-analysis are summarized in <Table 15D-1b> and <Table 15D-2b>, and <Figure 15D-3b> and <Figure 15D-6b>.

There are also other impact evaluations such as the feedwater piping, the effect of annulus pressurization and the consequences of Anticipated Transient Without Scram (ATWS). These evaluations concluded that the Perry design is adequate for PFH operation. Operation with feedwater heater(s) out-of-service during the operating cycle and operation at end of cycle with final feedwater temperature reduction are acceptable for PNPP.

15D.2 FUEL INTEGRITY - MCPR OPERATING LIMIT

15D.2.1 ABNORMAL OPERATING TRANSIENTS

All normal operating transients in <Chapter 15> were investigated for PFH operation. Three limiting abnormal operating transients are discussed here in detail. They are:

a. Generator Load Rejections with Bypass Failure (LRNBP)

b. Feedwater Flow Controller Failure (FWCF)

c. Loss of 100°F Feedwater Heating (LFWH)

The evaluations were initially performed at 3,729 MWt core power, 100% core flow with rated feedwater temperature of 370°F, 320°F, and 250°F at end of equilibrium cycle. Initial conditions for partial feedwater heating consistent with the MEOD analyses and new GEMINI set of analysis methods are applied to the reload analyses. Plant heat balance, core coolant hydraulic and nuclear transient data consistent with <Chapter 15> input were developed and used in the analyses. Full arc (FA) turbine control valve closure characteristics were assumed in the analyses which is more limiting than partial arc.

The end of equilibrium cycle exposure point with all the control rods fully withdrawn is the most limiting point in the cycle with the worst scram reactivity worth characteristics. A middle of the cycle point (2,000 MWd/t before end of equilibrium cycle) was also analyzed for 370°F and 320°F rated feedwater temperatures to demonstrate operation during the operating cycle at these feedwater temperatures. This point is chosen 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 but the void reactivity characteristics are different than end of cycle. Scram characteristics are significantly improved at exposure lower than this point and the transient responses will be bounded by the two point analyzed. It is shown that the end of equilibrium cycle condition bounds the middle of cycle conditions.

The computer model described in (Reference 15D.11-1) was used to simulate the transient a) and b) events. The results for the bounding cases are summarized in <Table 15D-1> and <Table 15D-2>. As shown in

<Table 15D-2>, the operating MCPR limit shall be 1.19 (for the initial cycle) for operation between rated feedwater temperature of 370°F and 320°F. Operation between 320°F and 250°F rated feedwater temperature requires a rated operating limit of 1.21 (for the initial cycle).

Lower initial operating pressure and steam flow rate (due to lower feedwater temperature) provide better overpressure protection for the limiting MSIV closure flux scram event. Hence, it is concluded that the pressure barrier integrity is maintained under partial feedwater heating (PFH) conditions.

The transient responses for transients a) and b) are presented in <Figure 15D-1 (1)>, <Figure 15D-1 (2)>, <Figure 15D-2 (1)>, <Figure 15D-2 (2)>, <Figure 15D-3a>, <Figure 15D-3b>, <Figure 15D-4 (1)>, <Figure 15D-4 (2)>, <Figure 15D-5 (1)>, <Figure 15D-5 (2)>, <Figure 15D-6a (1)>, <Figure 15D-6a (2)>, <Figure 15D-6a (3)>, <Figure 15D-6a (4)>, and <Figure 15D-6b>.

The 100°F loss of feedwater heating transient for the initial cycle was evaluated at 3,729 MWt core power, 100% core flow with rated feedwater temperatures of 250°F and 420°F at the end of equilibrium cycle using the computer model described in (Reference 15D.11-2) and methodology described in (Reference 15D.11-3). Results show that the 100°F loss of feedwater heating has less effect on colder feedwater than on the normal feedwater temperature of 420°F. Thus, the ΔCPR results for the case with 250°F initial rated feedwater temperature are bounded by the 420°F rated normal case. Moreover, it is less likely to have a sudden 100°F loss at an initial feedwater temperature of 250°F.

As part of the 3,758 MWt core power partial feedwater heating re-analysis, the Loss of Feedwater Heater transient was re-analyzed at a feedwater temperature loss of 100°F (from 425.5°F to 325.5°F). The results of this transient are similar to those reported in the initial core analysis, and the transient is not the limiting transient for operating limit determination.

15D.2.2 ROD WITHDRAWAL ERROR

A rod withdrawal error analysis case consistent with the BWR/6 generic rod withdrawal error analysis as discussed in <Section 15.4.2> was performed at initial feedwater temperature of 250°F to bound all rated feedwater temperature conditions. The analysis indicated that the initial steady-state feedwater temperature has negligible effect with regard to ΔCPR in a random rod withdrawal error condition. Thus, the ΔCPR values initiating from 250°F feedwater temperature condition fall within the statistical data base used to establish the Rod Withdrawal Limiter System setpoints. Therefore, the generic Rod Withdrawal Error Analysis adequately bounds PFH operation conditions.

15D.3 FUEL INTEGRITY - STABILITY

General Design Criterion 12 <10 CFR 50, Appendix A> states that power oscillations which result in exceeding specified acceptable fuel design limits be either not possible or be readily and reliably detected and suppressed. Historically, compliance to GDC-12 was demonstrated by assuring that neutron flux oscillations would not occur. This eliminated the need to perform fuel integrity calculations under limit cycle conditions. As a result of stability tests at operating BWRs and extensive development and qualification of GE analytical models, stability criteria have been developed, which also demonstrate compliance to GDC-12. (Reference 15D.11-4) provides these stability compliance criteria for GE fueled BWRs operating in the vicinity of limit cycles.

Operation in the partial feedwater heating (PFH) mode is bounded by the fuel integrity analyses in (Reference 15D.11-4). In general, the effect of reduced feedwater temperature results in a higher initial CPR which yields even larger margins than those reported in (Reference 15D.11-4). The analyses are independent of the stability margin since the reactor is already assumed in limit cycle oscillations. (Reference 15D.11-4) also demonstrates that for neutron flux limit cycle oscillations just below the 120% neutron flux scram setpoint, fuel design limits are not exceeded for those GE BWR fuel designs contained in General Electric Standard Application for Reactor Fuel (Reference 15D.11-5). These evaluations demonstrate that substantial thermal/mechanical margin is available for the GE BWR fuel designs even in the unlikely event of very large oscillations.

To provide assurance that acceptable plant performance is achieved during operation in the least stable region of the power/flow map, as well as during all plant maneuvering and operating states, a generic set of operator recommendations has been developed and communicated to all GE BWRs. These recommendations instruct the operator on how to reliably detect and suppress limit cycle neutron flux oscillations should they occur. The recommendations were developed to conservatively bound the expected performance of all current product lines.

When operating in the partial feedwater heating mode during a cycle, the colder feedwater flow increases the core inlet subcooling and will also result in power distribution changes. These changes result in reduced stability margin when operating in the high power/low flow region of the operating domain. Tests performed at an overseas BWR/6 in October 1984 evaluated the effects of reduced feedwater temperature on stability margins. The result shows that the reduction in stability margin is within the conservative basis of the operator recommendations and therefore the recommendations are applicable for partial feedwater heating during the cycle.

For operation at the end of the cycle with partial feedwater heating to extend the operating cycle, the power distribution approximates the target power shape (typically a Haling power distribution) with all control rods fully withdrawn. Reducing the feedwater temperature at this point will result in an increased peak but at a higher elevation in the core. The change in power shape partially offsets the reduced inlet enthalpy effect on stability and the result is a small change in stability margin. The change in stability margin is well within the conservative basis of the operator recommendations and therefore, the recommendations are applicable to operation with PFH down to rated feedwater temperature of 250°F at the end of cycle conditions.

15D.4 LOSS-OF-COOLANT ACCIDENT ANALYSIS

Refer to <Appendix 15B> for a description of the LOCA analysis relative to feedwater temperature reduction.

15D.5 CONTAINMENT RESPONSE ANALYSIS

The impact of partial feedwater heating (PFH) on the containment LOCA response was evaluated. Both Main Steam Line (MSL) break and recirculation line break were analyzed over the entire power/flow region. Reduced feedwater temperature increases the subcooling of the coolant, and the mass flow rate from the postulated recirculation pipe break also increases, but is limited to the critical flow of the break. The final outcome is that the peak drywell and containment pressures under the partial feedwater heating conditions are bounded by the design values in <Chapter 6>.

15D.6 ACOUSTIC LOAD AND FLOW INDUCED LOADS IMPACT ON INTERNALS

Acoustic loads are loads on vessel internals created by a sudden LOCA. Acoustic loading is proportional to total pressure wave amplitude to 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. Partial feedwater heating operation increases subcooling in the downcomer thus increases critical flow. However, PFH also increases density. The reactor internals most impacted by acoustic and flow induced loads are the shroud, shroud support and jet pump. The impacts on these components were evaluated over the operating power flow region. The analyses concluded that these components have been designed to handle the loading during reduced feedwater temperature conditions.

15D.7 FEEDWATER NOZZLE FATIGUE USAGE

An evaluation was performed on the PNPP feedwater nozzle with partial feedwater heating at rated feedwater temperature of 250°F for conservatism. An 18 month operating cycle with partial feedwater heating based on an 80% capacity factor is equivalent of 438 full power days per cycle. This results in an additional 0.0214 fatigue usage factor over 40 years of continuous operation at 250°F. Furthermore, if we assume additional end of cycle operation with feedwater temperature between 420°F and 250°F for 41 full power days per cycle for 40 years, the resultant fatigue usage factor would increase 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 partial feedwater heating operation is extremely conservative. The nozzle fatigue is expected to be much less than the results presented above. Hence, PFH operation is an acceptable mode even for the most "fatigue-critical" vessel nozzle.

15D.8 FEEDWATER SPARGER IMPACT EVALUATION

An evaluation was performed to examine the impact of partial feedwater heating operation on the feedwater sparger for PNPP. Six cases were analyzed to determine the number of days allowable per year (for 40 years) for partial feedwater heating operation without exceeding the feedwater sparger fatigue usage factor limit of 1.0. Results of this study are presented in <Table 15D-3>. This table indicates the annual average number of days allowable for partial feedwater heating, reducing from normal 420°F to 370°F or to 320°F rated feedwater temperature with an additional 41 end of cycle days at 250°F. For example, the feedwater sparger is designed to operate with 21 days of partial feedwater heating at rated 320°F during a fuel cycle and 41 days of partial feedwater heating at rated 250°F beyond the end of the fuel cycle for every fuel cycle for 40 years. The feedwater sparger is acceptable for partial feedwater heating operation within these limits.

15D.9 REACTOR PROTECTION SYSTEM SETPOINTS

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, vessel pressure scram and other normal scram functions provide sufficient protection.

Therefore, below 38% rated power, turbine stop valve and control valve scram functions are bypassed. The 38% NB rated power is sensed through the direct measurement of the turbine first stage pressure.

As feedwater temperature is reduced, steam flow decreases. If the core thermal power is maintained with partial feedwater heating, the steam flow change means that the turbine first-stage pressure versus power relationship is altered. Thus, it is necessary to readjust turbine stop and control valve scram bypass setpoints (sensed from turbine first stage pressure) for partial feedwater heating operation. A new setpoint is established for the trip units prior to commencement of each partial feedwater heating operation at each operating cycle.

15D.10 MISCELLANEOUS IMPACT EVALUATION

15D.10.1 FEEDWATER SYSTEM PIPING

The impact of partial feedwater heating operation on the feedwater system piping up to the first feedwater guide lug outside the containment has been evaluated for feedwater temperature at 250°F. Results of the study show that with the additional partial feedwater heating operations, the feedwater piping fatigue usage factor still meets the allowable limit of 1.0.

15D.10.2 IMPACT ON ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)

An impact evaluation performed for PNPP shows that reducing feedwater temperature helps to reduce the consequences of an ATWS event. As a result of reduced feedwater temperature, steam flow and core average void fraction are reduced. This results in lower void coefficient and greater CPR margin which corresponds to milder transients.

15D.10.3 ANNULUS PRESSURIZATION LOAD (APL) IMPACT

A boundary analysis was performed to determine the impact of partial feedwater heating operation on annulus pressurization loads (APL). It is found that partial feedwater heating has a small impact on annulus pressurization loads and is bounded by the normal operation APL limits.

15D.10.4 FUEL MECHANICAL PERFORMANCE

Evaluations were performed to determine the acceptability of PNPP partial feedwater heating 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 with partial feedwater heating conditions. These values were found to be bounded by those applied in the fuel rod and assembly design bases and therefore, PNPP with partial feedwater heating operation is acceptable and consistent with the fuel design basis.

15D.11 REFERENCES

15D.11-1 "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154, October 1978.

15D.11-2 "Three Dimensional BWR Core Simulator," NEDO-20953-A, January 1977.

15D.11-3 Letter, J. S. Charnley (GE) to F. J. Miraglia (NRC), "Loss of Feedwater Heating Analysis," July 5, 1983 (MFN-125-83).

15D.11-4 "Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria," NEDE-22277-P, December 1982.

15D.11-5 General Electric Company “General Electric Standard Application for Reactor Fuel,” including the US Supplement, NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision).

15D.11-6 Letter, David T. Shen (GE) to Eileen Buzzelli “Perry Transient Analysis Assuming 25% Steam Bypass,” October 28, 1987.

TABLE 15D-1a

SUMMARY OF TRANSIENT PEAK VALUE RESULTS

FOR 3729 MWt CORE POWER, 100% CORE FLOW

Transient	Exposure Point (MWd/t)	Rated Fdwtr. Temp. (°F)	Maximum Neutron Flux % (NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam-line Pressure (psig)
Load Rejection With Bypass Failure	EOEC[1]	250	235	1,193	1,221	1,189
	EOEC	320	246	1,198	1,224	1,201
	EOEC	370	245	1,202	1,230	1,209
Feedwater Controller Failure	EOEC	250	174	1,128	1,150	1,127
	EOEC	320	139	1,145	1,167	1,145
	EOEC	370	144	1,160	1,187	1,158

NOTE:

[1] End of equilibrium cycle

TABLE 15D-1b

3758 MWt CORE POWER

SUMMARY OF TRANSIENT PEAK VALUE RESULTS

100% POWER, 105% FLOW

Transient	Exposure Point (MWd/t)	Rated Fdwtr. Temp. (°F)	Maximum Neutron Flux % (NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam-line Pressure (psig)
Load Rejection With Bypass Failure	EOC	255.5	282.9	1165.1	1191.4	1164.7
Feedwater Controller Failure	EOC	255.5	221.7	1125.9	1150.0	1125.2

TABLE 15D-2a

INITIAL CYCLE
SUMMARY OF CRITICAL POWER RATIO RESULTS[1]
FOR 3729 MWt CORE POWER, 100% CORE FLOW

Transient	Exposure Point	Feedwater Temp. (°F)	Req'd[4] Initial MCPR	ΔCPR[4]	End Of[4] Transient MCPR
Load Rejection With Bypass Failure	EOEC[2]	250	1.18	0.11	1.07
	EOEC	320	1.18	0.11	1.07
	EOEC	370	1.18	0.10	1.08
Feedwater Controller Failure	EOEC	250	1.21	0.15[3]	1.06
	EOEC	320	1.19	0.13[3]	1.06
	EOEC	370	1.18	0.11[3]	1.07

NOTES:

(1) This table is applicable to initial core with a safety limit MCPR of 1.06.
(2) End of equilibrium cycle.
(3) Analysis has been performed to conclude that turbine bypass capacity as low as 25% NBR does not affect the bounding ΔCPR results (Reference 15D.11-6).
(4) The required initial MCPR, ΔCPR, and resulting end of transient MCPRs represent initial cycle results. Results for the current reload cycle are presented in <Appendix 15B>, Reload Safety Analysis.

