CHAPTER 15

ACCIDENT ANALYSES

15.0 GENERAL

Cycle 1

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

The scope of the situations analyzed includes anticipated (expected) operational occurrences (e.g., loss of electrical load), off-design abnormal (unexpected) transients that induce system disturbances, postulated accidents of low probability (e.g., the sudden loss of integrity of a major component), and finally, hypothetical events of extremely low probability (e.g., an anticipated transient without the operation of the entire control rod drive (CRD) system). These events and postulated accidents have been grouped into eight categories and are discussed in Sections 15.1 through 15.8.

This section describes the methodology utilized in performing the safety analysis that evaluates the core and system performance, the performance of the barriers provided and the radiological consequences of each event.

Power Uprate

Nine Mile Point Nuclear Station - Unit 2 (Unit 2) was originally licensed to operate at reactor core power levels not in excess of 3,323 MWt. Beginning with Cycle 5, the licensed core thermal power limit is increased by 4.3 percent to 3,467 MWt.

The Unit 2 stretch power uprate (SPU) program followed the General Electric (GE) Nuclear Energy generic guidelines and evaluations for boiling water reactor (BWR) power plants (References 6 and 7). Unit 2-specific analyses and evaluations were performed, consistent with the generic guidelines, for systems and components that might be affected to ensure their capability to support the increase in power output and steam flow, as reported in Reference 5. The limiting events for each Updated Safety Analysis Report (USAR) Chapter 15 transient category were reanalyzed using a representative equilibrium 18-month fuel cycle reactor core (GE8x8NB) to demonstrate the overall capability of the Unit 2 design to meet all transient safety criteria for uprated operation.

Direct or statistical allowance for 2 percent power uncertainty was included. No changes to the basic characteristics for any of the limiting events were caused by power uprate.

The radiological consequences for several postulated accidents were also recalculated for power uprate conditions. The calculated exposures and control room doses continue to meet the 10CFR100 guidelines and General Design Criterion (GDC)-19.

Specific transient analyses are performed, and core operating limits are developed, for each specific fuel cycle. These cycle-specific results, which include the effects of power uprate beginning with Cycle 5, are described in USAR Appendix A.

Extended Power Uprate

NMP2 was originally licensed to operate at reactor core power levels not in excess of 3,323 MWt. Beginning with Cycle 14, the licensed core thermal power limit was increased by 20 percent to 3988 MWt.

The Unit 2 Extended Power Uprate (EPU) program followed the GE Nuclear Energy generic guidelines and evaluations for boiling water reactor (BWR) power plants (References 9, 10, and 11). Unit 2-specific analyses and evaluations were performed, consistent with the generic guidelines, for systems and components that might be affected to ensure their capability to support the increase in power output and steam flow, as reported in Reference 8. The limiting events for each Updated Safety Analysis Report (USAR) Chapter 15 transient category were reanalyzed using a representative equilibrium 18-mo fuel cycle reactor core (GE14) to demonstrate the overall capability of the Unit 2 design to meet all transient safety criteria for uprated operation.

Direct or statistical allowance for 2-percent power uncertainty was included. No changes to the basic characteristics for any of the limiting events were caused by EPU.

The radiological consequences for several postulated accidents were also recalculated for EPU conditions. The calculated exposures and control room doses continue to meet the 10CFR50.67 guidelines and General Design Criterion (GDC) 19.

Specific transient analyses are performed, and core operating limits are developed, for each specific fuel cycle. These cycle-specific results, which include the effects of EPU beginning with Cycle 14, are described in USAR Appendix A.

Reload Cycles

Appendix A to the USAR represents the cycle-specific information and analytical results for each reload, which includes the fuel loaded in the core and the respective safety analysis. Appendix A is organized parallel to the USAR chapters for ease of use; appropriate cross-references are provided with the USAR chapters and Appendix A.

The reload safety analysis is based on the GE report, General Electric Standard Application for Reactor Fuel (GESTAR II), described in Reference 4. GESTAR II represents generic information relative to the GE fuel design and analysis and consists of a description of the fuel design and fuel thermal-mechanical, nuclear, and thermal-hydraulic analyses bases. It provides information and methods used to determine reactor limits that are independent of a plant-specific application. Plant-specific information and the transient and accident methods used are given in the United States supplement. Proposed changes to GESTAR II are submitted to the appropriate regulatory body for review and approval. A listing of Nuclear Regulatory Commission (NRC) approved amendments is provided in GESTAR II. All approved changes are incorporated as a revision into the text.

Unit 2 design is intended to be valid for the licensed life of the plant. The supplemental cycle-specific safety analysis assures that the plant can be operated safely and not pose any undue risk to the health and safety of the public. This is accomplished by demonstrating that radioactive releases from the plant for normal operation, anticipated operational occurrences, and postulated accidents meet applicable regulations.

Unit 2 operation for reload cycles must meet various safety requirements defined in the Code of Federal Regulations (CFR). In order to evaluate the safety impact of reload cycles, fuel lattice physics calculations and 3-dimensional simulation, transients, and accident evaluations were performed. NRC approved methodologies described in GESTAR II were used to license the operating states described in Appendix A. Operation in the maximum extended load line limit analysis plus region (MELLLA+), increased core flow (ICF) to 105 percent of rated core flow, two safety/relief valves (SRVs) and one main steam isolation valve (MSIV) out of service (OOS), and single recirculation loop are included in the analyses.

15.0.1 Analytical Objective

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

15.0.2 Analytical Categories

Transient and accident events contained in this report are discussed in individual categories as required by Regulatory Guide (RG) 1.70. The results of the events are summarized in

Table 15.0-1. Each event evaluated is assigned to one of the following applicable categories:

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

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

3. Decrease in Reactor Core Coolant Flow Rate A reduction in the core coolant flow rate threatens to overheat the cladding as the coolant becomes unable to adequately remove the heat generated by the fuel.

4. Reactivity and Power Distribution Anomalies Transient events included in this category are those which cause rapid increases in power which are due to ICF disturbance events. ICF reduces the void content of the moderator increasing core reactivity and power level.

5. Increase in Reactor Coolant Inventory Increasing coolant inventory could result in excessive moisture carry-over to the main turbine.

6. 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.

7. Postulated loss of integrity of components containing radioactive material.

8. Anticipated Transients Without Scram In order to determine the capability of plant design to accommodate an extremely low probability event, a multisystem maloperation situation is postulated.

15.0.3 Event Evaluation

15.0.3.1 Identification of Causes and Frequency Classification

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

Events for which inconclusive data exist are discussed separately within each event section.

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

1. Incident of Moderate Frequency This event may occur once per calendar year to once per 20 yr for a particular plant. It is referred to as an anticipated (expected) operational transient.

2. Infrequent Incident This event may occur during the life of the particular plant (spanning once in 20 yr to once in 100 yr). It is referred to as an abnormal (unexpected) operational transient.

3. Limiting Fault This event is an incident that is not expected to occur but is postulated because its consequences may result in the release of significant amounts of radioactive material. It is referred to as a design basis (postulated) accident.

4. Normal Operation Operations of high frequency are not discussed here, but are examined along with the preceding three frequency groups in the Nuclear Systems Operational Analyses in Appendix 15A.

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

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

1. A release of radioactive material to the environs that exceeds the limits of 10CFR20.

2. Reactor operation induced fuel cladding failure as a direct result of the transient analysis above the minimum critical power ratio (MCPR) uncertainty level (0.1 percent).

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

4. Containment stresses in excess of those allowed for the transient classification by applicable industry codes.

15.0.3.1.2 Unacceptable Results for Infrequent Incidents (Abnormal Operational Transients)

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

1. Release of radioactivity which results in dose consequences that exceed a small fraction of the criteria of 10CFR50.67.

2. Reactor operation induced fuel cladding failure as a direct result of the transient analysis above the MCPR uncertainty level (0.1 percent).