TABLE 15D-2b

3758 MWt CORE POWER
SUMMARY OF CRITICAL POWER RATIO RESULTS
100% POWER, 105% FLOW

Transient	Exposure Point (MWd/t)	Feedwater Temp. (°F)	Req'd Initial MCPR	ΔCPR	End Of Transient MCPR
Load Rejection With Bypass Failure	EOC	255.5	1.28	0.19	1.09
Feedwater Controller Failure	EOC	255.5	1.28	0.19	1.09

TABLE 15D-3

SUMMARY OF FEEDWATER SPARGER FATIGUE ANALYSIS

(for 3729 MWt)

Feedwater Temperature reduction to 250°F for 41 days at End of each 18-Month Cycle for 40 Years	Allowable Number of Days per Year[2] for 40 Years at Feedwater Temperature of	
	370°F	320°F
3 Step[1]	127	21
7 Step[1]	144	24
No end of Cycle reduction	256	61

NOTES:

[1] 3 Step means ~3 average steps of feedwater temperature reduction from 420°F to 370°F or 320°F.

7 Step means ~7 average steps of feedwater temperature reduction from 420°F to 370°F or 320°F.

[2] This evaluation assumes 70% capacity factor. Allowable number of days which results in a feedwater sparger fatigue usage factor of 1.0.

<APPENDIX 15E>

PNPP MAXIMUM EXTENDED OPERATING DOMAIN ANALYSIS

APPENDIX 15E

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES (Continued)

15E MAXIMUM EXTENDED OPERATING DOMAIN

This appendix provides the justification that the operating domain as shown in <Figure 4.4-2> can be extended <Figure 15E.2-1> and still meets all the requirements established by the Code of Federal Regulations with PNPP's 100% full power license conditions.

15E.1 DEFINITION OF THE CURRENT AND THE MAXIMUM EXTENDED POWER/FLOW OPERATING DOMAINS

The current power/flow operating domain as given in <Figure 4.4-2> of <Chapter 4> can be regarded as a map bounded by the following restrictions:

1. The 100% rated power limit.

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 resultant from restrictions on pump speed and FCV position.

The Maximum Extended Operating Domain is essentially extending additional operational power/flow areas to the operating domain given in <Figure 4.4-2>. They are:

a. The Extended Load Line Region (ELLR) - the areas where higher power can be achieved at lower than rated core flow conditions.

b. The Increased Core Flow Region (ICFR) - the area where core flow up to 105% rated is utilized.

15E.2 INTRODUCTION AND SUMMARY

This appendix presents the results of a safety and impact evaluation for operation of the Perry Nuclear Power Plant (PNPP) in an expanded operating envelope called the Maximum Extended Operating Domain (MEOD) to permit improved power ascension capability to full power as well as to provide additional flow range at rated power, including an increased flow region to compensate for reactivity reduction due to exposure during an operating cycle.

MEOD operation is reevaluated for the limiting transients as part of the reload safety analysis <Appendix 15B>.

The total Maximum Extended Operating Domain is shown in <Figure 15E.2-1>. The extended load line region (ELLR) boundary (or MEOD Boundary Line) is limited by 81% core flow at 100% power and the maximum power allowed by the MEOD Boundary Line for a given core flow as defined by (Reference 15E.12-11):

$$P = (22.191 + 0.89714 * F - 0.0011905 * F * F) * 1.149$$

Where: P = % Core Thermal Power

F = % Core Flow

This is determined based on a safety and impact evaluation in meeting thermal and reactivity margins. The Increased Core Flow Region (ICFR) is bounded by the 105% core flow line. This ICFR boundary is limited by plant recirculation system capability, acceptable flow induced vibration and force impact on the vessel internal components.

The MEOD evaluation for the initial cycle was performed for PNPP on an 18 month equilibrium cycle basis and is applicable to 12 month or 18 month cycle operation for both initial and reload cycles with the current GE 6 fuel design. The acceptability of operation in the MEOD operating regime is reverified for the limiting transients (with the current fuel design) as part of the reload safety analysis in <Appendix 15B>. The results of the initial cycle evaluation are:

a. The limiting normal and abnormal operating transients in <Chapter 15> were reevaluated for the MEOD conditions. It is also determined that the fuel mechanical limits are met for all transients occurring in the MEOD.

b. The Loss-of-Coolant Accident and Containment responses as described in <Chapter 6> were reevaluated in the MEOD. It is found that the responses are bounded by the current design analysis.

c. Thermal hydraulic stability was evaluated for its adequacy with respect to the General Design Criterion 12 <10 CFR 50, Appendix A>. It is shown that MEOD operation satisfies this stability criterion.

d. The effect of increased flow induced loads due to increased core flow on the reactor internal components and fuel channels are assured for their design adequacy. The effect of increased flow rate on the flow induced vibration response of the reactor internal will be monitored during startup testing and be evaluated to ensure the responses are within acceptable limits for PNPP.

e. Several impact evaluations were also performed to justify operation in the MEOD. It was found that acceptance criteria and design limits are met.

f. This appendix also justifies Partial Feedwater Heating (PFH) operation described in <Appendix 15D> for rated feedwater temperature ranging from 425.5°F to 325.5°F during and beyond the operating cycle in the MEOD (ELLR and ICFR), and rated feedwater temperature ranging from 325.5°F to 255.5°F beyond the end of cycle in the ICFR. All evaluations described in <Appendix 15D> were reevaluated or reviewed in the MEOD to ensure that PFH operation in this maximum extended operation region is safe and feasible with the required additional modifications to the Technical Specification MCPR limits.

g. A summary of the rated operating limit MCPR values (for the initial cycle) for various modes of operation is tabulated in <Table 15E.2-1>.

Even though the MEOD boundary is set at 105% rated core flow in the ICFR, the transient analyses covered in this appendix include core flow as high as 116% rated <Table 15E.3-5>. It is customary during the reactor startup test program to test the recirculation flow at the full-open position of the recirculation flow control valve as long as no operating limit is exceeded. For Perry, this high flow rate is estimated to be about 108% at 100% power and 112% at 56% power.

TABLE 15E.2-1

Summary of Rated Operating Limit MCPR Values for the Initial Cycle[2][3]

Mode of Operation	Initial Cycle Rated OLMCPR[1]	Current Cycle Rated OLMCPR
Original USAR Power Flow Map <Figure 4.4-2>	1.18	See Note[4]
PFH (420°F to 370°F rated FWT)	1.18	See Note[4]
PFH (370°F to 320°F rated FWT)	1.19	See Note[4]
PFH[5] (320°F to 250°F rated FWT)	1.21	See Note[4]
MEOD Power Flow Map <Figure 15E.2-1>	1.18	See Note[4]
PFH (420°F to 370°F rated FWT) in ELLR	1.18	See Note[4]
PFH (370°F to 320°F rated FWT) in ELLR	1.19	See Note[4]
PFH (420°F to 370°F rated FWT) in ICFR	1.19	See Note[4]
PFH[5] (320°F to 250°F rated FWT) in ICFR	1.21	See Note[4]

NOTES:

(1) All OLMCPRs are for initial core only.
(2) All evaluations and results above are for the initial cycle with GE6 fuel for PNPP with EOC target Haling exposure distribution.
(3) Nomenclature:
PFH = Partial Feedwater Heating operation to be applied both during the operating cycle and beyond the end of cycle <Appendix 15D>
ELLR = Extended Load Line Region
ICFR = Increased Core Flow Region
(4) For the current reload cycle rated OLMCPR for all temperature regimes see <Appendix 15B>, Reload Safety Analysis.
(5) For beyond end of cycle only.

15E.3 MCPR OPERATING LIMIT

15E.3.1 ABNORMAL OPERATING TRANSIENTS

All abnormal operating transients described in <Chapter 15> were examined for Maximum Extended Operating Domain (MEOD) operation. Three Limiting Abnormal Operating Transients are discussed here in detail. They are:

a. Generator Load Rejection With Bypass Failure (LRNBP)

b. Feedwater Flow Controller Failure (FWCF)

c. 100°F Loss of Feedwater Heating

The reevaluations were performed at various MEOD bounding power flow conditions of <Figure 15E.2-1> at the end of the 18 month equilibrium cycle. Initial conditions for the reload safety analysis including MEOD re-evaluation for the limiting transients are consistent with the GEMINI set of methods described in GESTAR (Reference 15E.12-6). For the current cycle initial conditions refer to <Appendix 15B>, Reload Safety Analysis. Plant heat balance, core coolant hydraulics and nuclear transient parameter data were developed and used in the above transient analysis. Full arc (FA) turbine control valve closure characteristics were assumed which is more limiting than partial arc. The initial condition for the lowest and highest flow points at rated power are presented in <Table 15E.3-1> and <Table 15E.3-2>. The computer model described in (Reference 15E.12-1) was used to simulate both the Generator Load Rejection With Bypass Failure and Feedwater Controller Failure events. The transient peak values results and critical power ratio (CPR) results for the two cases for the initial cycle analyzed at 3,729 MWt core power (lowest and highest flow) are summarized in <Table 15E.3-3> and <Table 15E.3-4> respectively. The transient responses are presented in <Figure 15E.3-1 (1)>, <Figure 15E.3-1 (2)>,

<Figure 15E.3-2 (1)>, <Figure 15E.3-2 (2)>, <Figure 15E.3-3 (1)>, <Figure 15E.3-3 (2)>, <Figure 15E.3-4 (1)>, and <Figure 15E.3-4 (2)>. Several other power flow conditions on the MEOD boundary rod line and the maximum core flow boundary were also analyzed. <Table 15E.3-5> identified these additional analyzed power flow points. The results of this evaluation show that the ΔCPR results for all the cases analyzed in the MEOD for the initial cycle are bounded by the original Technical Specification limits. See <Appendix 15E.11> for Partial Feedwater Heating Operation). Technical Specification limits for the current reload cycle are describe in <Appendix 15B>, Reload Safety Analysis.

The 100°F Loss of Feedwater Heating (LFWH) Transient results in <Chapter 15> for the initial cycle are applicable to the MEOD. The Perry plant specific 100°F LFWH analysis was performed at the most limiting exposure point in cycle using the computer models described in (Reference 15E.12-2). The critical power results are given in <Table 15E.3-4>. The results show that the LFWH event is not the limiting transient in the MEOD for the initial cycle. The plant specific LFWH ΔCPR values are bounded by the value documented in <Chapter 15> for this event. The LFWH transient is reperformed for each reload safety analysis.

Additionally, overpressure protection transient analysis using the computer model described in (Reference 15E.12-1) is performed at the various power flow conditions <Table 15E.3-5>. The bounding MSIV closure flux scram event for the initial cycle resulted in a peak pressure of 1,273 psig at a postulated 110% core flow condition. Therefore, it is shown that the peak vessel pressure for the MEOD is below the ASME code limit of 1,375 psig. Hence, adequate pressure margin is present for the <Chapter 15> transients in the MEOD. Over pressure protection analysis is reperformed for each reload safety analysis.

Furthermore, the pressure controller downscale failure event has been examined to show that establishing an operating limit MCPR value for this event is no longer necessary. The PNPP specific steam bypass failure (when the turbine control valves close) is only possible if there is a short in the cabling for the ground of the bypass and the test card. According to IEEE 500-1984, "Reliability for Nuclear Power Generating Stations," the probability for such unique and untimely failure is so remote that even an infrequent transient classification is too conservative.

15E.3.2 ROD WITHDRAWAL ERROR

The rod withdrawal error (RWE) transient documented in <Section 15.4.2> 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 throughout the operating power/flow map including the MEOD region. Therefore, the current Technical Specification MCPR limit is adequate to protect the RWE in the MEOD.

TABLE 15E.3-1

Input Parameters and Initial Conditions for[3]
Transients and Accidents for MEOD, 3729 MWt Core Power, 73.6% Flow

1.	Thermal Power Level, MWt Analysis Value	3,729.3 (104.2% rated)
2.	Steam Flow, lb per sec Analysis Value	4,468 (104.4% rated)
3.	Core Flow, lb per hr	76.5×10^6
4.	Feedwater Flow Rate, lb per sec Analysis Value	4,468
5.	Feedwater Temperature, °F	425
6.	Vessel Dome Pressure, psig	1,044
7.	Core Exit Pressure, psig	1,052
8.	Turbine Bypass Capacity, % NBR	35[1]
9.	Core Coolant Inlet Enthalpy Btu per lb	520.2
10.	Turbine Inlet Pressure, psig	960
11.	Fuel Lattice	P8x8R
12.	Core Leakage Flow, %	12.9
13.	Required MCPR Operating Limit Initial Core	1.27
14.	MCPR Safety Limit for Incidents of Moderate Frequency	
	Initial Core	1.06
15.	Doppler Coefficient (-)¢/°F Analysis Data	0.132[2]
16.	Void Coefficient (-)¢/% Rated Voids Analysis Data for Power Increase Events	14.0[2]
	Analysis Data for Power Decrease Events	4.0[2]

TABLE 15E.3-1 (Continued)

17.	Core Average Rated Void Fraction, %	48.7
18.	Jet Pump Ratio, M	2.25
19.	Safety/Relief Valve Capacity, % NBR	
	@1,210 psig	111.4
	Manufacturer	Dikker
	Quantity Installed	19
20.	Relief Function Delay, seconds	0.4
21.	Relief Function Response Time Constant, sec.	0.1
22.	Analyses Inputs for Safety/Relief Valves	
	Safety Function, psig	1,175, 1,185, 1,195 1,205, 1,215
	Relief Function, psig	1,145, 1,155, 1,165, 1,175
23.	Number of Valve Groupings Simulated	
	Safety Function, No.	5
	Relief Function, No.	4
24.	High Flux Trip, % NBR Analysis Setpoint (122x1.042), % NBR	127.2
25.	High Pressure Scram Setpoint, psig	1,095
26.	Vessel Level Trips, Feet Above Separator Skirt Bottom	
	Level 8 - (L8), feet	5.89
	Level 4 - (L4), feet	4.04
	Level 3 - (L3), feet	2.165
	Level 2 - (L2), feet	(-) 1.739
27.	APRM Thermal Trip Setpoint, % NBR	118.8
28.	RPT Delay, seconds	0.14
29.	RPT Inertia Time Constant for Analysis, seconds	5
30.	Total Steamline Volume, ft^3	3,850

TABLE 15E.3-1 (Continued)

NOTES:

(1) See <Section 10.2.1> and <Table 15E.3-4>.
(2) These values for (Reference 15E.12-4) analysis only. (Reference 15E.12-1) values are calculated within the code.
(3) These input parameters and initial conditions pertain to the initial cycle MEOD analysis based on initial cycle safety limit MCPR of 1.06. The <Appendix 15B>, Reload Safety Analysis provides the values for the current cycle.

TABLE 15E.3-2

Input Parameters and Initial Conditions for[3]
Transients and Accidents for MEOD, 3729 MWt Core Power, 105% Flow

1.	Thermal Power Level, MWt Analysis Value	3,729.3 (104.2% rated)
2.	Steam Flow, lb per sec Analysis Value	4,468 (104.4% rated)
3.	Core Flow, lb per hr	109.2×10^6
4.	Feedwater Flow Rate, lb per sec Analysis Value	4,485
5.	Feedwater Temperature, °F	425
6.	Vessel Dome Pressure, psig	1,045
7.	Core Exit Pressure, psig	1,057
8.	Turbine Bypass Capacity, % NBR	35[1]
9.	Core Coolant Inlet Enthalpy Btu per lb	529.4
10.	Turbine Inlet Pressure, psig	960
11.	Fuel Lattice	P8x8R
12.	Core Leakage Flow, %	12.9
13.	Required MCPR Operating Limit Initial Core	1.18
14.	MCPR Safety Limit for Incidents of Moderate Frequency	
	Initial Core	1.06
15.	Doppler Coefficient (-)¢/°F Analysis Data	0.132[2]
16.	Void Coefficient (-)¢/% Rated Voids Analysis Data for Power Increase Events	14.0[2]
	Analysis Data for Power Decrease Events	4.0[2]

TABLE 15E.3-2 (Continued)

17.	Core Average Rated Void Fraction, %	41.7
18.	Jet Pump Ratio, M	2.25
19.	Safety/Relief Valve Capacity, % NBR	
	@1,210 psig	111.4
	Manufacturer	Dikker
	Quantity Installed	19
20.	Relief Function Delay, seconds	0.4
21.	Relief Function Response Time Constant, seconds	0.1
22.	Analyses Inputs for Safety/Relief Valves	
	Safety Function, psig	1,175, 1,185, 1,195 1,205, 1,215
	Relief Function, psig	1,145, 1,155, 1,165, 1,175
23.	Number of Valve Groupings Simulated	
	Safety Function, No.	5
	Relief Function, No.	4
24.	High Flux Trip, % NBR Analysis	
	Setpoint (122x1.042), % NBR	127.2
25.	High Pressure Scram Setpoint, psig	1,095
26.	Vessel Level Trips, Feet Above Separator Skirt Bottom	
	Level 8 - (L8), feet	5.89
	Level 4 - (L4), feet	4.04
	Level 3 - (L3), feet	2.165
	Level 2 - (L2), feet	(-) 1.739
27.	APRM Thermal Trip Setpoint, % NBR	118.8
28.	RPT Delay, seconds	0.14
29.	RPT Inertia Time Constant for Analysis, seconds	5
30.	Total Steamline Volume, ft^3	3,850

TABLE 15E.3-2 (Continued)

NOTES:

(1) See <Section 10.2.1> and <Table 15E.3-4>.

(2) These values for (Reference 15E.12-4) analysis only. (Reference 15E.12-1) values are calculated within the code.

(3) These input parameters and initial conditions pertain to the initial cycle MEOD analysis based on initial cycle safety limit MCPR of 1.06. The <Appendix 15B>, Reload Safety Analysis provides the values for the current cycle.

TABLE 15E.3-3

Summary of Transient Peak Values Results - 3729 MWt Core Power - MEOD[1]

Transient	Core Flow (% NBR)	Peak Neutron Flux (% NBR)[1]	Peak Dome Pressure (psig)	Peak Vessel Pressure (psig)	Peak Steamline Pressure psig	Figure
Load Rejection With Bypass Failure	107.2[2]	268	1,217	1,251	1,217	-
Load Rejection With Bypass Failure	105.0	259	1,217	1,251	1,217	15E.3-1
Load Rejection With Bypass Failure	73.6	171	1,220	1,243	1,222	15E.3-2
Feedwater Controller Failure, Max. Demand	107.2[2]	149	1,172	1,203	1,172	-
Feedwater Controller Failure, Max. Demand	105.0	148	1,172	1,203	1,171	15E.3-3

TABLE 15E.3-3 (Continued)

Transient	Core Flow (% NBR)	Peak Neutron Flux (% NBR)[1]	Peak Dome Pressure (psig)	Peak Vessel Pressure (psig)	Peak Steamline Pressure psig	Figure
Feedwater Controller Failure, Max. Demand	73.6	111	1,175	1,195	1,172	15E.3-4

NOTES:

(1) Feedwater is at 425°F.
(2) Maximum achievable core flow with 425°F feedwater.

TABLE 15E.3-4

Summary of CPR Results - 3729 MWt Core Power MEOD[1]

Transient	Core Flow (% NBR)	ICPR[4]	ΔCPR[4]	MCPR[4]
Load Rejection With Bypass Failure	107.2[2]	1.18	0.11	1.07
Load Rejection With Bypass Failure	105.0	1.18	0.11	1.07
Load Rejection With Bypass Failure	73.6	1.27	0.06	1.21
Feedwater Controller Failure, Max. Demand	107.2[2]	1.18	0.09[3]	1.09
Feedwater Controller Failure, Max. Demand	105.0	1.18	0.09[3]	1.09
Feedwater Controller Failure, Max. Demand	73.6	1.27	0.09[3]	1.18
Loss of Feedwater Heating (100°F)	105.0	1.18	0.07	1.11
Loss of Feedwater Heating (100°F)	100.0	1.18	0.07	1.11

TABLE 15E.3-4 (Continued)

Transient	Core Flow (% NBR)	ICPR[4]	ΔCPR[4]	MCPR[4]
Loss of Feedwater Heating (100°F)	75.0	1.27	0.09	1.18

NOTES:

[1] Feedwater Temperature is 425°F

[2] Maximum Achievable Core Flow With 425°F Feedwater

[3] Analysis has been performed to conclude that turbine bypass capacity as low as 25% NBR does not affect the bounding ΔCPR results (Reference 15E.12-9).

[4] The required initial MCPR, ΔCPR, and resulting end of transient MCPRs represent initial cycle results based on initial cycle safety limit MCPR of 1.06. Results for the limiting transients for the current reload cycle at bounding conditions, are presented in <Appendix 15B>, Reload Safety Analysis.

TABLE 15E.3-5

Analysis Power-Flow Points for
Perry Initial Cycle Bounding Transient Evaluation

Power (%)/Flow (%)	Transients[1]
70/40	LRNBP FWCF CLDLP FCVO
83/55	LRNBP
104.2/75	LRNBP FWCF FCVO
104.2/100	LRNBP FWCF
104.2/110	LRNBP FWCF
53.5/116	LRNBP FWCF

NOTE:

[1] 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 15E.12-1) and the CLDLP and FCVO transients are analyzed using (Reference 15E.12-4).

The CLDLP assumes a 50°F delta temperature between the active and idle recirculation loops. This limit ensures not only thermal stresses but also thermal limits are not exceeded during an idle loop start event (Reference 10).

15E.4 FUEL INTEGRITY - STABILITY

The General Electric company has established stability criteria to demonstrate compliance to requirements set forth in <10 CFR 50, Appendix A>, General Design Criteria (GDC) 10 & 12. These stability compliance criteria consider potential limit cycle response within the limits of safety system or operator intervention and assure that for GE 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 stability compliance of those GE BWR fuel designs contained in the General Electric Standard Application for Reactor Fuel (Reference 15E.12-6) is demonstrated on a generic basis in (Reference 15E.12-5) (for operation in the normal as well as the extended operating domain). The NRC has reviewed and approved this in (Reference 15E.12-8) therefore, a specific analysis for each cycle is not required.

For operation in the Maximum Extended Operating Domain (MEOD) the stability margin (defined by the core decay ratio) is reduced as power increases for a given core flow. However, the normal realistic operating region has the lowest stability margin at the rated pump speed/minimum valve position flow (minimum forced circulation) which corresponds to about 43% core flow for PNPP <Figure 15E.2-1>. Operating at this core flow relative to natural circulation results in adequate stability margin for the maximum extended operating domain as demonstrated by tests at operating BWRs. Inadvertent operation below minimum forced circulation flows can only occur during transients, e.g., two recirculation pump trip. Operation in the high power/low core flow corner of the power flow map is addressed in a set of GE operating recommendations (Reference 15E.12-7) which have been approved by NRC and will be utilized at PNPP.

Stability tests were performed in the MEOD region in September and October of 1984 at an overseas BWR6 plant during the initial cycle startup testing. The test objectives were to obtain stability data at high power/flow ratios, to obtain data at reduced feedwater temperature conditions and to evaluate the core behavior when the plant was operated beyond the inception point of limit cycle oscillations. The tests were conducted under a range of power/flow conditions and data were recorded during the approach to limit cycles, during limit cycles, and during core flow changes once limit cycles were achieved. The oscillations observed during the test conditions were compared to those analyzed in (Reference 15E.12-5) and are shown to be bounded by the analyses. Therefore, consistent with the analyses of (Reference 15E.12-5), large margin to thermal/mechanical limits existed during the test conditions.

15E.5 LOSS-OF-COOLANT ACCIDENT ANALYSES

Refer to <Appendix 15B> for a description of the LOCA analysis relative to MEOD.

15E.6 CONTAINMENT RESPONSE ANALYSIS

A containment response analysis is performed with the most limiting postulated recirculation piping line break at the most limiting condition of MEOD for PNPP. The results show that the peak drywell pressure and the peak containment pressure and temperature are still lower than the reported in <Chapter 6>.

15E.7 LOAD IMPACT ON INTERNALS

15E.7.1 ACOUSTIC AND FLOW INDUCED LOADS

The acoustic loads are loads on the vessel internals from propagation of the decompression wave created by a sudden vessel nozzle break. The acoustic loading on the vessel internals is proportional to the total pressure wave amplitude in the vessel upon the postulated break. The additional subcooling in the downcomer resulting from operating in the increased core flow region of MEOD leads to an increase in critical flow and, therefore, in flow induced loads. However, the maximum subcooling in the MEOD is less than the partial feedwater heating operation described in <Appendix 15D.6>. Therefore, it is concluded, based on analysis presented in <Appendix 15D.6>, the reactor internal components have enough margin to handle the acoustic and flow induced loads.

15E.7.2 REACTOR INTERNAL PRESSURE DIFFERENCE LOADS

A reactor internals pressure difference analysis is performed for the increased core flow region 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 for the reactor internal impact evaluation.

15E.7.3 IMPACT ON REACTOR INTERNALS

The reactor internals most affected by pressure under increased core flow conditions are the core plate, guide tube, shroud head, upper shroud, lower shroud, shroud support ring, shroud top guide, fuel channel wall, grid, 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 limits.

15E.8 FLOW INDUCED VIBRATIONS

To ensure that the flow-induced vibration response of the reactor internals is acceptable, Perry must undergo an extensive vibration test during initial plant startup in accordance with <Regulatory Guide 1.20>. PNPP startup test procedure will include flow induced vibration tests at increased core flow region to verify safe operation as defined in this appendix.

15E.9 IMPACT ON ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)

An ATWS performance impact evaluation was performed for PNPP in the MEOD based on the MSIV closure event which is the limiting transient for ATWS. Results show that all ATWS consequences from initial MEOD conditions satisfy the criteria for ATWS in <Appendix 15C>. <Table 15E.9-1> lists the initial conditions for the analysis. <Table 15E.9-2> summarizes the peak values for the MSIV closure event. Furthermore, analysis was also conducted to assure that there is no unacceptable stability consequences if an ATWS were postulated from initial MEOD operating conditions. Therefore, it is concluded that MEOD operation is acceptable from ATWS requirements including ATWS stability considerations.

TABLE 15E.9-1

Initial Conditions Table for MSIV Closure Event

	MEOD
Power (% NBR)	100
Core Flow (% NBR)	81.0
Dome Pressure (psia)	1,040
Dynamic Void Coefficient (cents/%)	-13.0
Void Fraction (%)	51.8
Inlet Subcooling (BTU/lb)	31

TABLE 15E.9-2

Peak Value Summary Table for MSIV Closure Event

	MEOD (100/81)	Design Limit
Peak Neutron Flux (%)	262	N/A
Peak Vessel Pressure (psia)	1,284	1500
Peak Suppression Pool Temp. (°F)	178	190
Peak Average Heat Flux (%)	135	N/A
Peak Clad Temperature (°F)	See note(1)	2200
Maximum Local Oxidation Fraction (%)	See note(1)	≤17

NOTE:

(1) The Peak Clad Temperature and Maximum Local Oxidation Fraction values are dispositioned generically in (Reference 12-12).

15E.10 FUEL MECHANICAL PERFORMANCE

Evaluations were performed to determine the acceptability of PNPP MEOD operation on GE fuel rod and assembly thermal/mechanical performance. Component pressure differentials 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, PNPP 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.

15E.11 PARTIAL FEEDWATER HEATING (PFH) OPERATION IN THE MAXIMUM EXTENDED OPERATING DOMAIN

NOTE: This section reflects the original analysis (i.e., 100% power = 3579 MWt and 104.2% power = 3729 MWt) prior to Power Uprate.

This section presents the results of the safety evaluation for operation of PNPP with partial feedwater heating at steady-state condition during the operating cycle in the entire MEOD region and beyond the end of cycle in the Increased Core Flow Region (ICFR) of the MEOD as illustrated in <Figure 15E.2-1>. The evaluation is performed for the GE6 fueled PNPP on a equilibrium cycle basis and is applicable to initial and reload cycles operation. Initial conditions for the reload safety analysis including MEOD operation with PFH for the limiting transients are consistent with the GEMINI set of methods described in GESTAR (Reference 15E.12-6). For the current cycle initial conditions refer to <Appendix 15B>, Reload Safety Analysis. The conditions of operation are those of continued 100% thermal power operation during and beyond the operating cycle with rated feedwater temperature ranging from 420°F to 320°F in the entire MEOD region and beyond the operating cycle with rated feedwater temperature ranging from 320°F to 250°F in the increased core flow region up to 105% core flow.

All the impact evaluation described in <Appendix 15D> were reevaluated or reviewed in the entire MEOD for PFH operation. Most conclusions made in <Appendix 15D> are directly applicable to the PFH operation in the extended region except the abnormal operating transients for which additional increase in operating limit MCPR values was required for PFH operation in the extended regions for the initial cycle. MEOD operation with PFH is reevaluated for the limiting transients as part of the reload safety analysis <Appendix 15B>. <Table 15E.2-1> summarizes the required operating limit MCPR values for PFH modes of operation in various operating regions.

15E.11.1 ABNORMAL OPERATING TRANSIENTS

Two limiting abnormal operating transients (Load Rejection With Bypass Failure and Feedwater Controller Failure) were reevaluated for PFH operation in the extended operating region. Furthermore, the PNPP plant specific analysis using the (Reference 15E.12-2) model was also performed for the 100°F Loss of Feedwater Heating event for PFH in the extended operating regions.

The Load Rejection with Bypass Failure and Feedwater Controller Failure events were reanalyzed at: 250°F rated feedwater temperature beyond the end of cycle with 105% rated core flow, 370°F and 320°F rated feedwater temperatures at the end of cycle and also 2,000 MWd/t exposure before end of cycle in the MEOD condition. Consistent assumptions and initial conditions in <Appendix 15D> are used in the analyses. The transient peak value results are summarized in <Table 15E.11-1> to <Table 15E.11-3>. The Critical Power Ratio (CPR) results are summarized in <Table 15E.11-4>, <Table 15E.11-5>, and <Table 15E.11-6>. The transient responses for the most limiting end of equilibrium cycle cases are presented in <Table 15E.11-6>.

In <Appendix 15D.2.1> it was concluded that the 100°F LFWH event is less severe when initiated from lower initial feedwater temperature than from the rated feedwater temperature. The analysis results in the extended operating regions also show that this trend is not affected by initial core flow. Thus, the LFWH analysis for PFH operation in the MEOD is adequately bounded by the 420°F ΔCPR results given in <Table 15E.3-4>.

The results of the evaluations show that the ΔCPRs for both the Load Rejection with Bypass Failure event and the Feedwater Controller Failure event exceeded the standard operating limit basis for the initial cycle analysis. Therefore, a rated operating limit MCPR value of 1.19 (1.20 for reload cores) was required for PFH operation during the initial operating cycle and beyond the end of cycle in the MEOD for rated feedwater temperature in the range of 420°F and 320°F. However, a rated operating limit MCPR value of 1.21 (1.22 for reload cores) was required for PFH operation beyond the end of the initial cycle in the increased core flow region (core flow >100% rated) for rated feedwater temperature in the range of 320°F to 250°F. Operating limit MCPR values for the current cycle are presented in <Appendix 15B>, Reload Safety Analysis.