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

4. Containment stresses in excess of those allowed for the transient classification by applicable industry codes.

15.0.3.1.3 Unacceptable Results for Limiting Faults (Design Basis Accidents)

The following are considered to be unacceptable safety results for limiting faults (design basis accidents [DBA]):

1. Radioactive material release that results in dose consequences that exceed the criteria of 10CFR50.67.

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

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

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

5. Radiation exposure to plant operations personnel in the main control room in excess of 5 Rem total effective dose equivalent (TEDE).

15.0.3.2 Sequence of Events and Systems Operations

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

1. Step-by-step sequence of events from initiation to final stabilized condition.

2. Extent to which normally-operating plant instrumentation and controls are assumed to function.

3. Extent to which plant and reactor protection systems (RPS) are required to function.

4. Credit taken for the functioning of normally-operating plant systems.

5. Operation of engineered safety systems that is required.

6. Effect of a single failure or an Operator error on the event.

15.0.3.2.1 Single Failures or Operator Errors

Single active component failure (SACF) criteria have been required and successfully applied on past NRC-approved docket applications to DBA categories only. RG 1.70 infers that a "single failures and Operator errors" requirement should be applied to transient events (both high-, moderate-, and low-probability occurrences) as well as accident (very low probability) situations.

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

RG 1.70 suggests that the transient and accident scenarios should now include "and" (multifailure) event sequences. The format request follows:

1. For initiating occurrence, an equipment failure or an Operator error.

2. For SEF, or Operator error analysis, another equipment failure, or failures and/or another Operator error or errors.

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

<u>Initiating Event Analysis</u>

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

2. Undesired starting or stopping of any single component, or

3. Malfunction or maloperation of any single control device, or

4. Any single electrical component failure, or

5. Any single Operator error.

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

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

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

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

Examples of SOEs are as follows:

1. Increase in power above the established flow control power limits by control rod withdrawal in the specified sequences.

2. Selection and complete withdrawal of a single control rod out of sequence.

3. Incorrect calibration of an average power range monitor (APRM).

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

<u>Single Active Component Failure or Single Operator Failure Analysis</u>

This analysis is as follows:

1. Undesired action or maloperation of a single active component, or

2. Any SOE where Operator errors are defined in this section under Initiating Event Analysis.

Section 15.X.Y.2.3, where X and Y refer to different event sections, discusses the impact and compliance of a single failure

and Operator error for each event in Chapter 15. For events of moderate frequency, the inclusion of any additional failure shifts the event to the infrequent category that has less stringent compliance criteria, i.e., 10CFR50.67 criteria.

The most limiting transient event is a feedwater controller failure with maximum demand without turbine bypass. The increase in ΔCPR (critical power ratio) due to this event for Cycle 1 would be less than 0.11 above the event with turbine bypass. Since this transient is one of only 2- to 3-sec duration, no fuel failure would be expected. In terms of peak pressure limits, the overpressurization protection case described in Chapter 5 bounds the most limiting transient event with the worst single failure or Operator error.

As shown on Figure 15.0-3, a 0.8 conservatism factor on scram reactivity is used for all REDY cases. For ODYN cases, the analytical scram time corresponding to the slowest Technical Specification scram speed is used in combination with the reactivity calculated in the ODYN model. The scram reactivity conservatism has been built into Option A and Option B adders on CPR using a statistical model as described in GE NEDO-24154. For TRACG methods see References 12 and 13.

The key input parameters in Table 15.0-3 used for transient analysis are conservative. For all instrument setpoints, the worst analytical limits to account for accuracy, calibration, and drift uncertainties are used. The nuclear characteristics used for the REDY model include sufficient conservatism. For the ODYN model, sufficient statistical conservatisms by either Option A or Option B adders have been added to the nuclear characteristics. Overall, the input parameters for analysis were chosen in a conservative fashion to produce the worst consequences of various transients in Chapter 15.

15.0.3.3 Core and System Performance

15.0.3.3.1 Introduction

Section 4.4 describes the various fuel failure mechanisms. Avoidance of unacceptable results 1 and 2 (Section 15.0.3.1.1) 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[1]. This criterion is met by demonstrating that incidents of moderate frequency do not result in a MCPR less than the safety limit CPR (1.06 for Cycle 1; as stated in Appendix A for reload cores).

Determination of the steady-state operating limit is accomplished as follows:

1. The change in the CPR which would result in the safety limit CPR being reached is calculated for each event. These values are shown in Table 15.0-1 for Cycle 1 and are discussed in Appendix A for reload cores.

2. The ΔCPR value is then added to the safety limit CPR value to obtain the event-based MCPR except for events whose ΔCPR is calculated using the ODYN computer code model.

3. For events whose ΔCPR is determined by ODYN (all rapid pressurization events), the event-based MCPR is determined in conjunction with correction factors, the ΔCPR, and the safety limit ΔCPR. These correction factors are explained in detail in the Bases for Technical Specification 3.2.2.2 and the Core Operating Limits Report (COLR).

The resultant operating limit CPR values are given in Table 15.0-2 for Cycle 1. See Appendix A for details on determination of the cycle specific steady-state operating limit using TRACG.

The operating limit MCPR is the maximum value of the event MCPRs calculated from the transient analysis. The maximum calculated transient MCPR is depicted by the solid line on Figure 15.0-1 for Cycle 1, and is specified in the COLR for reload cores. Maintaining the CPR operating limit at or above this operating limit assures that the safety limit CPR is never violated.

The one-dimensional core model used in the ODYN computer code is utilized in the analysis of key pressurization events. ODYN provides improved core thermal-hydraulic, reactivity feedback, and power distribution simulation which are important in pressurization events. System modeling, which features the prediction of interactive response of the RPSs and their principal controllers, is basically the same as used in the point kinetics reactor model code. A detailed description of the analytical models may be found in NEDO-10802 and NEDE-24154-P[2,3]. Starting with Cycle 15, ODYN has been replaced with TRACG methods for transient analysis. See Appendix A and References 12 and 13 for details on the TRACG model.

Maintaining the MCPR greater than the safety MCPR limit is a sufficient, but not necessary, condition to assure that no fuel damage occurs (Section 4.4). 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 (Sections 4.4 and 6.3).

15.0.3.3.2 Input Parameters and Initial Conditions for Analyzed Events

In general, the events analyzed in this section have values for input parameters and initial conditions as specified in Table 15.0-3. The input parameters and initial conditions for the EPU evaluation are listed in Table 15.0-3a. Analyses which assume data inputs different from these values are designated accordingly in the appropriate event discussion.

The initial core transient analysis was performed using REDY/ODYN methodology. After obtaining startup test data on steam line pressure drop, turbine bypass capacity, and feedwater controller runout flow, a reanalysis of five limiting cases was performed using GEMINI methodology[4]. The five limiting cases from the standpoint of MCPR are:

1. Feedwater controller failure - maximum demand,
2. Generator load rejection without bypass,
3. Feedwater controller failure - maximum demand - without bypass,
4. Feedwater controller failure - maximum demand - with end-of-cycle recirculation pump trip (EOC RPT) OOS, and
5. Generator load rejection with bypass failure and EOC RPT OOS.

The reanalysis results were bounded by the initial analysis results, thus demonstrating the conservatism of the set of input conditions as listed in Table 15.0-3 for the initial core.