15E.11.2 OTHER EVALUATIONS, PFH OPERATION IN MEOD

All other evaluations described in <Appendix 15D> for PFH are directly applicable to the PFH operation in the extended operating Region. The 100°F Loss of Feedwater Heating results are bounding for MEOD conditions. The rod withdrawal error analysis described in <Appendix 15D> is directly applicable to the MEOD because the bounding generic RWE analysis is performed based on the MEOD. The stability criteria are met at the PFH condition in the MEOD. The stability discussion in <Appendix 15D.3> and <Appendix 15E.4> are directly applicable to PFH operation in the MEOD. Both Loss-of-Coolant Accident and Containment Response Analysis are shown to be bounded by the design values of <Chapter 6>. Acoustic and flow induced loads on the vessel internal are also demonstrated to be within limits. Feedwater nozzle, sparger fatigue and system piping are independent of core flow rates. Impact on ATWS, annulus pressurization loads and fuel mechanical performance are also shown to be acceptable for PFH operation in the extended operating region.

TABLE 15E.11-1

Summary of Transient Peak Values Results - Partial Feedwater Heating in MEOD Beyond End of Equilibrium Cycle[(1)]

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	105.0	286	1,192	1,220	1,194	252
Feedwater Controller Failure, Max. Demand	105.0	191	1,128	1,151	1,127	252

NOTE:

(1) Initial power and heat flux is 104.2% NBR for analysis. Rated feedwater temperature is 250°F.

TABLE 15E.11-2

Summary of Transient Peak Values Results - Partial Feedwater Heating in MEOD Beyond End of Equilibrium Cycle[1]

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	108.7[2]	304	1,203	1,233	1,211	373
Load Rejection With Bypass Failure	100.0	245	1,202	1,230	1,209	373
Load Rejection With Bypass Failure	74.8	172	1,205	1,225	1,212	373
Feedwater Controller Failure, Max. Demand	108.7[2]	163	1,159	1,187	1,159	373
Feedwater Controller Failure, Max. Demand	100.0	144	1,160	1,187	1,158	373

TABLE 15E.11-2 (Continued)

Transient	Core Flow (% NBR)	Peak Neutron Flux (% NBR) [1]	Peak Dome Pressure (psig)	Peak Vessel Pressure (psig)	Peak Steamline Pressure (psig)	Fdwtr. Temp. (°F)
Feedwater Controller Failure, Max. Demand	74.8	116	1,160	1,179	1,160	373
Load Rejection With Bypass Failure	110.0 [2]	303	1,199	1,227	1,201	322
Load Rejection With Bypass Failure	100.0	246	1,198	1,224	1,201	322
Load Rejection With Bypass Failure	73.7	162	1,200	1,218	1,201	322
Feedwater Controller Failure, Max. Demand	110.0 [2]	219	1,151	1,177	1,151	322
Feedwater Controller Failure, Max. Demand	100.0	130	1,145	1,167	1,145	322

TABLE 15E.11-2 (Continued)

Transient	Core Flow (% NBR)	Peak Neutron Flux (% NBR) [1]	Peak Dome Pressure (psig)	Peak Vessel Pressure (psig)	Peak Steamline Pressure (psig)	Fdwtr. Temp. (°F)
Feedwater Controller Failure, Max. Demand	73.7	120	1,149	1,167	1,148	322

NOTE:

[1] Initial power and heat flux is 104.2% NBR for analysis. Rated feedwater temperature 370°F and 320°F.
[2] Maximum achievable core flow for the given feedwater temperature.

TABLE 15E.11-3

Summary of Transient Peak Value Results - Partial Feedwater Heating in MEOD
2,000 MWd/t Before End of Equilibrium Cycle[1]

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	108.7	104.2	1,189	1,217	1,191	373
Load Rejection With Bypass Failure	74.8	104.3	1,192	1,212	1,196	373
Feedwater Controller Failure, Max. Demand	108.7	116.7	1,139	1,162	1,137	373
Feedwater Controller Failure, Max. Demand	74.8	115.3	1,149	1,167	1,148	373
Load Rejection With Bypass Failure	110.0	104.2	1,182	1,206	1,191	322

TABLE 15E.11-3 (Continued)

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	73.7	104.3	1,189	1,209	1,190	322
Feedwater Controller Failure, Max. Demand	110.0	122.6	1,117	1,141	1,116	322
Feedwater Controller Failure, Max. Demand	73.7	119.5	1,125	1,142	1,124	322

NOTE:

(1) Initial power and heat flux is 104.2% NBR for analysis. Rated feedwater temperature 370°F and 320°F.

TABLE 15E.11-4

Summary of CPR Results - Partial Feedwater Heating in MEOD Beyond End of Equilibrium Cycle[1]

Transient	Core Flow (% NBR)	ICPR[2]	ΔCPR[2]	MCPR[2]	Fdwtr. Temp. (°F)
Load Rejection With Bypass Failure	105.0	1.19[3]	0.13	1.06	252
Feedwater Controller Failure, Max. Demand	105.0	1.21[3]	0.15	1.06	252

NOTE:

(1) Initial power and heat flux is 104.2% NBR for analysis. Rated feedwater temperature is 250°F.
(2) The required initial MCPR, ΔCPR, and resulting end of transient MCPRs represent initial cycle results. Results for the limiting transients for the current reload cycle at bounding conditions, are presented in <Appendix 15B>, Reload Safety Analysis.
(3) Requires operating limit CPR change.

TABLE 15E.11-5

Summary of CPR Results - Partial Feedwater Heating in MEOD Beyond End of Equilibrium Cycle[1]

Transient	Core Flow (% NBR)	ICPR[1][3][4]	ΔCPR[4]	MCPR[4]	Fdwtr. Temp. (°F)
Load Rejection With Bypass Failure	108.7[2]	1.19[5]	0.13	1.06	373
Load Rejection With Bypass Failure	100	1.18	0.10	1.08	373
Load Rejection With Bypass Failure	74.8	1.27	0.06	1.21	373
Feedwater Controller Failure, Max. Demand	108.7[2]	1.18	0.11	1.07	373
Feedwater Controller Failure, Max. Demand	100	1.18	0.11	1.07	373
Feedwater Controller Failure, Max. Demand	74.8	1.27	0.11	1.16	373

TABLE 15E.11-5 (Continued)

Transient	Core Flow (% NBR)	ICPR[1][3][4]	ΔCPR[4]	MCPR[4]	Fdwtr. Temp. (°F)
Load Rejection With Bypass Failure	110.0[2]	1.19[5]	0.13	1.06	322
Load Rejection With Bypass Failure	110.0	1.18	0.11	1.07	322
Load Rejection With Bypass Failure	73.7	1.27	0.06	1.21	322
Feedwater Controller Failure, Max. Demand	110.0[2]	1.19[5]	0.13	1.06	322
Feedwater Controller Failure, Max. Demand	100	1.19[5]	0.13	1.06	322

TABLE 15E.11-5 (Continued)

Transient	Core Flow (% NBR)	ICPR[(1)(3)(4)]	ΔCPR[(4)]	MCPR[(4)]	Fdwtr. Temp. (°F)
Feedwater Controller Failure, Max. Demand	73.7	1.27	0.12	1.15	322

NOTES:

(1) Initial power and heat flux is 104.2% NBR for analysis. Rated feedwater temperature 370°F and 320°F.
(2) Maximum achievable core flow for the given feedwater temperature.
(3) Based on initial core safety limit of 1.06.
(4) The required initial MCPR, ΔCPR, and resulting end of transient MCPRs represent initial cycle results based on initial cycle safety limit MCPR of 1.06. Results for the limiting transients for the current reload cycle at bounding conditions, are presented in <Appendix 15B>, Reload Safety Analysis.
(5) Requires operating limit CPR change.

TABLE 15E.11-6

Summary of CPR Results - Partial Feedwater Heating in MEOD
2,000 MWd/t Before End of Equilibrium Cycle[1]

Transient	Core Flow (% NBR)	ICPR[2]	ΔCPR[2]	MCPR[2]	Fdwtr. Temp. (°F)
Load Rejection With Bypass Failure	108.7	1.18	0.05	1.13	373
Load Rejection With Bypass Failure	74.8	1.27	0.05	1.22	373
Feedwater Controller Failure, Max. Demand	108.7	1.18	0.11	1.07	373
Feedwater Controller Failure, Max. Demand	74.8	1.27	0.11	1.16	373
Load Rejection With Bypass Failure	110.0	1.18	0.05	1.13	322
Load Rejection With Bypass Failure	73.7	1.27	0.05	1.22	322
Feedwater Controller Failure, Max. Demand	110.0	1.19[3]	0.13	1.06	322

TABLE 15E.11-6 (Continued)

Transient	Core Flow (% NBR)	ICPR[2]	ΔCPR[2]	MCPR[2]	Fdwtr. Temp. (°F)
Feedwater Controller Failure, Max. Demand	73.7	1.27	0.12	1.15	322

NOTE:

[1] Initial power and heat flux is 104.2% NBR for analysis. Rated feedwater temperature 370°F and 320°F.
[2] The required initial MCPR, ΔCPR, and resulting end of transient MCPRs represent initial cycle results based on initial cycle safety limit MCPR of 1.06. Results for the limiting transients for the current reload cycle at bounding conditions, are presented in <Appendix 15B>, Reload Safety Analysis.
[3] Requires operating limit CPR change.

15E.12 REFERENCES

15E.12-1 “Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors” NEDO-24154, October 1978.

15E.12-2 “Three Dimensional BWR Core Simulator” NEDO-20953-A, January 1977.

15E.12-3 (Deleted)

15E.12-4 R. B. Linford “Analytical Methods of Plant Transients Evaluations for General Electric Boiling Water Reactors” NEDO-10802, April 1973.

15E.12-5 “Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria” NEDE-22277-P-1, October 1984.

15E.12-6 General Electric Company “General Electric Standard Application for Reactor Fuel” including the United States Supplement, NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision).

15E.12-7 “BWR Core Thermal Hydraulic Stability,” SIL No. 380 Revision 1, February 10, 1984.

15E.12-8 Acceptance for Referencing of Licensing Topical Report NEDE-24011 Revision 6, Amendment 8, “Thermal Hydraulic Stability Amendment to GESTAR II” NRC’s C. O. Thomas to H. C. Pfefferlen, April 24, 1985.

15E.12-9 Letter, David T. Shen (GE) to Eileen Buzzelli "Perry Transient Analysis Assuming 25% Steam Bypass," October 28, 1987.

NOTE: (Reference 15E.12-5) above is the principal supporting document for GE's Topical Report NEDE-24011 Amendment 8.

15E.12-10 GE Services Information Letter, SIL No. 517 Supplement 1, "Analysis Basis for Idle Recirculation Loop Startup."

15E.12-11 "Perry Nuclear Power Plant Asset Improvement Project Task G1-02: Reactor Power/Flow Map," GE-NE-A2200084-02-01-R0.

15E.12-12 NEDC-33270P, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," Revision 5, May 2013.

<APPENDIX 15F>

PNPP SINGLE LOOP OPERATION ANALYSIS

APPENDIX 15F

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

LIST OF TABLES

15F RECIRCULATION SYSTEM SINGLE LOOP OPERATION

This appendix justifies that PNPP can safely operate with a single recirculation loop up to approximately 70% of rated thermal power. This appendix presents the results of a safety evaluation for operation of the Perry Nuclear Power Plant (PNPP) with a single recirculation loop operating. This evaluation was performed for PNPP on an equilibrium cycle basis with the GE6 (8x8R) fuel design and is applicable to initial and reload cycles operation with normal feedwater heating and within the standard operating domain shown in <Figure 4.4-2>. The applicability of single loop operation for each cycle is reverified as part of the Reload Safety Analysis, <Appendix 15B>.

15F.1 INTRODUCTION AND SUMMARY

Single loop operation (SLO) is desirable in the event recirculation pump or other component maintenance renders one loop inoperative. To justify single loop operation, accidents and abnormal operational transients associated with power operation, as presented in <Chapter 6> and <Chapter 15>, were reviewed with one recirculation pump in operation. The safety limit minimum critical power ratio (SLMCPR) is determined using the General Electric Thermal Analysis Basis (GETAB), a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. During SLO, the total core flow and TIP readings uncertainty input to the GETAB model is slightly increased resulting in a small increase in the safety limit MCPR.

The results of the evaluation are summarized below:

a. Accounting for the uncertainties in the core total flow and Traversing In-Core Probe (TIP) readings, an increase is required in the minimum critical power ratio (MCPR) fuel cladding integrity safety limit during single loop operation. For the current cycle MCPR safety limit value refer to <Table 15B.15.0-1>.

b. For the current cycle MCPR operating limit value refer to <Table 15B.15.0-3>.

c. The recirculation flow rate dependent rod block and scram setpoint equations given in the Technical Specifications for two loop operation must be adjusted for one-pump operation.

d. Thermal-hydraulic stability was evaluated for its adequacy with respect to General Design Criterion (GDC) 12 <10 CFR 50, Appendix A>. The increase in neutron noise observed during SLO is independent of system stability margin. For the initial fuel cycles, operation with a single recirculation loop was acceptable based on avoiding the region of potential instability by adherence to the recommendations of SIL-380. For the current fuel cycle, stability is addressed in <Appendix 15B.4.4>.

e. The Maximum Average Planar Linear Heat Generation Rate curves for two loop operation in the Core Operating Limits Report must be reduced (SLO MAPLHGR reduction factor of 0.84 for the initial cycle) for single loop operation. The SLO MAPLHGR reduction factor is adjusted for each cycle as necessary, based on the results of the reload safety analysis. See <Appendix 15B> for the current cycle value.

f. 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.

g. The consequences of single loop operation on the Anticipated Transient Without Scram (ATWS) analysis are bounded by those from two loop operation.

h. It was determined that the fuel will operate within the fuel mechanical design bases during normal steady-state operation and anticipated operational occurrences during single recirculation loop operation.

i. A recirculation pump drive flow limit of 48,500 gpm is imposed for SLO. This is based on the analysis of reactor internals vibration testing data obtained during the startup test program at PNPP.

15F.1.1 SINGLE LOOP OPERATION POWER/FLOW OPERATING DOMAIN

The power/flow map for two recirculation loop operation within the standard operating domain is shown in <Figure 4.4-2>. The single recirculation loop operation power/flow operating domain is based on this map but is smaller due to restrictions placed on the maximum thermal power, core flow and avoiding jet pump cavitation while operating in the SLO mode. The boundaries for single loop operation are as follows:

a. Natural Circulation Line: The operating state of the reactor moves along this line for the normal control rod withdrawal sequence in the absence of recirculation pump operation.

b. 105% Steam Flow Rod Line or Approximately 70% of Rated Power (whichever is less): The 105% steam flow rod line passes through 104.2% power at 100% flow. The operating state of the reactor follows this rod line (or similar ones) during recirculation flow changes with a fixed control rod pattern; however, 2500 Megawatts-Thermal power (or approximately 70% of rated thermal power) may not be exceeded.

c. Maximum Core Flow from a Single Operating Recirculation Loop or the Maximum Core Flow allowed by Reactor Internals Vibration Considerations (whichever is less): The operating state of the reactor moves along this line for the normal control rod withdrawal sequence at the maximum core flow capable under single recirculation loop operation; however, the recirculation loop drive flow limit of 48,500 gpm determined from the reactor internals vibration analysis results for SLO may not be exceeded and operation may not occur below the cavitation protection line (see item d).

d. Cavitation Protection Line: This line results from the recirculation pump, flow control valve, and jet pump NPSH requirements during single recirculation loop operation.