15.0.3.3.3 Initial Power/Flow Operating Constraints

The analysis basis for the most limiting transient safety analyses uses an adjusted simulation of licensed (100 percent) thermal power at ICF (105 percent). At least 102 percent of rated power is considered either directly as the initial condition or as a statistical adjustment to the output (e.g., TRACG and GEMINI MCPR evaluations). This is the highest power operating condition of a bounded operating power/flow map which, in response to most classified abnormal operational transients, yields the minimum pressure and thermal margins of any operating point within the bounded map. Referring to Figure 15.0-2, this power condition represents the rated power limit line G-F-E for core flow between 80 and 105 percent. At lower core flows, the upper bound is the rod line which passes through the 100 percent of current licensed thermal power (CLTP)/80 percent of rated core flow (RCF) point (line E-B). The lower bound is the pump and valve cavitation limit (line L-I), the right bound is 105 percent of the RCF line G-I, and the left bound is line B-L-Q. The power/flow map G-E-B-Q-L-I-G represents the operational constraints for abnormal operational transient evaluations for dual-loop operation.

For single-loop operation (SLO) (see Appendix 15B), the operating map is reduced to the region P-M-Q-N-O-P.

The power/flow map for EPU/MELLLA+ is shown on Figure 15.0-2a.

Certain special events are evaluated at other than the previously mentioned conditions. These conditions are discussed in conjunction with the appropriate events.

15.0.3.3.4 Results

The results of analytical evaluations are provided for each event. For Cycle 1, important parameters are listed in Table 15.0-1. From the data in Table 15.0-1, an evaluation of the limiting event for that particular category and parameter can be made. Evaluation is made with respect to vessel pressure, water level, and fuel thermal margin (using GETAB[1]). Note that limiting cases are also provided for certain special operating conditions involving equipment being OOS (see Section 15.0.3.6). Table 15.0-4 compares the GE-calculated amount of failed fuel to that used in worst-case radiological calculations for applicable accidents.

The results of analytical evaluations of limiting transients for reload cores, including the effects of operation at the uprated core thermal power, are discussed in Appendix A, Section A.15.

15.0.3.4 Barrier Performance

This section primarily evaluates the performance of the RCPB and the containment system during transients and accidents. During transients that occur with no release of coolant to the containment, only RCPB performance is considered. If release to the containment occurs as in the case of limiting faults, then challenges to the containment are evaluated as well.

15.0.3.5 Radiological Consequences

In this section, the consequences of radioactivity release during three types of events are considered: 1) incidents of moderate frequency (anticipated operational transients), 2) infrequent incidents (abnormal operational transients), and 3) limiting faults (DBAs). For all events whose consequences are limiting, a detailed quantitative evaluation is presented. For nonlimiting events, a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

For limiting faults (DBAs) two quantitative analyses are considered:

1. The first is based on conservative assumptions considered to be acceptable to the NRC for the purposes of worst-case bounding the event and determining the adequacy of the plant design to meet 10CFR50.67 criteria. This analysis is referred to as the DBA.

2. The second is based on realistic assumptions considered to reflect expected radiological consequences. This analysis is referred to as the realistic analysis.

Results for both are shown to be within NRC guidelines.

15.0.3.6 Special Capability Studies

Special capabilities of the plant to operate outside of the base case domain have also been analyzed. Each of these operating conditions/options is supported by its own systems performance evaluation and requires modifications to pertinent limits such as Technical Specifications and maximum average planar linear heat generation rate (MAPLHGR). These case studies are documented in the following appendices:

Appendix 15B	Recirculation System Single Loop Operation
Appendix 15C	Two Safety Relief Valves (ADS Function) Out Of Service
Appendix 15D	One Main Steam Isolation Valve Out of Service
Appendix 15G	Maximum Extended Load Line Limit Analysis

Some of the special cases have also been postulated to occur concurrently in order to further expand the operating domain. The case combination is summarized in Table 15.0-6.

15.0.4 Nuclear Safety Operational Analysis Relationship

Appendix 15A is a comprehensive, total plant, system-level, qualitative failure modes and effects analysis (FMEA), relative to all the Chapter 15 events considered, the protective sequences utilized to accommodate the events and their effects, and the systems involved in the protective actions. Interdependency of analysis and cross-referral of protective actions is an integral part of this chapter and Appendix 15A.

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 Loss of Instrument Air

Although a complete analysis of the loss of instrument air transient has not been performed, an expected sequence of events and Operator actions for this event are provided below.

The Unit 2 design uses excess flow check valves to subdivide the system into discrete elements. Most line breaks would affect only certain portions of the system. Additionally, recent operating experience indicates that complete loss of instrument air is a remote possibility. However, reports of partial loss of

instrument air appear to have had no serious effects on the capability to shut down the reactor safely.

Further, the stored capacity of the instrument air system (IAS) provides additional assurance that a total loss of instrument air is not expected.

The IAS is comprised of two parts. The first is the compressed air system, including the control and service air system described in Sections 9.3.1.1 and 9.3.1.2. The second is a nitrogen system (a subsystem of instrument air) for all pneumatic valves inside the primary containment. Since the nitrogen and air subparts are separate and independent, the loss of either system does not affect the capability of the other subsystem to perform its design function.

In the event of a loss of the compressed air subsystem, the following can be expected to occur in sequence, depending on the location of the failure.

1. CRD system - The scram inlet and outlet valves will open, shutting down the reactor. The CRD flow control valve will close to approximately 2 percent open. The drain and vent valves for the scram discharge volume (SDV) will close.

2. Reactor water cleanup (RWCU) system - During system operation, all precoat isolation valves and the reject valves to radwaste and the main condensers will remain closed. During maintenance operations, the inlet and outlet isolation valves remain closed.

3. Containment atmosphere control valves and containment ventilation isolation valves fail closed on loss of instrument air.

4. The steam supply to the steam jet air ejectors (SJAE) will be lost as a result of MSIV closure, eventually resulting in loss of condenser vacuum.

5. The MSR block (admission valves) will close.

6. Spent fuel pool cooling and cleanup (SFC) system - The demineralized makeup to the pool skimmer surge tank will fail closed, but Class 1E level instrumentation is provided with a safety-related makeup source (service water). Also, the fuel pool filter/demineralizer inlet and outlet valves will close, and the valves supplying the heat exchangers will open, allowing recirculation and cooling of the fuel pool.

7. The ventilation supply isolation dampers to the reactor building ventilation fail closed.

8. The standby gas treatment system (SGTS) will align itself to take suction from the secondary containment.

9. The reactor core isolation cooling (RCIC) steam line drain and residual heat removal (RHR) heat exchanger steam supply control valves will close upon loss of air. The RHR heat exchanger steam supply is normally closed by a motor-operated valve (MOV). RHR steam supply control valves also are normally closed. The RCIC lube oil cooler cooling water pressure control valve will fail open upon loss of air.

10. The minimum flow bypasses for the condensate, condensate booster, and feedwater pumps will open, bypassing feedwater to the condenser. This could cause the reactor water level to drop to Level 3, thereby initiating a scram signal.

11. Automatic hotwell level control is lost as the air-operated makeup valve fails open. Reject valves fail closed.

12. Standby liquid control (SLC) - The level indication for the storage tank will decrease to zero.

13. MSIV will receive a "Close" signal due to loss of condenser vacuum or on low main steam line pressure with the mode switch on "Run," or loss of instrument air.

14. Loss of instrument air has no effect on high-pressure core spray (HPCS).

15. Nonsafety systems which do not affect safe plant shutdown are affected by complete or partial loss of instrument air. However, complete or partial loss of air does not adversely affect any safety systems required to shut down the plant.

The Operator response will be based upon the degree and failure of the IAS. Depending on the equipment affected, the Operator may take the following actions, although not necessarily in this sequence:

1. Confirm that the reactor has scrammed and is subcritical.

2. Operate RCIC and/or HPCS according to normal procedures to maintain normal reactor water level.

3. Continue the cooldown of the reactor with the RHR system after reactor pressure and temperature have decreased to the operating limits of RHR.