15F.2 MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

The safety limit minimum critical power ration (SLMCPR) is determined using the General Electric Thermal Analysis Bases (GETAB) which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. Except for core total flow and TIP reading, the uncertainties used in 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 USAR. 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 15F.8-1). 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 <Appendix 15F.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. An analysis was performed to show the net effect of these two revised uncertainties is an increase in the required MCPR fuel cladding integrity safety limit. For the current cycle MCPR safety limit value refer to <Table 15B.15.0-1>.

15F.2.1 CORE FLOW UNCERTAINTY

15F.2.1.1 Core Flow Measurement During Single Loop Operation

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 (when the active loop pump flow is above approximately 35%). 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:

$$\begin{matrix}\text{Total Core} \\ \text{Flow}\end{matrix} = \begin{matrix}\text{Active Loop} \\ \text{Indicated Flow}\end{matrix} - C \begin{matrix}\text{Inactive Loop} \\ \text{Flow}\end{matrix}$$

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. The analytical expected value of the "C" coefficient for PNPP is ~0.83.

15F.2.1.2 Core Flow Uncertainty Analysis

The uncertainty analysis procedure used to establish the core flow uncertainty for one-pump operation is essentially the same as for two-pump operation, with some exceptions. The core flow uncertainty analysis is described in (Reference 15F.8-1). The analysis of one-pump core flow uncertainty is summarized below.

For single loop operation, the total core flow can be expressed as follows <Figure 15F.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

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

$$\sigma^2_{W_C} = \sigma^2_{W_{sys}} + \left(\frac{1}{1-a}\right)^2 \sigma^2_{W_{Arand}} + \left(\frac{a}{1-a}\right)^2 \left(\sigma^2_{W_{Irand}} + \sigma^2_C\right)$$

where:

σ_{W_C} = uncertainty of total core flow;

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

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

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

σ_C = uncertainty of "C" coefficient; and

a = ratio of inactive flow (W_I) to active loop flow (W_A).

From an uncertainty analysis, the conservative, bounding values of

$\sigma_{W_{sys}}$, $\sigma_{W_{Arand}}$ $\sigma_{W_{Irand}}$ and σ_C are 1.6%, 2.6%, 3.5%, and 2.8%.

respectively. Based on the above uncertainties and a bounding value of 0.36[1] for "a," the variance of the total flow uncertainty is approximately:

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

$$= (5.0\%)^2$$

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

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

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.

<u>NOTE</u>:

[1] 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 PNPP is ~0.28.

15F.2.2 TIP READING UNCERTAINTY

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 of 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.

There would be no significant difference in the TIP noise seen at 59.3% of rated thermal power and the TIP noise values that would be expected for SLO at 70% of rated thermal power. This is because the major contributors to the TIP noise, as determined in previous GE generic BWR analyses, are geometric mislocation of the TIP detector and neighboring fuel channels with respect to their nominal design positions, and the random neutron, electronic and boiling noise in the reactor, as discussed in the GE licensing topical report, "Process Computer Performance Evaluation Accuracy" (Reference 15F.8-12). Additionally, the generic analyses were performed with neutron TIP detectors which are much more sensitive to geometric mislocation than the gamma TIP detectors currently inservice at Perry. As a result the actual TIP noise uncertainty with gamma TIPs would even be less than the values assumed in the generic analyses.

15F.3 DETERMINATION OF THE MCPR OPERATING LIMIT

15F.3.1 ABNORMAL OPERATING TRANSIENTS

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 operational transients from one loop operation will be considerably less severe than those analyzed from a two loop operational mode. The pressurization, flow increase, flow decrease, and cold water injection transients results presented in <Chapter 15> bound both thermal and overpressure consequences of one loop operation.

<Figure 15F.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 the operating power level (about 30% less than during one loop operation).

The consequences of flow decrease transients are also bounded by the full power analysis. A single pump trip from one loop operation is less severe than a two-pump trip from full power because of the reduced initial power level. The one recirculation pump seizure accident has been reviewed for single loop operation. As described within USAR <Section 15.3.3> and Section S.2.2.3.4 of the GE topical report GESTAR II (Reference 15F.8-5) the recirculation pump seizure accident is very mild in relation to other accidents such as a design basis LOCA. Because the recirculation pump seizure event is classified as an accident based on low probability of occurrence, a failure to scram (an ATWS) is not required to be postulated to occur in addition to a recirculation pump seizure. However, redundant capability is available to insert control rods or inject borated water with the Standby Liquid Control System during a recirculation pump seizure without scram. Results show that this accident poses no threats to thermal limits.

The worst flow increase transient results from recirculation flow controller failure, and the worst cold water injection transient results from the sudden loss of feedwater heating. For the former, the $MCPR_f$ curve is derived assuming both recirculation loop flow controllers fail. This condition produces the maximum possible power increase and, hence, the maximum delta CPR for transients initiated from less than rated power and flow. When operating with only one recirculation loop, the flow and power increase associated with the recirculation flow controller failure with only one loop will be bounded by 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, sudden 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 heating 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 <Section 15.4.4> and is still applicable for single loop operation. The idle recirculation loop startup event discussed in <Section 15.4.4> and for the ATWS case discussed in the GE generic BWR ATWS evaluation document, "Assessment of BWR Mitigation of ATWS" (Reference 15F.8-13) were both based on power levels less than 70%. However, the event at the 70% power level is less severe than at the power levels assumed in these documents due to the power rise from beginning to peak being less for the 70% power case. Also, the relative core flow change that would be caused by an idle recirculation loop startup at 70% power is smaller because the reactor is already operating at a higher core flow than it would be at for lower power levels. The evaluations of (Reference 15F.8-13) are bounding and therefore, the idle recirculation loop startup event at 70% power need not be analyzed for ATWS.

From the above discussions, it is concluded that the transient consequences from one loop operation are bounded by the full power two loop analyses presented in the USAR. The maximum power level that can be attained with one loop operation is restricted by one pump operation flow capability, MCPR and the overpressure limits established from the full-power analysis.

In the following sections, two of the most limiting transients are discussed for single loop operation. They are:

a. feedwater flow controller failure (maximum demand), (FWCF)

b. generator load rejection with bypass failure, (LRNBP),

The plant initial conditions are given in <Table 15F.3-1>.

15F.3.1.1 Feedwater Controller Failure - Maximum Demand

15F.3.1.1.1 Core and System Performance

Mathematical Model

The computer model described in (Reference 15F.8-2) was used to simulate this event.

Input Parameters and Initial Conditions

The analysis has been performed with the plant conditions tabulated in <Table 15F.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 is reached resulting in higher heat fluxes. The initial conditions used in the analysis are consistent with those in <Table 15.0-1> except those which are related to SLO.

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 was simulated by programming an upper limit failure in the feedwater system such that 130% of rated feedwater flow (143% is assumed in the current cycle reload analyses based on startup test results) occurs at the reference pressure of 1,065 psig.

<u>Results</u>

The simulated feedwater controller transient is shown in <Figure 15F.3-2> for the case of 69.9% power 53.2% core flow. The high-water level turbine trip and feedwater pump trip are initiated at approximately 4.4 seconds. Scram occurs simultaneously from Level 8, and limits the peak neutron flux. The turbine bypass system opens to limit peak pressure in the steamline near the safety valves to 1,046 psig and the pressure at the bottom of the vessel to about 1,059 psig. MCPR is considerably above the safety limit.

15F.3.1.2 <u>Generator Load Rejection With Bypass Failure</u>

15F.3.1.2.1 Core and System Performance

<u>Mathematical Model</u>

The computer model described in (Reference 15F.8-2) was used to simulate this event.

<u>Input Parameters and Initial Conditions</u>

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in <Table 15F.3-1>.

The turbine electro-hydraulic control system (EHC) power/load inbalance 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 seconds. Full arc mode is more limiting than partial arc.

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 recirculation 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 15F.3-3>.

<Table 15F.3-2> shows for the case of bypass failure, peak neutron flux reaches about 70.1% of rated and peak pressure at the valves reaches 1,169 psig. The peak nuclear system pressure reaches 1,180 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1,375 psig. The calculated MCPR is well above the safety limit.

15F.3.1.3 Summary and Conclusions

The peak value results of the above abnormal transients are summarized in <Table 15F.3-2>. The Critical Power Ratio (CPR) results are summarized in <Table 15F.3-3>. This table indicates that for the most limiting transient events analyzed here, the MCPRs are well above the single loop operation initial cycle MCPR safety limit value of 1.07. For the current cycle MCPR safety limit value refer to <Table 15B.15.0-1>.

For pressurization events, lower initial pressure at lower power during single loop operation assures that the bounding MSIV flux scram event is bounded by the full power analysis. As expected, <Table 15F.3-2> also indicates the peak pressures from the two limiting transients are well below the ASME code value of 1,375 psig. Hence, it is concluded the pressure barrier integrity is maintained under single loop operation conditions.

15F.3.2 ROD WITHDRAWAL ERROR

The rod withdrawal error (RWE) transient for two loop operation documented in <Section 15.4.2> 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 Limiter (RWL) system. This analysis covered all off-rated conditions in the power/flow operating map with additional MCPR safety limit margin. For the current cycle MCPR operating limit value refer to <Table 15B.15.0-3>.

The Average Power Range Monitor (APRM) rod block system provides additional alarms and rod blocks when power levels are grossly exceeded. Since the APRM rod block setpoints are recirculation drive flow biased, modification of the APRM rod block equation in the Technical Specification is required to maintain the two loop rod block versus power relationship when in one loop operation. This is because the direct active-loop flow measurement may not indicate the actual core flow due to backflow through the inactive jet pump during single loop operation. Since the APRM scram trip settings are also recirculation drive flow biased, they are subject to the same modifications as the rod block settings.

15F.3.3 OPERATING LIMIT MCPR DISCUSSION

The increased uncertainties in total core flow and TIP readings result in an increase in the MCPR fuel cladding integrity safety limit during single loop operation. For single loop operation at off-rated conditions, the steady-state operating limit MCPR is established by the SLO $MCPR_p$ and $MCPR_f$ curves. This ensures that the 99.9% statistical limit requirement is always satisfied for any postulated abnormal operational occurrence.

For the current cycle MCPR operating limit value refer to <Table 15B.15.0-3>.

TABLE 15F.3-1

INPUT PARAMETERS AND INITIAL CONDITIONS FOR[4]
TRANSIENTS AND ACCIDENTS FOR SINGLE LOOP OPERATION

1.	Thermal Power Level Analysis Value, MWt	2,500 (69.9% Rated)
2.	Steam Flow, lb/sec	2,834
3.	Core Flow, lb/hr	55.3×10^6 (53.2% Rated)
4.	Feedwater Flow Rate, lb/sec	2,834
5.	Feedwater Temperature, °F	392
6.	Vessel Dome Pressure, psig	970
7.	Vessel Core Pressure, psig	974
8.	Turbine Bypass Capacity, % NBR	35[1]
9.	Core Coolant Inlet Enthalpy, Btu/lb	507.4
10.	Turbine Inlet Pressure, psig	934
11.	Fuel Lattice	P8x8R
12.	Core Leakage Flow, %	12.9
13.	Required MCPR Operating Limit First Core	1.42[2]
14.	MCPR Safety Limit for incident of Moderate frequency during SLO	
	First Core	1.07[5]
15.	Doppler Coefficient (-)¢/°F Analysis Data	0.132[3]
16.	Void Coefficient (-)¢/% Rated Voids Analysis Data for Power Decrease Events	4.0[3]
	Analysis Data for Power Increase Events	14.0[3]
17.	Core Average Void Fraction, %	46.1[3]

TABLE 15F.3-1 (Continued)

18.	Jet Pump Ratio, M	3.30
19.	Safety/Relief Valve Capacity, % NBR	
	1,210 psig	111.4
	Manufacturer	DIKKER
	Quantity Installed	19
20.	Relief Function Delay, Seconds	0.4
21.	Relief Function Response, Seconds	0.1
22.	Analyses Inputs for Safety/Relief Valves	
	Safety Function, psig	1,175, 1,185, 1,195, 1,205, 1,215
	Relief Function, psig	1,145, 1,155, 1,165, 1,175
23.	Number of Valve Groupings Simulated	
	Safety Function, No.	5
	Relief Function, No.	4
24.	High Flux Trip, % NBR Analysis	
	Setpoint (1.21x1.043), % NBR	127.2
25.	High Pressure Scram Setpoint, psig	1,095
26.	Vessel Level Trips, Feet Above Separator Skirt Bottom	
	Level 8 - (L8), Feet	5.89
	Level 4 - (L4), Feet	4.04
	Level 3 - (L3), Feet	2.165
	Level 2 - (L2), Feet	(-) 1.739
27.	APRM thermal trip	
	Setpoint, % NBR @ 100% Core Flow	118.8
28.	RPT Delay, Seconds	0.14
29.	RPT Inertia Time Constant for Analysis, Seconds	5
30.	Total Steamline Volume, ft^3	3,850

TABLE 15F.3-1 (Continued)

NOTES:

(1) See <Section 10.2.1> and <Table 15F.3-3>.

(2) This value corresponds to an operating limit MCPR of 1.18 at rated conditions.

(3) Parameters used in (Reference 15F.8-3) analysis only. (Reference 15F.8-2) values are calculated within the code for the end of cycle condition.

(4) These input parameters and initial conditions are those used for the single loop analysis and correspond to the initial cycle values. Single loop operation is bounded by the transient results of two loop operation for each cycle <Section 15F.3.1.3>, these values are representative of the values that would be used in later reload cycles if the single loop analysis were to be reperformed using the same analysis codes and methods.

(5) Value corresponds to initial cycle only. The single loop operation MCPR Safety Limit is calculated in the current reload analysis. Refer to <Table 15B.15.0.1> for the single loop value.

TABLE 15F.3-2

SUMMARY OF TRANSIENT PEAK VALUE RESULTS
SINGLE LOOP OPERATION

PARAGRAPH	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX (% NBR)	MAXIMUM DOME PRESSURE (psig)	MAXIMUM VESSEL PRESSURE (psig)	MAXIMUM STEAMLINE PRESSURE (psig)	FREQUENCY[1] Category
		Initial Condition	69.9	970	982	982	N/A
15F.3.1.1	15F.3.2	Feedwater flow Controller Failure (Maximum Demand)	83.7	1,047	1,059	1,046	a
15F.3.1.2	15F.3.3	Generator Load Rejection With Bypass Failure	70.1	1,167	1,180	1,169	b

NOTE:

[1] a = Moderate frequency incident
b = infrequent

TABLE 15F.3-3

SUMMARY OF CRITICAL POWER RATIO RESULTS - SINGLE-LOOP OPERATION

	FWCF	LRNBP
Initial Operating Condition (% power/% flow)	69.9/53.2[6]	69.9/53.2[6]
MCPR Operating Limit at SLO Conditions (Initial Condition)	1.42[1][5]	.42[1][5]
DCPR	0.10[2][5]	.00[3][5]
Resulting Transient MCPR at SLO	1.32[5]	1.42[5]
SLMCPR for SLO for the Initial Cycle[4]	1.07	1.07
Margin Above the SLMCPR for the Initial Cycle	0.25[5]	0.35[5]
Frequency Category	Moderate frequency incident	Infrequent incident

NOTES:

(1) This value corresponds to an operating limit MCPR of 1.18 at rated conditions for the initial cycle.

(2) Analysis has been performed to conclude that turbine bypass capacity as low as 25% NBR does not affect the bounding DCPR results (Reference 15F.8-11).