4. Upon receipt of alarm of loss of reactor building ventilation, manually initiate operation of the SGTS.

5. Manually control hotwell level as required.

6. Initiate a scheduled surveillance of the SLC storage tank to confirm proper level.

The instrument air (compressed air) is designed to fail to a position that is consistent with the safe shutdown of the plant. Although specific Operator action to shut down the plant safely is not required after a loss of instrument air, the plant procedures will reflect the best actions to be taken.

In the event of the loss of the nitrogen subsystem, the following can be expected to occur in sequence, depending on the location of the failure:

1. The main steam SRVs will not open as a direct result of loss of any nitrogen supply. There is sufficient nitrogen in each accumulator (one for safety relief and automatic depressurization system (ADS), if equipped) to provide actuation of each relief valve with additional capacity available from the nitrogen accumulators in the reactor building for the ADS-designed SRVs.

2. Isolation valves inside the containment which use pneumatic operators are normally closed during operation. They are also fail closed on loss of nitrogen.

3. MSIVs are pneumatic to open, spring and pneumatic to close valves, and loss of nitrogen will cause the MSIVs to close.

The response of plant and Operator actions are the same as those for a loss of instrument air previously described.

15.0.6 Effect of Nonsafety Grade Equipment

Cycle-specific information is discussed in Appendix A, Section A.15.0.6.

The performance of some structures, systems, and components that are not identified as safety related is assumed in transient analyses.

Table 15.0-5 lists the nonsafety-grade equipment that has been assumed in the mitigation of the consequences. The assumed performance of these nonsafety-related equipment is based on extensive failure rate data for equipment of similar design and quality requirements.

Among these nonsafety-grade systems, the failure of the Level 8 trip and the failure of the turbine bypass are the events that would affect ΔCPR. For the limiting transient, a feedwater controller failure maximum demand, the assumed failure of either the Level 8 trip or the turbine bypass would have negligible effect on the final CPR such that the analytical conclusion with respect to the fuel integrity as presented in Section 15.1.2 is still valid. In addition, the Unit 2 Technical Specifications include appropriate provisions regarding operability, allowable values, and surveillance testing of these equipment.

The peak vessel pressures for the analysis of transients not taking credit for nonsafety-grade systems and components are bounded by the peak pressure limit of the overpressure protection analysis described in Chapter 5.

15.0.7 References

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

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

3. Qualification of One Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154 and NEDE-24154-P (Proprietary).

4. GESTAR II, General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-9US; Approved by Letter from G. C. Lainas (NRC) to J. S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, Supplement to Amendment 11," March 22, 1986.

5. Licensing Topical Report, Power Uprate Licensing Evaluation for Nine Mile Point Nuclear Power Station, Unit 2, NEDC-31994P, Revision 1, May 1993.

6. General Electric Nuclear Energy, Generic Guidelines for GE Boiling Water Reactor Power Uprate, NEDC-31897P-1, May 1991.

7. General Electric Nuclear Energy, Generic Evaluations of GE BWR Power Uprate, NEDC-31984P, July 1991.

8. GE Hitachi Nuclear Energy, Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate, NEDC-33351P, Revision 0, May 2009.

9. GE Nuclear Energy, Constant Pressure Power Uprate, NEDC-33004P-A, Revision 4, Class III (Proprietary), July 2003; and NEDO-33004, Class I (Non-proprietary), July 2003.

10. GE Nuclear Energy, Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate, NEDC-32424P-A,

Class III (Proprietary), February 1999; and NEDO-32424, Class I (Non-proprietary), April 1995.

11. GE Nuclear Energy, Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate, NEDC-32523P-A, Class III (Proprietary), February 2000; NEDC-32523P-A, Supplement 1 Volume I, February 1999; and Supplement 1 Volume II, April 1999.

12. Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, NEDE-32906P, Supplement 3-A, Revision 1, April 2010.

13. TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses, NEDE-32906P-A, Revision 3, September 2006.

TABLE 15.0-1

SUMMARY OF EVENT RESULTS - CYCLE 1[1]

(Cycle-specific values are discussed in Appendix A, Section A.15.0.3)

Section I.D.	Figure I.D.	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)	Change of Critical Power Ratio[2]	Frequency Category[3]	No. of Valves First Blowdown	Duration of Blowdown (sec)
15.1		Decrease in core coolant temperature									
15.1.1	15.1-1	Loss of feedwater heater, automatic flow control	112	1,020	1,060	1,007	105.9	[4]	a	0	0
15.1.1	15.1-2	Loss of feedwater heater, manual flow control	124	1,032	1,072	1,017	116.9	0.15	a	0	0
15.1.2	15.1-3	Feedwater control failure, max demand	252.4	1,175	1,203	1,170	114.3	0.16	a	18	4.5
15.1.3	15.1-4	Pressure regulator open-130% flow	104	1,117	1,141	1,114	100.3	[5]	a	10	6.6
15.2		Increase in reactor pressure									
15.2.2	15.2-1	Generator load rejection, bypass on, RPT on[4]	265.2	1,169	1,197	1,168	107.3	0.10	a	18	4.8
15.2.2	15.2-2	Generator load rejection, bypass off, RPT on[4]	407.1	1,204	1,233	1,203	113.2	0.16	b	18	6.9
15.2.3	15.2-3	Turbine trip, bypass on, RPT on[4]	213	1,167	1,194	1,165	104.5	0.07	a	18	4.8
15.2.3	15.2-4	Turbine trip, bypass off, RPT on[4]	331	1,203	1,232	1,200	111.1	0.14	b	18	6.8

TABLE 15.0-1 (Cont'd.)

Section I.D.	Figure I.D.	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)	Change of Critical Power Ratio[2]	Frequency Category[3]	No. of Valves First Blowdown	Duration of Blowdown (sec)
15.2.4	15.2-5	Main steam line isolation, position scram	138	1,184	1,215	1,177	100.1	[5]	a	18	5.8
15.2.5	15.2-6	Loss of condenser vacuum at 2 in/sec	215	1,167	1,194	1,165	104.7	[5]	a	18	N/A
15.2.6	15.2-7	Loss of auxiliary power transformer	104	1,140	1,156	1,137	100.1	[5]	a	18	6.4
15.2.6	15.2-8	Loss of all grid connections	183	1,164	1,187	1,161	102.4	[5]	a	18	N/A
15.2.7	15.2-9	Loss of all feedwater flow	104	1,093	1,103	1,093	100.0	[5]	a	2	6.7
15.3		Decrease in reactor coolant system flow rate									
15.3.1	15.3-1	Trip of one recirculation pump motor	104	1,093	1,106	1,092	100.0	[5]	a	2	6.1
15.3.1	15.3-2	Trip of both recirculation pump motors	104	1,106	1,119	1,103	100.1	[5]	a	6	6.4
15.3.2	15.3-3	Fast closure of one recirculation valve	104	1,100	1,115	1,099	100.1	[5]	a	2	7.8
15.3.2	15.3-4	Fast closure of two recirculation valves	104	1,109	1,120	1,107	100.1	[5]	a	6	6.8
15.3.3	15.3-5	Seizure of one recirculation pump	104	1,108	1,120	1,105	100.1	[5]	c	6	6.8
15.4		Reactivity and power distribution anomalies									

TABLE 15.0-1 (Cont'd.)

Section I.D.	Figure I.D.	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)	Change of Critical Power Ratio[2]	Frequency Category[3]	No. of Valves First Blowdown	Duration of Blowdown (sec)
15.4.4	15.4-6	Startup of idle recirculation loop	122	980	996	974	154.5	[6]	a	0	0
15.4.5	15.4-7	Fast opening of one recirculation valve	308	974	994	970	142.2	[6]	a	0	0
15.4.5	15.4-8	Fast opening of two recirculation valves	241	971	991	966	135.1	[6]	a	0	0
15.5		Increase in reactor coolant inventory									
15.5.1	15.5-1	Inadvertent HPCS pump start	104	1,020	1,060	1,007	100.1	[5]	a	0	0

[1] Pressurization events analyzed with ODYN code (see Note 2).
[2] CPR based on an initial CPR which yields an MCPR of 1.06.
[3] a = Incidents of moderate frequency
b = Infrequent incidents
c = Limiting faults
[4] ODYN results (not including adjustment factors) are based on EOC-1 nuclear data.
[5] No significant change in CPR.
[6] Not started from full power and no significant change in CPR.

TABLE 15.0-2

REQUIRED OPERATING LIMIT CPR VALUES - CYCLE 1

(Cycle-specific values are discussed in Appendix A, Section A.15.0.3)

Pressurization Events	CPR (Option A)[1]	CPR (Option B)[1]
Load rejection without bypass	1.28[2]	1.19
Turbine trip without bypass	1.25	1.17
Feedwater controller failure	1.27	1.24[3]
Load rejection	1.21	1.13
Turbine trip with bypass	1.18	1.10

Nonpressurization Events	CPR
Rod withdrawal error	1.24
Loss of feedwater heater	1.21

(1) Includes adjustment factors as specified in the NRC safety evaluation report on ODYN, NEDO-24154 and NEDE-24154-P.

(2) Required OLCPR using Option A and neglecting infrequent category of the turbine generator trip events with bypass failure.

(3) Required OLCPR using Option B adjustment factor regardless of frequency category of the turbine generator trip events with bypass failure.

TABLE 15.0-3

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS

(Cycle-specific values are provided in Appendix A, Table A.15.0-4)

	Original Rating	Power Uprate
1. Thermal power level, MWt Warranted value Analysis value	 3,323 3,466(4)	 3,467 3,467/3,536/3,988/4,067(9)
2. Steam flow, lb/hr Warranted value Analysis value	 $14.296x10^6$ $15.01x10^6$(4)	 $15.0x10^6$ $15.0x10^6/15.35x10^6$
3. Core flow, lb/hr Rated core flow range, % NBR	$108.5x10^6$(5) -	$108.5x10^6$ 87-105
4. Feedwater flow rate, lb/hr Warranted value Analysis value	 $14.296x10^6$ $15.01x10^6$(4)	 $15.0x10^6$ $15.0x10^6/15.35x10^6$
5. Feedwater temperature, °F	425(4)	425/427
6. Vessel dome pressure, psig	1,020(1)(4)	1020
7. Vessel core pressure, psig	1,031(4)	1031
8. Turbine bypass capacity, % NBR	25(2)	21.7
9. Core coolant inlet enthalpy, Btu/lb	528.3	528 to 530
10. Turbine inlet pressure, psig	960(1)	988
11. Fuel lattice	P8x8R	See Appendix A
12. Core average gap conductance, Btu/sec-ft^2-°F	0.1744(4)	See Appendix A
13. Core bypass flow, %	11.84	See Appendix A
14. Required MCPR operating limit	See Table 15.0-2 and Figure 15.0-1	See Appendix A
15. MCPR safety limit	1.06	See Appendix A
16. Doppler coefficient (-)¢/°F Nominal EOC-1 Analysis data(3)	 0.184 0.175	 (6) (6)

TABLE 15.0-3 (Cont'd.)

	Original Rating	Power Uprate
17. Void coefficient (-)¢/% rated voids Nominal EOC Analysis data for power increase events Analysis data for power decrease events	 9.68 12.7[(3)] 8.73[(3)]	 (6) (6) (6)
18. Core average rated void fraction, %	41.28[(3)]	4.13 to 47.7
19. Scram reactivity, $ K Analysis data	 Figure 15.0-3[(3)]	 (6)
20. Control rod drive speed, Position versus time	 Figure 15.0-3[(4)]	 Figure 15.0-3
21. Jet pump ratio, M	2.33	See Appendix A
22. SRV capacity, % NBR @ 1,212 psig Manufacturer Quantity installed	113.8% Dikkers 18	 Dikkers 18/16[(7)]
23. Relief function delay, sec	0.4	0.4
24. Relief function response Time constant, sec	 0.1	 0.1
25. Safety function delay, sec	0.0	0.0
26. Safety function response Time constant, sec	 0.2	 0.2
27. Setpoints for SRVs Safety function, psig Relief function, psig	 1177, 1187, 1197, 1207, 1217 1106, 1116, 1126, 1136, 1146	 1200, 1210, 1221, 1231, 1241[(7)] 1121, 1131, 1141, 1151, 1161[(7)]
28. Number of valve groupings simulated Safety function, No. Safety function, No.	 5 5	 5 5
29. High flux trip, % NBR Analysis setpoint	 126.2[(4)] (121x1.043)	 123[(8)]
30. High pressure scram setpoint, psig	1,071	1,086

TABLE 15.0-3 (Cont'd.)

		Original Rating	Power Uprate
31.	Vessel level trips, ft above bottom of separator skirt bottom Level 8 - (L8) Level 4 - (L4) Level 3 - (L3) Level 2 - (L2)	 6.175 3.75 1.75 -4.708	 6.175 3.75 1.75 -4.708
32.	APRM simulated thermal power trip, % NBR Analysis setpoint	 122[4] (117x1.043)	 118
33.	Recirculation pump trip delay, sec	0.19	0.19
34.	Recirculation pump trip inertia time Constant for analysis, sec[3a]	 min 4.0 to max 6.0	 4.0 to 6.0
35.	Total steam line volume, ft^3	4,012	4,036

(1) A pressure drop of 29 psid was measured from vessel dome to turbine inlet in N2-SUT-20-WR. Transient results from GEMINI using the measured values are bounded by results from REDY/ODYN using the nominal 60 psid as tabulated in Items 6 and 10 of this table.

(2) The bypass capacity was measured to be 22.8 percent NBR in N2-SUT-27. Transient results from GEMINI using this measured value are bounded by results from REDY/ODYN using the nominal 25 percent as tabulated.

(3) Reactivity parameters applicable only for events analyzed with REDY. Values are calculated with the code for EOC-1 condition.

(3a) 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
n = Rated pump speed, rps
g = Gravitational constant, ft/sec
T_o = Pump shaft torque, lb-ft

TABLE 15.0-3 (Cont'd.)

(4) Cycle 1 limiting transients were reanalyzed as identified in Section 15.0.3.3.2. These analyses were based upon the GEMINI methods and startup test data as identified in Note (2), Note (3) and Section 15.1.2.3.2. The GEMINI methods use nominal values for input parameters with uncertainties applied to the results. The GEMINI methodology is accepted by the NRC per SER approving Amendment 11 to NEDE-24011-P. The following nominal values were used as initial input data: Thermal Power = 3323 MW, Steam Flow = 14.3 MLB/HR, FW Flow = 14.3 MLB/HR, Vessel Dome Pressure = 1005 psig, Vessel Core Pressure = 1016 psig, FW Temperature = 420°F, High Flux Trip = 121% NBR, Simulated Thermal Flux = 117% NBR. The gap conductors are calculated by a detailed model, GESTR. The control rod scram time is based upon representative times of BWR4s and BWR5s. The nominal speed assumed in these analyses is the following: (percent inserted, time (sec)), (0, 0), (0, 0.2), (5, 0.324), (20, 0.694), (50, 1.459), (90, 2.535), (100, 2.80). Detailed information on these analyses is contained in GE document EAS-48-0889. Chapter 15 analyses bound these reanalyses.

(5) The reanalysis described in Note (4) was performed at 105 percent rated core flow for the five limiting transients identified in FSAR Section 15.0.3.3.2. The reanalysis at 105 percent of rated core flow was performed to support implementation of increased core flow for Cycle 1 operation.

(6) Values are calculated within the analysis computer model for EOC conditions.

(7) Sixteen safety valves are assumed to be operational; the two lowest setpoint valves are assumed to be out of service for all transient cases.