(3) DCPR is less than 0.002.

(4) Values shown for initial cycle. The single loop operation MCPR safety limit is calculated in the current reload analysis. Refer to <Table 15B.15.0-1> for the two-loop value.

(5) The operating limit MCPR, DCPR, resulting margin above the SLMCPR, and resulting end of transient MCPRs represent initial cycle values or results. For the current cycle MCPR operating limit, refer to <Table 15B.15.0-3>.

(6) Analysis concludes that an increase in core flow to 60% of rated does not reduce the margin to the safety limit (Reference 15F.8-17).

15F.4 FUEL INTEGRITY - STABILITY

Background

The stability licensing basis for all U.S. nuclear power plants is set forth in <10 CFR 50, Appendix A>, General Design Criterion (GDC) 12. This requires assurance that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are either not possible or can be reliably and readily detected and suppressed. Historically, compliance to GDC-12 was achieved by demonstrating that thermal-hydraulic stability induced neutron flux oscillations were not expected. More recently, operating experience indicates that the thermal-hydraulic stability characteristics of a BWR are strongly influenced by a number of design and operating parameters which will vary depending upon the specific plant conditions at the time of the event. This was recognized and recommendations for special operator actions were provided in GE Service Information Letter (SIL) 380 Revision 0 (August 11, 1982) and Revision 1 (February 10, 1984) (Reference 15F.8-9a). These recommendations were accepted by the NRC as providing adequate compliance with the detection and suppression provision of GDC-12.

Following the March 9, 1988 LaSalle-2 event, evaluations indicated that under certain conditions, margins to safety limits may be less than previously expected. These findings were promptly reported to the NRC and interim corrective actions (ICAs) recommended by GE and the BWR Owners' Group were implemented at all U.S. plants.

In 1994, the ICA's were updated to incorporate subsequent industry experience. In addition, a commitment was made to implement a long-term solution to install the oscillation power range monitor (OPRM) instrumentation (Reference 15F.8-20). This system is capable of detecting and suppressing thermal-hydraulic instability and is described in <Section 7.2.1> and <Appendix 15B>.

The OPRM System was installed for startup into Cycle 9 (May 2001). Should the OPRMs become inoperable, Backup Stability Protection Guidelines (BSP) as described in <USAR 15B 4.4> will be implemented to comply with Technical Specification requirements.

Discussion

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 inservice 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 decreasing the stabilizing 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 smaller 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 flux 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 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 smaller stability margin as SLO core flow was increased, an evaluation was performed which phenomenologically accounts for single loop operation effects on stability (Reference 15F.8-7). Additionally, 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 test data (Reference 15F.8-7). The results of these analyses and tests indicate that the stability characteristics for single loop operation are not significantly different than those for 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). These are consistent with observed behavior in stability tests at operating BWRs (Reference 15F.8-8).

To provide assurance that acceptable plant performance is achieved during operation in the least stable region of the power/flow map, as well as during all plant maneuvering and operating states, the plant has in place the OPRM system, procedures and Technical Specifications. The OPRM system is provided to detect the evidence of thermal-hydraulic instability and respond appropriately (e.g., by scramming of control rods if specified conditions are met). The OPRM system is described in <Section 7.2.1> and <Appendix 15B>. Should the OPRM become unavailable, procedures and Technical Specifications provide actions to be taken which includes the BSP. The stability BSP are also appropriate for operation in the single loop mode.

15F.5 LOSS-OF-COOLANT ACCIDENT ANALYSIS

Refer to <Appendix 15B> for a description of the LOCA analysis relative to single loop operation.

15F.6 CONTAINMENT RESPONSE ANALYSIS

A single loop operation containment analysis was performed for PNPP. The peak wetwell pressure, peak drywell pressure, chugging loads, condensation oscillation, and pool swell containment response were evaluated over the entire single loop operation power/flow region.

The highest peak wetwell pressure during single loop operation occurred at the maximum power maximum flow condition. The peak wetwell pressure decreased by about fifteen percent compared to the rated two loop operation pressure given in <Chapter 6>. The peak drywell pressure evaluated at the worst power/flow condition during single loop operation was found to decrease by about 6 percent compared to the values given in <Section 6.2>. 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.

15F.7 MISCELLANEOUS IMPACT EVALUATION

15F.7.1 ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) IMPACT EVALUATION

The principal difference between single loop operation (SLO) and normal two loop operation affecting Anticipated Transient Without Scram (ATWS) performance is that of initial reactor conditions. The limiting ATWS events have been determined generically, based on part, on their relative severity, as described within the GE licensing topical report, Assessment of BWR Mitigation of ATWS (Reference 15F.8-13). For example, the report concluded that for ATWS events for a BWR-6, the idle recirculation loop startup event is less severe than the recirculation flow controller failure with increasing flow event which in turn is less severe than the MSIV closure or turbine trip ATWS cases (a Perry specific MSIV closure case is analyzed in <Appendix 15C>). PNPP specific event analysis within <Chapter 15> show the same relative severity. Since the SLO initial power/flow condition is less than the rated condition used for the two loop ATWS analysis, the transient response is less severe and therefore, bounded by the two loop analyses. All ATWS acceptance criteria are met during SLO. Therefore, SLO is an acceptable mode of operation for ATWS considerations.

15F.7.2 FUEL MECHANICAL PERFORMANCE

The thermal and mechanical duty for the transients analyzed have been evaluated and found to be bounded by the fuel design bases.

It has been observed that due to the reverse flow established during SLO, both the Average Power Range Monitor (APRM) noise and core plate differential pressure noise are slightly increased. Evaluations were performed to determine the acceptability of single loop operation on fuel rod and assembly thermal/mechanical performance. Component pressure differential and fuel rod overpower values were determined for anticipated operational occurrences initiated from SLO conditions. An analysis has been carried out to show 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. These values were found to be bounded by those applied in the fuel assembly design bases. In addition, fuel rod performance was determined to be within the design basis and limits specified in (Reference 15F.8-5).

15F.7.3 VESSEL INTERNALS VIBRATION

Maximum flow for two-pump balanced operation is equal to rated volumetric core flow at normal reactor operating conditions. Maximum flow for single recirculation pump operation is equal to the highest recirculation pump drive flow that is within acceptance criteria for reactor internals vibration, if not limited first by the runout capacity of a single operating recirculation pump. The maximum drive flow for PNPP is 48,500 gpm (based on the vibration acceptance criteria).

The purpose of vibration monitoring is to confirm the structural integrity of major components in the reactor with respect to flow-induced vibrations (FIVs) in accordance with requirements of Nuclear Regulatory Commission (NRC) <Regulatory Guide 1.20> for prototype plants. <Regulatory Guide 1.20> states that "the level of cumulative fatigue damage should be identified." A simpler alternative to evaluating the cumulative fatigue damage is to show that the vibration stress amplitudes are below the endurance limit of the material. This would indicate that the actual cumulative fatigue damage is zero. This is done by comparing the measured vibration amplitudes (strain or displacement) against a set of acceptance criteria and showing that they are less than the criteria. The acceptance criteria are basically a set of frequencies and corresponding allowable amplitudes (for each sensor) derived from an analytical model. This ensures that the stresses everywhere are below the material allowable stress. The structural integrity of components which exceed this simplified acceptance criteria can be demonstrated through evaluation of cumulative fatigue damage of the component undergoing FIV, as specified in <Regulatory Guide 1.20>.

The original report indicated that during two loop operation, all components met the simplified vibration acceptance criteria. Various components were monitored during startup testing; including the core shroud, jet pumps, core spray lines, control rod and in-core guide tubes. Analysis in the original report used the simplified and conservative “absolute sum of all modal responses” method in developing the vibration criteria. This method is known to be conservative in that the criteria are based on the assumption of vibration at a constant sustained maximum amplitude, whereas actual vibration amplitudes are generally random and seldom reach maximum recorded values. Components which satisfy this vibration acceptance criteria are expected to have an unlimited fatigue life, since the cyclic stress amplitudes are below the endurance limit.

Test results in the original report show that during SLO, all components met the simplified vibration acceptance criteria at drive flows up to 48,500 GPM, except for the in-core guide tubes. To allow operation at this flow rate, the fatigue usage of the in-core guide tubes was evaluated using test data from the test point with the highest drive flow and thus the highest vibration amplitudes.

For components which exceed the simplified vibration criteria, the NRC <Regulatory Guide 1.20> specified cumulative fatigue analysis methodology can be used. This method entails counting the number of stress cycles at a given stress level above the endurance limit. The fatigue usage factor corresponding to that stress level and the number of cycles is then calculated. The stress level is then increased and the fatigue usage factor is again calculated for the number of cycles corresponding to this second stress level. This is done for all stress levels of interest. Finally, the sum of the calculated usage factors is used as the cumulative usage factor for the period of interest. A cumulative usage factor of 1 would indicate that crack initiation may occur in the component.

Using this cumulative usage factor method, it was found that the in-core guide tubes would experience a fatigue usage factor of 0.11 for 1 year of operation in the single loop mode at 48,500 GPM (Reference 15F.8-15).

The expected plant-life fatigue usage for events other than single loop operation is 0.1. This leaves a fatigue usage of 0.9 for SLO. Thus;

$$(\text{number of years of SLO}) \times (0.11) = 0.9$$

$$(\text{number of years of SLO}) = \frac{0.9}{0.11} = 8.18$$

This evaluation indicates a fatigue life for the in-core guide tubes of 8.18 years. It can conservatively be concluded from these results that the in-core guide tubes are structurally adequate to withstand flow-induced vibrations caused by 8 years of single loop operation at a drive flow of 48,500 GPM. All the other monitored reactor internal components have vibration stress amplitudes below the endurance limit at this flow condition (48,500 GPM) and thus are expected to have an unlimited fatigue life during SLO. Based on these results the recirculation pump drive flow limit for SLO is 48,500 GPM.

Operation of the plant in the single loop mode at high drive flows has no impact on two loop operation, since the fatigue usage for other events was taken into account in the determination of the fatigue life of the in-core guide tubes during SLO.

15F.8 REFERENCES

15F.8-1 "General Electric BWR Thermal Analysis Basis (GETAB); Data, Correlation, and Design Application," NEDO-10958-A, January 1977.

15F.8-2 "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154, October 1978.

15F.8-3 R. B. Linford, "Analytical Methods of Plant Transients Evaluation for the General Electric Boiling Water Reactor," NEDO-10802, April 1973.

15F.8-4 "Compliance of the General Electric Boiling Water, Reactor Fuel Designs to Stability Licensing Criteria," NEDE-22277-P1, October 1984.

<u>NOTE</u>: (Reference 15F.8-4) above is the principal supporting document for GE's Topical Report NEDE-24011 Amendment 8.

15F.8-5 General Electric Company "General Electric Standard Application for Reactor Fuel," including the US Supplement, NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision).

15F.8-6 Acceptance for Referencing of Licensing Topical Report NEDE-24011 Revision 6, Amendment 8, "Thermal Hydraulic Stability Amendment to GESTAR II" NRC's C. O. Thomas to H. C. Pfefferlen, April 24, 1985.

15F.8-7 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.

15F.8-8 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).

15F.8-9a “BWR Core Thermal Hydraulic Stability” SIL No. 380, Revision 0, August 11, 1982 and Revision 1, February 10, 1984.

15F.8-9b <NRC Bulletin 88-07> Supplement 1, “Power Oscillations in Boiling Water Reactors (BWRs).”

15F.8-10 “General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with <10 CFR 50, Appendix K> Amendment No. 2 - One Recirculation Loop Out-of-Service,” NEDO-20566-2 Revision 1, July 1978.

15F.8-11 Letter, David T. Shen (GE) to Eileen Buzzelli “Perry Transient Analysis Assuming 25% Steam Bypass,” October 28, 1987.

15F.8-12 General Electric Company Topical Report, “Process Computer Performance Evaluation Accuracy,” NEDO-20340, June 1974.

15F.8-13 Generic Electric Company Topical Report, “Assessment of BWR Mitigation of ATWS, Volume II (<NUREG-0460> Alternate No. 3),” NEDE-24222, December 1979.

15F.8-14 USNRC <Regulatory Guide 1.20>, “Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing,” Revision 2, May 1976.

15F.8-15 General Electric Company Report, “Perry 1 Reactor Internals Vibration Measurements,” NEDE-31567-P-Addendum, by D. M. Fahrner, January 1991.

15F.8-16 Letter from D. T. Shen (GE) to K. P. Donovan (CEI), "USAR <Appendix 15F> on Single Loop Operation for PNPP - Resubmittal," June 25, 1991.

15F.8-17 Letter from D. T. Shen (GE) to K. P. Donovan (CEI), "Perry Single Loop Operation up to a Recirculation Pump Drive Flow of 48,500 GPM," June 25, 1991.

15F.8-18 General Electric Information document 23A4088, Revision 0, "Single Recirculation Pump Operation" issued March 2, 1984.

15F.8-19 Condition Report 98-1739.

15F.8-20 Letter from R. A. Stratman (CEI) to NRC, "Response to <Generic Letter 94-02>", PY-CEI/NRR-1855L, September 9, 1994.

<APPENDIX 15G>

CONTROL SYSTEM INTERACTIONS

15G CONTROL SYSTEM INTERACTIONS

15G.1 INTRODUCTION AND SUMMARY

The analyses reported in <Chapter 15> of the USAR are intended to demonstrate the adequacy of safety systems to mitigate anticipated operational occurrences and accidents. The <Chapter 15> analyses evaluated the worst failures of various system components. However, <Chapter 15> did not specifically look at combinations of failures resulting from the possible additive effects of nonsafety systems, which could place the plant in a condition which has not been previously analyzed. Therefore, to provide assurance that the conservative assumptions made in defining these design-basis events result in adequate bounding of other, more fundamental credible failures, the following evaluation was performed.

This evaluation first identified those functions which are required in order to maintain the reactor in a safe condition. These functions are: reactivity, reactor vessel level, reactor system pressure, and decay heat removal. Then all systems were identified (both safety and nonsafety) which are connected to and could significantly impact the required safety functions.

In order to identify the systems which impact reactor safety functions, a control volume of the reactor vessel was developed, showing all the systems which are connected to it. Since PNPP has two safety-related redundant trains powered by 1E qualified systems, only interactions between nonsafety and safety systems resulting from control or electrical failures in the nonsafety systems and propagating through the fluid process were considered.

If there appeared to be a system or combination of systems which could significantly impact the required safety functions, an investigation was made to determine whether any control or electrical dependencies existed which could cause interactions among these systems and thereby affect their function. Once the nonsafety-related control system components were identified, locations were determined from plant layout drawings. Components were then assigned to their respective environmental zones. All equipment within a single zone was assumed failed and the impact on system alignments then determined. The process was repeated for each environmental zone.

Potential system interactions were identified for further evaluation. A plant walkdown was then performed to determine the location of high energy lines with respect to potentially affected components. The results of the plant walkdown along with engineering analysis allowed determination of the consequences of each adverse condition.

15G.2 RESULTS

The following systems are Class 1E and did not require further evaluation for electrical or instrumentation control failures:

(1) Automatic depressurization

(2) Reactor core isolation cooling

(3) Residual heat removal

(4) High pressure core spray

(5) Low pressure core spray

(6) Control rod drive (scram portion)

(7) Nuclear boiler instrumentation

For those systems with non-Class 1E components, there is a spectrum of possible flow rates to and from the reactor vessel ranging from a few gpm for the sampling system to tens of thousands of gpm for the feedwater system. The systems with the largest capacity will have the quickest and greatest potential impact on the required safety functions. Each system was evaluated to determine whether the system capacity warranted further analysis. The results of the evaluation are:

(1) The standby liquid control system inserts negative reactivity into the core. The insertion of negative reactivity is always a conservative step in an accident or transient analysis. The SLC system cannot remove water; it is a manually initiated system which is strictly a backup to a redundant safety-related system. Its failure or malfunction will have no adverse impact on plant safety.

(2) The control rod drive system is required to assure scram capability. The components required to scram the reactor are redundant and Class 1E. The control rod drive pumps are non-1E and pump a relatively small quantity of fluid into the reactor compared with the feedwater system. Any failure of the non-1E portions will have no adverse impact on plant safety.

(3) The reactor plant sampling system is a low capacity system which is manually controlled. The system is automatically isolated by redundant Class 1E isolation valves. There are no non-1E failures associated with this system which could have an adverse impact on plant safety.

(4) The reactor water cleanup system is a normally operating system with minimal capacity compared to the feedwater system. The system is automatically isolated by redundant Class 1E

isolation valves on low reactor water level. The failure or malfunction of non-1E portions of this system will not seriously impact plant safety.

(5) The head vent system valves are manually operated. Valve position is not affected by a bus failure.

The conservatisms in the design of the safety-related systems are adequate to absorb the small system effects from the above systems. The systems remaining to be evaluated are:

(1) Recirculation

(2) Feedwater (including extraction steam)

(3) Main steam

(4) Turbine bypass (condenser vacuum)

These systems or their equivalents are the same ones which potentially cause problems in PWR's. Because of their large capacities, these systems, along with the subsystems which control them, are the key non-1E systems which affect plant safety.

During the evaluation, it became clear that the control features of the systems were the key to system interactions. The following control systems are the key contributors and have been evaluated extensively:

(1) Recirculation valve flow control system

(2) Steam bypass and pressure regulator system

(3) Turbine driven feedwater pump control system

(4) Extraction steam valve control system

The following adverse system alignments and operating states were analyzed for conditions which could result in a transient not analyzed in USAR <Chapter 15> analyses, i.e., any effect of these systems on the capability to maintain control of reactivity, reactor vessel water level, reactor system pressure, and decay heat removal.

Adverse Condition No. 1

Failure of 13.8 kV Bus L-10, 1R22S001.

EVALUATION: Loss of this bus could potentially cause a main turbine trip due to the loss of circulating water pump C and its subsequent effect on condenser vacuum. This main turbine trip, without additional complications, is bounded by the <Chapter 15> load rejection analysis.

Adverse Condition No. 2

Failure of 13.8 kV Bus L11, 1R22S002, affects the following:

Recirculation Pump - Trip
Feedwater Pumps - Trip
Feedwater Booster Pumps - Trip

The postulated scenario, inadvertent recirculation pump trip and concurrent reduced feedwater flow, is bounded by USAR <Chapter 15> analysis. Normal water level will decrease to Level 3 at which time scram will be initiated. After that, RCIC and HPCS will maintain water level.

Loss of this bus causes coast down of recirculation pump A and main turbine trip on low condenser vacuum due to loss of offgas refrigerators and circulation pump A. Since partial loss of recirculation flow would immediately start reducing reactor power, an immediate or delayed turbine trip would produce an equal or less severe transient than the load rejection event of <Chapter 15>. Therefore, this event is bounded.

Adverse Condition No. 3

Failure of 13.8 kV Bus L12, 1R22S003, affects the following:

Recirculation Pump - Trip
Recirculation Valves - Lock in place
Feedwater Pumps - Trip
Feedwater Booster Pumps - Trip
Extraction Steam - Valves Close

EVALUATION: Loss of this bus or associated lower busses will produce some or all of the following effects: Immediate main turbine trip and reduction in feedwater temperature, reduction in feedwater flow, recirculation flow decrease. Of these, the only effect capable of causing a new positive reactivity insertion is the loss of feedwater heating.

Failure of Bus 1R25S012, ancillary to Bus 1R22S003, results in simultaneous main turbine trip and start of feedwater temperature reduction. This event, immediate main turbine trip with start of feedwater temperature reduction, does not result in consequences more severe than the load rejection transient event of <Chapter 15>. Therefore, this event is bounded.

Adverse Condition No. 4

Failure of 480V Bus F-1-D, 1R23S004, affects the following:

Feedwater Pumps - Trip
Recirculation Valves - Lock in place

Extraction Steam Valves - Fail closed

EVALUATION: There is no reactivity addition since there is no feedwater flow and recirculation flow remains constant. The transient is covered by <Chapter 15> analysis.

Adverse Condition No. 5

Failure of the Unit Auxiliary Transformer, 1S11S003, affects the following:

Feedwater Pumps - Trip
Extraction Steam Valves - Closed
Recirculation Valves - Lock in place
Recirculation Pumps - Trip

EVALUATION: In this state, the known failure mode of the extraction steam valves causes an increase in reactivity as a result of colder feedwater. If the feedwater pump flow increases and/or recirculation flow increases, we have a transient not analyzed in <Chapter 15>.

However, the failure of this bus also trips the steam supply valves to one of the feedwater pumps and trips the feedwater booster pumps. The net result is a loss of feedwater which means that the net adverse alignment, assuming the least favorable results, is an increase in recirculation flow, except that a recirculation pump also trips. The postulated scenario, inadvertent recirculation pump trip and concurrent reduced feedwater flow, is bounded by USAR <Chapter 15> analysis. Normal water level will decrease to Level 3 at which time scram will be initiated. After that, HPCS will maintain water level (RCIC is also available but not required).

Adverse Condition No. 6

Failure of 480V Distribution Panel F1D12, 1R25S004, affects the following:

Recirculation Valves - Lock in place
Extraction Steam Valves - Closed

EVALUATION: There is some reactivity addition from colder feedwater, but this addition is less severe than the reactivity addition analyzed in <Chapter 15>.

Adverse Condition No. 7

Failure of Control Complex 120 Vac Miscellaneous Distribution Panel K-1-R, 1R25S122, affects the following:

Recirculation Valves - Lock in place
Extraction Steam Valves - Closed

EVALUATION: There is some reactivity addition from colder feedwater, but this addition is less severe than the reactivity addition analyzed in <Chapter 15>.

Adverse Condition No. 8

Failure of Static Transfer Switch, 1R14S008, or station normal 125V, Battery 1A, 1R42S001, or 125V DC Bus D-1-A, 1R42S021, affects the following:

Turbine Bypass Valves - No effect
Feedwater Pumps - Increase flow

EVALUATION: Loss of this bus will result in an increase in feedwater flow and subsequent Level 8 trip causing main turbine trip. This event is similar to and bounded by the feedwater runout event analyzed in <Chapter 15>.

Adverse Condition No. 9

Pipe break at Environmental Zone CT2[1] could affect the following:

INSTRUMENT	PANEL	BUILDING	ELEVATION	COORDINATES
1B33N001A[2]	1H22-P025	Containment	620	C1/12
1B33N001B[2]	1H22-P041	Containment	620	C1/17
1C34N003B[3]	1H22-P025	Containment	620	C1/12
1C34N003D[3]	1H22-P041	Containment	620	C1/17

NOTES:

[1]

HIGH ENERGY LINE	BUILDING	ELEVATION	COORDINATES
C11-Control Rod Hydraulic System	Containment	630	C1/11
C41-Standby Liquid Control	Containment	602	C1/17

[2] Recirculation Valves - Failure mode unknown.

[3] Feedwater Pumps - Failure mode unknown.

All of the above instruments and high energy lines are located between the drywell and the reactor building wall. Exact locations may be identified by referencing <Figure 3.6-4>, <Figure 3.6-6>, <Figure 3.6-12>, and <Figure 3.6-14> of the USAR.

EVALUATION: Instruments 1B33N011B and 1C34N003D are located in Panel 1H22P041 between the drywell and containment wall on Level 620’. The remaining two instruments are in Panel 1H22-P025 on the same level.

The only high energy piping in the vicinity of these panels are the control rod drive hydraulics.

High energy lines between the drywell and reactor building wall are restrained from whipping by pipe restraints. Therefore, we may eliminate the effects of whipping pipe <Section 3.6>. The control rod drive piping does not carry high temperature water so we can neglect increases in temperature in the area of these transmitters. Therefore, a postulated CRD line break would not affect the operability of the instruments in this area.

Adverse Condition No. 10

Pipe break in the Heater Bay[1] could affect the following:

INSTRUMENT	PANEL	BUILDING	ELEVATION	COORDINATES
1N27N156A[2]	1H51P1147	Heater Bay	600	B/01
1N27N156B[2]	1H51P098	Heater Bay	620	C/02
1N27N087A[2]	1H51P098	Heater Bay	620	C/02
1N27N087B[2]	1H51P098	Heater Bay	620	C/02
1N25N263A[3]	Heater 6A	Heater Bay	600	D/03
1N25N263B[3]	Heater 6B	Heater Bay	600	D/03
1N25N303A[3]	Heater 5A	Heater Bay	600	D/02
1N25N303B[3]	Heater 5B	Heater Bay	600	D/02
1N36N030A[3]	1H51P1305	Heater Bay	620	B/01
1N36N030B[3]	1H51P1330	Heater Bay	620	B/01
1N36N030C[3]	1H51P1330	Heater Bay	620	B/01
1N26N153A	Heater 3A	Heater Bay	647	D/03
1N26N153B	Heater 3B	Heater Bay	647	D/04
1N21N339	Heater 4	Heater Bay	580	C/02

NOTES:

(1) HEATER BAY HIGH ENERGY LINES	ELEVATION
N22 Main, reheat extraction and misc. drains	580, 600, 620, 647
N25 H.P. Htr. drain and vent	580, 600, 620
N27 Feedwater system	580, 600, 620, 647
N33 Steam seal system	600, 620, 647
N36 Extraction steam system	600, 620, 647
N11 Main steam system	620, 647
P61 Auxiliary steam system	580, 600, 620, 647
N21 Condensate system	600, 620, 647

(2) Feedwater Pumps - Fail as is.

(3) Extraction Steam Valves - Fail closed.

Refer to plant layout drawings, <Figure 1.2-4> and <Figure 1.2-5>, for the Heater Bay layout.

EVALUATION: It was determined that the only high energy lines located in the vicinity of these instruments were feedwater lines. Instruments 1N25N263A(B) and 1N25N303A(B) are isolated from the other instruments by a concrete wall. Instrument 1N27N156A is also isolated from the other instruments by walls.

Therefore, that leaves only the remaining instruments that could cause multiple failures. Instruments 1N27N156B and 1N27N087A(B), which are located in Panel H51-P098, are located approximately forty feet from Panel H51-P1330 which contains 1N36N030A, B and C. Failure of 1N36N030A, B and C will cause a main turbine trip at the same time the extraction steam valves go closed. Feedwater pumps would fail as is; this would not increase flow to the reactor.

This event is bounded by <Chapter 15> analysis.

Adverse Condition No. 11

Pipe break at Environmental Zone TB1 could affect the following:

INSTRUMENT	PANEL	BUILDING[1]	ELEVATION	COORDINATES
	1R25S002	TP	620	A/04
	1R42S022	TP	620	C/06
	1R11S004	TP	620	B/04
	1R11S005	TP	620	B/02
	1R14S008	TP	647	B/05
	1R22S002	TP	620	A/04
	1R22S003	TP	620	A/04
	1R22S004	TP	647	A/05
	1R22S005	TP	647	A/04
	1R23S004	TP	647	B/06
	1R23S006	TP	647	B/06
	1R22S016	TP	620	A/03
	1R23S017	TP	647	A/05
	1R23S003	TP	647	B/05
	1R24S034	TP	620	B/06
	1R42S017	TP	620	C/06
	1R42S021	TP	620	C/06
	1R42S023	TP	620	C/07
1C85N001A		TB	593	E/16
1C85N001B		TB	593	E/16
1C85N002A	1H51P187	TB	647	G/08
1C85N002B	1H51P187	TB	647	G/08
1C85N002C	1H51P187	TB	647	G/08

NOTE:

[1] TP - Turbine Power Complex
TB - Turbine Building

Refer to plant layout drawings, <Figure 1.2-5> and <Figure 1.2-6>.

EVALUATION: The R-system components are located in the Turbine Power Complex, which does not contain any high energy lines. Therefore, the possibility of multiple failures due to a high energy line break does not exist.

15G.3 CONCLUSION

The failures of power sources, sensors or sensor impulse lines which provide power or signals to two or more control systems will not result in consequences outside the bounds of the <Chapter 15> analyses or beyond the capability of operators or safety systems to mitigate those consequences.

Multiple control system malfunctions, due to harsh environment caused by a high energy line break, do not result in consequences more severe than those analyzed in <Chapter 15> nor, are they beyond the capability of operators or safety systems to mitigate those consequences.

<APPENDIX 15H>

STATION BLACKOUT (SBO)

APPENDIX 15H

TABLE OF CONTENTS

APPENDIX 15H

LIST OF TABLES

15H STATION BLACKOUT (SBO)

The Perry specific SBO assessment which conforms to the requirements of <10 CFR 50.63> and <Regulatory Guide 1.155> follows.

15H.1 BACKGROUND

Due to power uprate to 3,758 MWt, Station Blackout was re-evaluated using the guidelines of NUMARC 87-00. The plant response and coping capabilities for a Station Blackout event are affected slightly by operation at the uprated power level due to the increase in decay heat. There are no changes to the systems and equipment used to respond to the SBO, nor is the required coping time changed. <Table 15H-1> reflects the uprate analysis.

The term "Station Blackout" refers to the complete loss of alternating current (AC) electric power to the essential and nonessential switchgear buses. Station Blackout therefore, involves the loss of offsite power concurrent with a turbine trip and failure of the onsite emergency AC power system, but not the loss of AC power to buses fed from plant batteries through inverters or from the loss of power from "alternate AC sources." Because many safety systems required for decay heat removal and containment heat removal are dependent upon AC power, the consequences of a SBO could be severe.

The concern about SBO arose because of the accumulated experience regarding the reliability of AC power supplies. Many plants have experienced loss of offsite power events, but in almost every case, the onsite emergency AC power system was available to immediately supply power to safety equipment. However, in some cases, either some or all of the emergency AC power supplies were not available. It should be noted that during these loss of all AC power events, power was restored in a short time without any serious consequences. In addition, there have been numerous instances when emergency diesel generators have failed to start and run in response to tests conducted at various operating nuclear plants.

The results of the Reactor Safety Study (Reference 1) showed that SBO could be an important contributor to the total risk from nuclear power plant accidents. Though this total risk is small, the relative importance of SBO was established. The accumulated diesel generator failure experience increased the concern about SBO.

Given the above, the NRC conducted an analysis of the SBO issue. The results of the NRC analyses are contained in (Reference 2) (Reference 3) (Reference 4) (Reference 5) (Reference 6) (Reference 7). In summary, the analyses show that SBO does not pose an undue risk to public health and safety. The findings show that recovery from the event occurs in the most part in less than 4 hours, diesel generator reliability is high, and that given a SBO, core damage is more dependent upon decay heat removal systems that are <u>not</u> AC-dependent. However, plant design and operational characteristics plus site-dependent factors (e.g., weather conditions) introduce a variability that warrants a need for plant specific analyses to provide greater assurance that core cooling can be maintained until power is restored. Thus, the NRC amended <10 CFR 50> to incorporate <10 CFR 50.63>.

<10 CFR 50.63> requires all nuclear power plants to be capable of coping with a SBO for some specified time period. The specific period of time for a plant is based on a comparison of the individual plant's design with factors that have been identified as the main contributors to the risk of core damage should a SBO event occur.