(8) Uprate analysis uses slightly higher relative setpoints to allow greater flexibility in selecting high flux scram setpoints.

(9) For AST, the radiological consequence analyses were performed for a power level corresponding to 120% of the original licensed thermal power level (4,067 MWt).

TABLE 15.0-3a

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS
(Cycle-specific values are provided in Appendix A, Table A.15.0-4)

	Original Rating	Extended Power Uprate
1. Thermal power level, MWt Warranted value Analysis value	 3,323 3,466[4]	 3,988 3,988/4,068
2. Steam flow, lb/hr Warranted value Analysis value	 $14.296\text{x}10^{6}$ $15.01\text{x}10^{6}$[4]	 $17.64\text{x}10^{6}$ $17.64\text{x}10^{6}/18.07\text{x}10^{6}$
3. Core flow, lb/hr Rated core flow range, %NBR	$108.5\text{x}10^{6}$[5] -	$108.5\text{x}10^{6}$ 80-105
4. Feedwater flow rate, lb/hr Warranted value Analysis value	 $14.296\text{x}10^{6}$ $15.01\text{x}10^{6}$[4]	 $17.60\text{x}10^{6}$ $17.60\text{x}10^{6}/18.03\text{x}10^{6}$
5. Feedwater temperature, °F	425[4]	440.5/442.9
6. Vessel dome pressure, psig	1,020[1][4]	1,020/1,035
7. Vessel core pressure, psig	1,031[4]	1,034
8. Turbine bypass capacity, %NBR	25[2]	18.5
9. Core coolant inlet enthalpy, Btu/lb	528.3	528.9/530.7
10. Turbine inlet pressure, psig	960[1]	976
11. Fuel lattice	P8x8R	See Appendix A
12. Core average gap conductance, Btu/sec-ft^2-°F	0.1744[4]	See Appendix A
13. Core bypass flow, %	11.84	See Appendix A
14. Required MCPR operating limit	See Table 15.0-2 and Figure 15.0-1	See Appendix A
15. MCPR safety limit	1.06	See Appendix A
16. Doppler coefficient (-)¢/°F Nominal EOC-1 Analysis data[3]	 0.184 0.175	 [6] [6]

TABLE 15.0-3a (Cont'd.)

		Original Rating	Extended Power Uprate
17.	Void coefficient (-)¢/% rated voids Nominal EOC Analysis data for power increase events Analysis data for power decrease events	 9.68 12.7(3) 8.73(3)	 (6) (6) (6)
18.	Core average rated void fraction, %	41.28(3)	45.8 - 52.3
19.	Scram reactivity, $ K Analysis data	 Figure 15.0-3(3)	 (6)
20.	Control rod drive speed, Position vs. time	 Figure 15.0-3(4)	 Figure 15.0-3
21.	Jet pump ratio, M	2.33	See Appendix A
22.	SRV capacity, % NBR @ 1,212 psig Manufacturer Quantity installed	113.8% Dikkers 18	91% (at 1145 psig) Dikkers 18/16(7)
23.	Relief function delay, sec	0.4	0.4
24.	Relief function response Time constant, sec	 0.1	 0.1
25.	Safety function delay, sec	0.0	0.0
26.	Safety function response Time constant, sec	 0.2	 0.2
27.	Setpoints for SRVs Safety function, psig Relief function, psig	 1177, 1187, 1197, 1207, 1217 1106, 1116, 1126, 1136, 1146	 1200, 1210, 1221, 1231, 1241(7) 1121, 1131, 1141, 1151, 1161(7)
28.	Number of valve groupings simulated Safety function, No. Safety function, No.	 5 5	 5 5
29.	High flux trip, % NBR Analysis setpoint	 126.2(4) (121x1.043)	 123(8)
30.	High pressure scram setpoint, psig	1,071	1,086

TABLE 15.0-3a (Cont'd.)

		Original Rating	Extended Power Uprate
31.	Vessel level trips, ft above bottom of separator skirt bottom Level 8 - (L8) Level 4 - (L4) Level 3 - (L3) Level 2 - (L2)	 6.175 3.75 1.75 -4.708	 6.175 3.75 1.225 -4.708
32.	APRM simulated thermal power trip, % NBR Analysis setpoint	 122(4) (117x1.043)	 118
33.	Recirculation pump trip delay, sec	0.19	0.199
34.	Recirculation pump trip inertia time Constant for analysis, sec[3a]	 min 4.0 to max 6.0	 4.0 to 6.0
35.	Total steam line volume, ft^3	4,012	4,036

[1] A pressure drop of 29 psid was measured from vessel dome to turbine inlet in N2-SUT-20-WR. Transient results from GEMINI using the measured values are bounded by results from REDY/ODYN using the nominal 60 psid as tabulated in Items 6 and 10 of this table.

[2] The bypass capacity was measured to be 22.8 percent NBR in N2-SUT-27. Transient results from GEMINI using this measured value are bounded by results from REDY/ODYN using the nominal 25 percent as tabulated.

[3] Reactivity parameters applicable only for events analyzed with REDY. Values are calculated with the code for EOC-1 condition.

[3a] 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
n = Rated pump speed, rps
g = Gravitational constant, ft/sec
T_o = Pump shaft torque, lb-ft

TABLE 15.0-3a (Cont'd.)

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(4) Cycle 1 limiting transients were reanalyzed as identified in Section 15.0.3.3.2. These analyses were based upon the GEMINI methodsand startup test data as identified in Note (2), Note (3) and Section 15.1.2.3.2. The GEMINI methods use nominal values for input parameters with uncertainties applied to the results. The GEMINI methodology is accepted by the NRC per SER approving Amendment 11 to NEDE-24011-P. The following nominal values were used as initial input data: Thermal Power = 3323 MW, Steam Flow = 14.3, MLB/HR, FW Flow = 14,3 MLB.HR, Vessel Dome Pressure = 1005 psig, Vessel Core Pressure = 1016 psig, FW Temperature = 420°F, High Flux Trip = 121% NBR, Simulated Thermal Flux = 117% NBR. The gap conductors are calculated by a detailed model, GESTR. The control rod scram time is based upon representative times of BWR 4s and BWR 5s. The nominal speed assumed in these analyses is the following: (percent inserted, time (sec)), (0, 0), (0, 0.2), (5, 0.324), (20, 0.694), (50, 1.459), (90, 2.535), (100, 2.80). Detailed information on these analyses is contained in GE document EAS-48-0889. Chapter 15 analyses bound these reanalyses.

(5) The reanalysis described in Note (4) was performed at 105 percent rated core flow for the five limiting transients identified in FSAR Section 15.0.3.3.2. The reanalysis at 105 percent of rated core flow was performed to support implementation of increased core flow for Cycle 1 operation.

(6) Values are calculated within the analysis computer model for EOC conditions.

(7) Sixteen safety valves are assumed to be operational; the two lowest setpoint valves are assumed to be out of service for all transient cases.

(8) The EPU analysis uses slightly higher relative setpoints to allow greater flexibility in selecting high flux scram setpoints.

TABLE 15.0-4

SUMMARY OF ACCIDENTS

		Failed Fuel Rods	
Section	Title	GE-Calculated Value	NRC Worst-Case Assumption
15.3.3	Seizure of one recirculation pump	None	None
15.3.4	Recirculation pump shaft break	None	None
15.4.9	Rod drop accident	<770	770
15.6.2	Instrument line break	None	None
15.6.4	Steam system pipe break outside containment	None	None
15.6.5	LOCA within RCPB inside containment	None	100%
15.6.6	Feedwater line break - outside containment	None	None
15.7.1	Main condenser gas treatment system failure	N/A	N/A
15.7.3	Liquid radwaste tank failure	N/A	N/A
15.7.4	Fuel handling accident	<125	*
15.7.5	Cask drop accident	None	None

* Two entire fuel assemblies.

TABLE 15.0-5

NONSAFETY-GRADE SYSTEMS/COMPONENTS ASSUMED IN FSAR TRANSIENT ANALYSIS

FSAR Section	Transient	Nonsafety-Grade System or Component
MODERATE FREQUENCY EVENTS		
15.1.2	Feedwater Controller Failure with Maximum Demand	Level 8 Turbine and Feedwater Pump Trip, Turbine Bypass, Relief Valves[(1)]
15.1.3	Pressure Regulator Failure, Open	Relief Valves
15.2.2	Load Rejection	Turbine Bypass, Relief Valves
15.2.3	Turbine Trip	Turbine Bypass, Relief Valves
15.2.4	Closure of all MSIVs	Relief Valves
15.2.5	Loss of Condenser Vacuum	Turbine Bypass, Relief Valves
15.2.6	Loss of Ac Power	Turbine Bypass, Relief Valves
15.2.7	Loss of All Feedwater Flow	Recirculation Runback, Relief Valves
15.3.1	Trip of One or Both Recirculation Pumps	Level 8 Turbine and Feedwater Pump Trip, Turbine Bypass, Relief Valves
15.3.2	Recirculation Flow Control Failure with Decreasing Flow	Level 8 Turbine and Feedwater Pump Trip, Turbine Bypass, Relief Valves
15.4.2	Rod Withdrawal Error at Power	Rod Block Monitor
INFREQUENT EVENTS		
15.2.2	Load Rejection w/o Bypass	Relief Valves
15.2.3	Turbine Trip w/o Bypass	Relief Valves
LIMITING EVENTS		
15.3.3	Recirculation Pump Seizure	Level 8 Turbine and Feedwater Pump Trip, Turbine Bypass, Relief Valves

TABLE 15.0-5 (Cont'd.)

FSAR Section	Transient	Nonsafety-Grade System or Component
15.3.4	Recirculation Pump Shaft Break	Level 8 Turbine and Feedwater Pump Trip, Turbine Bypass, Relief Valves

(1) "Relief Valves" refers to nonsafety-grade instrumentation in the relief mode of the SRVs.

NOTE: Level 8 Trip itself provides a safety-grade initiation signal and is then isolated from the nonsafety-related controls circuitry to initiate turbine and feedwater pump trip.

TABLE 15.0-6

OPERATING OPTIONS ANALYSIS

OPTIONS	ALL ELSE OPER'L**	EOC-RPT OOS	TURB BP OOS	2 SRV OOS
a. Standard* + MELLL	X	X	X	X
b. SLO + Std	X	X	X	X
c. 1 MSIV OOS + Std + MELLL	X	NA	NA	X
d. SLO + 1 MSIV OOS + Std	X	NA	NA	X

X - Analyzed concurrently
NA - Concurrent OOS conditions not analyzed
* - Standard operational analysis includes ICF and 2 SRV OOS
** - "All" only applies to features discussed in this table
EOC-RPT - End of cycle recirculation pump trip
TURB BP - Turbine bypass
OOS - Out of service
SRV - Safety relief valve (ADS function)
SLO - Single loop operation
MELLL - Maximum extended load line limit
ICF - Increased core flow
MSIV - Main steam isolation valve

15.1 DECREASE IN REACTOR COOLANT TEMPERATURE

Cycle-specific analyses are discussed in Appendix A, Section A.15.1.

15.1.1 Loss of Feedwater Heating

This transient event is evaluated for each reload as a potentially limiting event and was one of the events evaluated for operation at uprated (3,467 MWt) power conditions.[3] Results of the cycle-specific evaluation at uprated power conditions are provided in Appendix A.

The LFWH event rarely sets the OLMCPR, which is confirmed each reload. The severity of the LFWH event is primarily dependent upon the LFWH change in temperature, which does not change with EPU. As a result, the event is not evaluated for EPU.

15.1.1.1 Identification of Causes and Frequency Classification

15.1.1.1.1 Identification of Causes

Feedwater heating can be degraded in the following ways:

1. Steam extraction line to heater is closed.

2. Condensate is bypassed around heater.

The first case produces a gradual cooling of the feedwater. In the second case, the 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 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 its 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 assuming conditions of a 100°F loss and full power. The probability of occurrence of this event is regarded as small.

15.1.1.2 Sequence of Events and Systems Operation

15.1.1.2.1 Sequence of Events

Tables 15.1-1 and 15.1-2 list the sequence of events for this transient and its effect on various parameters is shown on Figures 15.1-1 and 15.1-2.

In the automatic flux/flow control mode, the reactor settles out at a lower recirculation flow with no change in steam output.

In the manual flow control mode, a reactor scram occurs. The sequence of events is listed in Table 15.1-2.

No Operator actions are required.

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 will occur in the plant instrumentation and controls, plant protection, and RPSs.

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

A description of the simulated thermal power trip is provided in Section 7.2.1.2, Average Power Range Monitors (APRM), and in Table 7.6-6.

The standard NRC Technical Specifications cover the limiting operating conditions and surveillance requirements in Sections 3/4 2.2, APRM Setpoints, and 3/4 3.1, Reactor Protection System Instrumentation.

Required operation of engineered safeguard features (ESF) is not expected for either of these transients.

15.1.1.2.3 Effect of Single Failures and Operator Errors

These two events generally lead to an increase in reactor power level. The simulated thermal power trip (Section 15.1.1.2.2) is the mitigating system and is designed to be single-failure proof. Therefore, single failures are not expected to result in a more severe event than analyzed (see Appendix 15A for a detailed discussion of this subject).

15.1.1.3 Core and System Performance

15.1.1.3.1 Mathematical Model

The predicted dynamic behavior has been determined using a computer-simulated, analytical model of a generic direct-cycle BWR. This model is described in detail in NEDO-10802[1]. This computer model has been verified through extensive comparison of its predicted results with actual BWR test data.

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

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

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

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

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

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

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

15.1.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-3. The plant is assumed to be operating at 105 percent of nuclear boiler rated (NBR) steam flow and at thermally-limited conditions. Both automatic and manual modes of flow control are considered.

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

15.1.1.3.3 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. 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 simulated 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.

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 RCPB is not threatened. If scram occurs, the results become very similar to the manual flow control case. This transient is illustrated on Figure 15.1-1.

In manual mode, no compensation is provided by core flow, and thus the power increase simulated is greater than in the automatic mode. A scram on high APRM may occur. Vessel steam flow increases and the initial system pressure increase is slightly larger. Peak heat flux is 117 percent of its initial value. The increased core inlet subcooling aids core thermal margins, and ΔCPR for this event is 0.15.

After the reactor scram, water level rises briefly to the narrow range, high-level trip setpoint (L8). At this time both the main turbine and feedwater pumps are tripped and RPT is initiated. The transient responses of the key plant variables for this mode of operation are shown on Figure 15.1-2. This transient is less severe from lower initial power levels because lower initial power levels will have initial values of CPR greater than the limiting initial CPR value assumed.

15.1.1.3.4 Considerations of Uncertainties

Important factors (such as reactivity coefficient, scram characteristics, and 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 previously and shown on Figures 15.1-1 and 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 event is evaluated for each reload as a potentially limiting event and was one of the limiting cases evaluated for operation at uprated (3,467 MWt) power conditions.[3] This event was also analyzed for EPU (3,988 MWt) for overpower results in accordance with Reference 4. Results of the cycle-specific analysis at uprated power conditions are provided in Appendix A.

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, which shows the changes in important variables during this transient.

The Operator should verify automatic functions and monitor all parameters.

No Operator action is required.

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 RPSs. Important system operational actions for this event are the tripping of the main turbine and feedwater pumps, RPT, and low water level initiation of the RCIC system and the HPCS system to maintain long-term water level control following tripping of feedwater pumps.