On the basis of the SBO studies referenced above and the rule, the NRC issued <Regulatory Guide 1.155>, "Station Blackout," to provide guidance on an acceptable method of compliance. The regulatory guide provides guidance on selecting the appropriate SBO duration, which a plant need be capable of surviving without core damage, on maintaining a high level of diesel generator reliability, and on developing procedures to restore both onsite and offsite AC power should one or both become unavailable.

As part of <10 CFR 50.63>, licensees were required to submit to the NRC a proposed SBO duration, a description of SBO procedures, and a list of equipment and procedure modifications necessary to satisfy the rule. The rule further stated that once the information had been received, the NRC would notify the licensee as to the adequacy of the information with respect to compliance with the rule. (Reference 8) (Reference 9) (Reference 10) and (Reference 12) contain the Perry response to the SBO rule. The NRC's review <u>and</u> acceptance of the Perry response is contained in the NRC's Safety Evaluation and Supplemental Safety Evaluation, (Reference 11) and (Reference 13), respectively.

15H.2 <u>ASSESSMENT</u>

<Regulatory Guide 1.155>, "Station Blackout", provides the guidance the NRC finds acceptable for satisfying the requirements of <10 CFR 50.63.> The major items addressed by the regulatory guide are:

- determination of the facility's "coping" period,
- evaluation of the facility's capability to cope with an event,
- guidance for facility modifications necessary to enable the facility to cope with an event,
- evaluation on the need for developing procedures to permit personnel to respond and recover from an event, and
- guidance for quality assurance activities on nonsafety-related equipment that may be used for coping with the event.

<10 CFR 50.63> requires each licensee to determine the duration of time a facility might be subjected to a SBO. The term for this period is

called the “coping” duration. The determination of the coping duration involves consideration of a wide variety of factors that are unique to each nuclear facility. The major factors are:

- redundancy of the onsite emergency AC power system,
- the reliability of each of the onsite emergency AC power sources,
- the expected frequency of loss of offsite power, and
- the probable time needed to restore offsite power.

The coping duration analysis methodology is a five step process. The five steps are:

- Step 1: Determine the offsite power design characteristics.
- Step 2: Classify the onsite emergency AC power supply system configuration.
- Step 3: Determine the calculated diesel generator reliability.
- Step 4: Determine the target diesel generator reliability.
- Step 5: Determine the coping duration.

Once all five steps have been completed, the coping time of a facility will be known.

After the coping time is known, an evaluation of the facility will be prepared to determine the facility’s capability to withstand and recover from a SBO. The major factors that need to be reviewed are:

- use of an alternate AC power source,
- sufficient condensate inventory available for decay heat removal,
- capacity of Class 1E batteries to support safe shutdown equipment,
- compressed air for air-operated valves needed for safe shutdown,
- ventilation systems in areas needed to support safe shutdown equipment,

- ability to maintain the appropriate level of containment integrity, and
- ability to maintain adequate reactor coolant inventory.

Based upon the coping duration analysis and the evaluation of capability to cope with the SBO event, the facility may need to be modified. The regulatory guide provides guidance for some of the modifications that could be expected to be performed by a facility in order to better cope with a SBO event.

The regulatory guide notes that procedures should be developed which provide the guidance necessary for plant personnel to respond to and recover from a SBO event.

The regulatory guide also provides guidance on quality assurance activities for nonsafety-related equipment that a facility may need to use to cope with the SBO event.

15H.2.1 SBO COPING DURATION ANALYSIS

15H.2.1.1 Coping Duration Analysis Step 1

Step 1 of the methodology is a five part process which uses plant unique inputs to determine the offsite power design characteristics. The five parts of this step with the PNPP inputs follows.

Part 1 Determine the site susceptibility to grid-related loss of offsite power events.

PNPP is not expected to lose off-site power more often than once per 20 years. This is based upon the known reliability of the 345 KV grid.

Part 2 Estimate the frequency of loss of offsite power due to extremely severe weather.

The regulatory guide defines extremely severe weather as storms with wind velocities equal to or greater than 125 miles per hour. The methodology is divided into five Extremely Severe Weather groups, each based upon a probability of experiencing this type of weather condition. PNPP site specific data supports a probability factor of less than 0.000001 (Reference 14). Applying this data to the methodology, PNPP is within Extremely Severe Weather Group 1.

Part 3 Estimate the frequency of loss of offsite power due to severe weather.

The regulatory guide uses four factors to determine which of two Severe Weather groups a facility falls in. The four factors are: annual expected snowfall, annual expectation of tornadoes, annual expectation of storms with wind velocities between 75 and 124 MPH, and annual expectations of hurricanes. The annual snowfall for the vicinity of PNPP is 56.8 inches <Section 2.3.1.1.4>. The expectation of a tornado is 0.000165 <Section 2.3.1.2.2>. The expectation of storms with wind velocities between 75 and 124 mph is 0.01 (Reference 14). The expectation for a hurricane is 0.0. Using these values, PNPP is within Severe Weather Group 2.

Part 4 Evaluate the independence of offsite power systems.

The response to Part 4 is based upon the actual configuration of the facility's tie-in with offsite power and its use of offsite power. The PNPP configuration is classified as I 1/2.

Part 5 Determine the offsite AC power design characteristic group.

Based upon the responses from the first four parts, PNPP is considered an Offsite Power Group P1.

15H.2.1.2 Coping Duration Analysis Step 2

Step 2 of the methodology is a three part process which is used to classify the emergency AC power configuration. The three parts of this step with PNPP specific inputs follows.

Part 1 Determine the number of emergency AC power supplies not credited as alternate AC power sources.

PNPP has two dedicated diesel generators, Division 1 diesel generator and Division 2 diesel generator.

Part 2 Identify the smallest number of emergency AC power sources necessary for safe shutdown.

For PNPP, only one divisional diesel generator is necessary to operate safe shutdown equipment. Either division has the capability to achieve and maintain the facility in a safe shutdown condition.

Part 3 Select the Emergency AC Power configuration.

Based upon the inputs from Parts 1 and 2, PNPP is in Emergency AC Power Configuration Group C.

15H.2.1.3 Coping Duration Analysis Step 3

Step 3 of the methodology is the determination of the calculated diesel generator reliability. The regulatory guide requires the diesel generator reliability data be based upon the last 20, 50, and 100 demands and on the methodology contained in (Reference 15). The results for Perry ending October 1993 are as follows:

Unit Average	
Last 20	100%
Last 50	100%
Last 100	99%

15H.2.1.4 Coping Duration Analysis Step 4

Step 4 of the methodology is the determination of the target diesel generator reliability. Given the configuration group determined in Step 2 and the calculated reliability numbers from Step 3, the target reliability number for PNPP is 0.95.

PNPP has committed to maintaining the target reliability number of 0.95 as documented in (Reference 10), (Reference 12), and (Reference 13), respectively.

15H.2.1.5 Coping Duration Analysis Step 5

Step 5 of the methodology is the determination of the coping duration. Based upon the inputs from Steps 1 through 4, the coping duration for PNPP is 4 hours. This duration was approved by the NRC as detailed in (Reference 12) and (Reference 13).

15H.2.2 SBO CAPABILITY EVALUATION

The PNPP SBO strategy is to remove decay heat from the fuel by providing makeup water from the High Pressure Core Spray (HPCS) system and transferring the heat through the SRVs to the suppression pool. The HPCS system will be powered by the Division 3 (HPCS) diesel generator. The condensate storage tank will be the primary source of makeup water.

A dump of the upper containment pools will be used to moderate suppression pool temperatures. Refer to <Table 15H-1> for the SBO sequence of events.

The HPCS system will provide the means of decay heat removal. The HPCS pump and its supporting components will be powered by the Division 3 (HPCS) Diesel Generator. Additionally, the Division 3 diesel generator has enough excess capacity to power other components during a SBO event. To accomplish this, a manual cross-tie exists between the Division 2, and the Division 3 switchgears. During normal plant operations, physical and electrical separation of the two divisions will be maintained through design features of the cross-tie. If a SBO event should occur, plant operator will manually perform the cross-tie. Although the NRC did not consider the HPCS diesel generator to fully satisfy the strict definition of an Alternate AC (AAC) power source, the NRC has evaluated the use of the HPCS diesel generator in the PNPP coping analysis as acceptable (Reference 11) (Reference 12).

The condensate inventory required for decay heat removal from the fuel and to makeup for reactor coolant leakage (a requirement of the regulatory guide) for a SBO event is less than 150,000 gallons. The condensate storage tank maintains a minimum of 150,000 gallons dedicated for HPCS/RCIC use. This is based upon the suction source of other systems which draw from the tank being at levels above the region dedicated for HPCS/RCIC use. The 150,000 gallon reserve for these two systems is ensured by the operator isolating all nonsafety uses of the condensate supply before the storage tank level drops below this reserve. Additionally, approximately 1,000,000 gallons of recyclable water is available in the suppression pool and in the upper containment pools for use if needed. Therefore, there is sufficient condensate water to cope with and recover from a SBO (Reference 11) (Reference 13).

The PNPP Class 1E battery system is capable of supporting the necessary SBO loads for 4 hours. The class 1E system is comprised of four separate battery subsystems. Two of the subsystems are associated with Unit 1, one for Division 1 and one for Division 2. Similarly, Unit 2 has the same battery configuration. The Unit 2 batteries have been put into service to support Unit 1 operations. A maintenance cross-tie exists between the divisional Unit 1 and Unit 2 batteries. This cross-tie provides the 4 hour battery capacity. No Unit 1 load shedding was assumed in determining the capability to supply DC power to necessary loads during the 4 hour duration of the SBO event.

No air-operated valves, other than safety/relief valves (SRVs) are relied upon during the 4 hour duration of the SBO event when using the HPCS System. The SRVs will operate either on spring pressure for overpressure protection, or by manual operation by using air stored in the safety-related air system. Compressed air is also required for the operation of the Automatic Depressurization System (ADS) valves. Each ADS valve is equipped with an air accumulator, which is capable of opening and holding the valves open against the maximum drywell pressure. The accumulator capacity is sufficient to provide two actuations of each ADS valve against 70% of the maximum drywell pressure.

Loss of ventilation during the 4 hour duration of the SBO event will not cause temperature increases which impact the habitability and equipment qualification criteria in the areas containing equipment necessary for SBO.

The containment isolation valves which do not close as a result of the initiation of the SBO event and that must be either closed or operated (cycled) during the event can be properly positioned. The cross-tie between Division 2 and Division 3 will provide the necessary power to perform this function. Refer to <Table 15H-2> for a listing of these valves.

The HPCS System powered by the Division 3 diesel generator will maintain adequate reactor coolant inventory to ensure the reactor core is covered during the coping period. The reactor coolant inventory evaluation includes a 18 gpm per Reactor Recirculation Pump seal leakage factor. Consideration of this factor is a requirement of the regulatory guide.

15H.2.3 SBO PROCEDURES

Procedures have been implemented which provide the necessary guidance for site personnel to be able to respond to severe weather conditions, to respond to a SBO event, and to restore AC power if lost.

15H.2.4 SBO MODIFICATIONS

Two modifications were implemented as a result of the SBO analysis. The first is the cross-tie between the Division 2 and Division 3 switchgear described in <Appendix 15H.2.2>. The second is the installation of battery powered lights at locations which would facilitate in-plant operator actions.

15H.2.5 NRC REVIEW

As stated in <Appendix 15H.1>, the NRC reviewed the Perry SBO analysis and concluded that the Perry analysis is in compliance with the SBO rule.

15H.3 REFERENCES

1. U.S. Nuclear Regulatory Commission, "Reactor Safety Study," WASH-1400, October 1975

2. U.S. Nuclear Regulatory Commission, "Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issue A-44", <NUREG-1032>, June 1988

3. U.S. Nuclear Regulatory Commission, “Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout,” <NUREG-1109>, June 1988

4. U.S. Nuclear Regulatory Commission, “Collection and Evaluation of Complete and Partial Losses of Offsite Power at Nuclear Power Plants”, <NUREG/CR-3992> (ORNL/TM-9384), February 1985

5. U.S. Nuclear Regulatory Commission, “Reliability of Emergency AC Power System at Nuclear Power Plants,” <NUREG/CR-2989> (ORNL/TM-8545), July 1983

6. U.S. Nuclear Regulatory Commission, “Emergency Diesel Generator Operating Experience, 1981-1983”, <NUREG/CR-4347> (ORNL/TM-9739), December 1985

7. U.S. Nuclear Regulatory Commission, “Station Blackout Accident Analyses (Part of NRC Task Action Plan A-44)”, <NUREG/CR-3226> (SAND82-2450), May 1983

8. Letter from A. Kaplan, Cleveland Electric Illuminating Company, to the U.S. Nuclear Regulatory Commission, dated April 17, 1989 (CEI Letter PY-CEI/NRR-0995L)

9. Letter from A. Kaplan, Cleveland Electric Illuminating Company, to the U.S. Nuclear Regulatory Commission, dated March 30, 1990 (CEI Letter PY-CEI/NRR-1159L)

10. Letter from M. D. Lyster, Centerior Energy, to the U.S. Nuclear Regulatory Commission, dated March 15, 1991 (CEI Letter PY-CEI/NRR-1329L)

11. Letter from J. R. Hall, U.S. Nuclear Regulatory Commission, to M.D. Lyster, Centerior Energy, dated April 23, 1992 (CEI Letter PY-NRR/CEI-0590L)

12. Letter from M. D. Lyster, Centerior Energy, to the U.S. Nuclear Regulatory Commission, dated May 28, 1992 (CEI Letter PY-CEI/NRR-1497L)

13. Letter from J. R. Hall, U.S. Nuclear Regulatory Commission, to M. D. Lyster, Centerior Energy, dated July 22, 1992 (CEI Letter PY-NRR/CEI-0604L)

14. NUS Corporation Letter, PY-NUS/CEI-1184, Dated March 21,1986

15. Electric Power Research Institute, "Reliability of Emergency Diesel Generators at U.S. Nuclear Power Plants", NSAC-108, September 1986

16. Analysis Completion Notification, Perry Nuclear Power Plant, Asset Improvement Project, Task G1-45: Station Blackout, GE-NE-A2200084-45-01P, Revision O, Class III, July 1999

TABLE 15H-1

STATION BLACKOUT SEQUENCE OF EVENTS

Time	Event
0 sec	Initial Conditions: USAR <Table 15.0-1>.
	Loss of All AC Transmission Lines, Plant Undervoltage Relays Initiate Diesel Starts for Division 1, Division 2, and Division 3.
10 sec	Division 1 and Division 2 Diesels Fail to Reach Design Conditions.
13 sec	Division 3 Diesel at Design Conditions, Division 3 (HPCS) AC Power Available.
36 sec	HPCS Initiates on Level 2.
28 min	Manual Depressurization, Suppression Pool at 120°F.
133 min	Suppression Pool at 180°F, Upper Pool Dumped to Suppression Pool.
137.5 min	Condensate Storage Tank Volume Depleted, HPCS Pump Suction Transferred to Suppression Pool.
142 min	Upper Pool Dump Completed, Suppression Pool at 165°F.
240 min	Suppression Pool at 184.6°F.
	Division 1 or Division 2 AC Power Restored. Normal Decay Heat Removal and Suppression Pool Cooling Established.

TABLE 15H-2

CONTAINMENT ISOLATION VALVES THAT NEED ALTERNATE AC POWER TO FUNCTION DURING A STATION BLACKOUT EVENT[1]

1G61-F075	1P11-F090	1G50-F272
1E51-F063	1G33-F001	
1B21-F016		
1G61-F165	1G41-F140	

NOTE

[1] This listing is based upon a SBO occurring with the plant in a normal, full power configuration.