15.1.2.2.3 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) trip. Multiple level sensors are used to sense and detect when the water level reaches the L8 setpoint. At this point in the logic, a single failure will not initiate or prevent a turbine trip signal. Turbine trip signal transmission, however, is not built to single-failure criterion. The result of a failure at this point would have the effect of delaying the pressurization "signature." However, high moisture levels entering the turbine will be detected by high levels in the turbine's moisture separators, resulting in a trip of the unit.

Scram signals from the turbine stop valve and control valve closures are designed so that a single failure will neither initiate nor impede a reactor scram trip initiation (see Appendix 15A for a detailed discussion of this subject).

15.1.2.3 Core and System Performance

15.1.2.3.1 Mathematical Model

The predicted dynamic behavior has been determined using a computer-simulated, analytical model of a generic direct-cycle BWR. This model is described in detail in NEDO-24154. This computer model has been improved and verified through extensive comparison of its predicted results with actual BWR test data.

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

1. An integrated one-dimensional core model, which includes a detailed description of hydraulic feedback

effects, axial power shape changes, and reactivity feedbacks, is assumed.

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

3. The steam lines are modeled by eight pressure nodes incorporating mass and momentum balances that predict any wave phenomenon present in the steam line during pressurization transient.

4. 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 bypass flow. A model, representing liquid and vapor mass and energy conservation and mixture momentum conservation, is used to describe the thermal-hydraulic behavior. Changes in the flow split between the bypasses and active channel flow are accounted for during transient events.

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

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

7. The control systems and RPS models are, for the most part, identical to those employed in the point reactor model[2], 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 15.0-3. The analysis performed for EPU (120%) was performed with the plant conditions tabulated in Table 15.0-3a.

EOC-1 nuclear characteristics are assumed. The SRV action is conservatively assumed to occur with higher than nominal setpoints. The transient is simulated by programming an upper limit failure in the feedwater system, such that 145 percent NBR feedwater flow occurs at a system design pressure of 1,060 psig, and 155 percent NBR feedwater flow occurs at the reactor system pressure of 1,010 psig. The assumed feedwater runout capacity function is extremely conservative and its validity has been confirmed in startup test N2-SUT-23, which yielded an adjusted runout flow of 132.5 percent @ 1,010 psig.

15.1.2.3.3 Results

The simulated feedwater controller transient is shown on Figure 15.1-3. The high water level turbine trip and feedwater pump trip are initiated at approximately 12 sec. Scram occurs simultaneously and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs.

The change in CPR is 0.16, and peak surface heat flux is 114.3 percent of its initial valve.

The turbine bypass system and the SRVs open to limit the peak vessel bottom pressure to 1,203 psig, and the nuclear system process barrier pressure limit is not endangered. The bypass valves subsequently close to reestablish pressure control in the vessel during shutdown. Events caused by low water level trips, including closure of MSIVs and initiation of HPCS and RCIC system junctions, are not included in the simulation. Should these events occur, they will follow some time after the primary concerns of just thermal margin and overpressure effects have occurred. These events are less severe than those already experienced by the system.

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). Plant behavior is, therefore, expected to lead to a less severe transient.

15.1.2.4 Barrier Performance

As noted previously, 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 consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to 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 the ODCM; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.1.3 Pressure Regulator Failure - Open

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

The transient is non-limiting; therefore, it was not evaluated for EPU (3,988 MWt).

15.1.3.1 Identification of Causes and Frequency Classification

15.1.3.1.1 Identification of Causes

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

If either the controlling pressure regulator or the backup regulator fails to the open position, the turbine admission valves can be fully opened and the turbine bypass valves can be partially opened until the maximum steam flow is established.

15.1.3.1.2 Frequency Classification

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

15.1.3.2 Sequence of Events and Systems Operation

15.1.3.2.1 Sequence of Events

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

The Operator should verify automatic functions and monitor all parameters.

No Operator action is required.

15.1.3.2.2 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 RPS, except as otherwise noted.

Initiation of HPCS and RCIC system functions occurs when the vessel water level reaches the L2 setpoint. Normal startup and actuation can take up to 30 sec before effects are realized. If these events occur, they 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 Effect of Single Failures and Operator Errors

This transient leads to a loss of pressure control such that the increased steam flow causes a depressurization. Instrumentation for pressure sensing of the turbine inlet pressure is designed to be single-failure proof for initiation of MSIV closure.

Reactor scram originating from limit switches on the MSIV is designed to be single-failure proof. It is therefore concluded that the basic phenomenon of pressure decay is adequately terminated (see Appendix 15A for a detailed discussion of this subject).

15.1.3.3 Core and System Performance

15.1.3.3.1 Mathematical Model

The nonlinear dynamic model described briefly in Section 15.1.1.3.1 is used to simulate this event[1].

15.1.3.3.2 Input Parameters and Initial Conditions

This transient is simulated by setting the controlling regulator output to a high value, which causes the turbine admission valves and the turbine bypass valves to open fully. Since the controlling and backup regulator outputs are gated by a high value gate, the effect of such a failure in the backup regulator would be exactly the same. A regulator failure with 130 percent steam flow demand was simulated as a worst case, since 115 percent is the normal maximum flow limit.

A 5-sec isolation valve closure, instead of a 3-sec closure, is assumed when the turbine pressure decreases below the turbine inlet low-pressure setpoint for main steam isolation initiation. This is within the specification limits of the valve and represents a conservative assumption.

Reactor scram is initiated when the turbine stop valves reach the 10-percent closed position. Stop valve closure is initiated by a turbine trip when the reactor water level reaches the level setpoint (L8).

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

15.1.3.3.3 Results

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

Depressurization results in formation of voids in the reactor coolant and causes a decrease in reactor power almost immediately. The depressurization rate is large enough that

water level swells to the sensed level trip setpoint (L8), initiating main turbine and feedwater pump motors trips, and shuts down the reactor. After the turbine trip, the failed pressure regulator signals the bypass to open to full bypass flow of approximately 18.5 percent turbine rated steam flow. After the pressurization resulting from the turbine stop valve closure, pressure again drops and continues to drop until the turbine inlet pressure is below the low turbine pressure isolation setpoint, where main steam line isolation limits the duration and severity of the depressurization so that no significant thermal stresses are imposed on the RCPB. A short-duration neutron flux increase from about 74 percent to a maximum of 85 percent of NBR results after the turbine trip. No significant reductions of fuel thermal margins occur.

15.1.3.3.4 Considerations of Uncertainties

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

For example, the turbine control valves open to the valves-wide-open (VWO) state admitting slightly more than the rated steam flow. With the limiter in this analysis set to fail at 130 percent, it is expected that something less than 20 percent will be bypassed. Therefore, this is not a limiting factor on this plant. If the rate of depressurization does change, it will be terminated by the low turbine inlet pressure trip setpoint.

The depressurization rate has a proportional effect upon the voiding action in the core and the flashing in the vessel bulk water regions. If the rate is low enough, the water level may not swell to the high water level trip setpoint, and the isolation will occur earlier when pressure at the turbine decreases below 766 psig. The reactor will scram as a result of the MSIV closure.

Operator action during this transient may include placing the reactor mode switch in the shutdown position as required by the scram procedure or emergency operating procedures (EOPs). Depending upon when this action is taken it may defeat the MSIV low-pressure isolation. Should this be the case, both the scram procedure and EOPs provide direction to control RPV cooldown to less than 100°F/hr. Actions to complete this requirement may include closing the MSIVs manually. Thus for this scenario the severe depressurization is manually controlled as directed by procedures.

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 1,141 psig, which is below the ASME Code limit of 1,375 psig for the RCPB. Vessel dome pressure reaches 1,117 psig. Minimum vessel dome pressure of 808 psig occurs at about 48 sec.

15.1.3.5 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 (Section 15.1.2.5). Therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.1.4 Inadvertent Safety/Relief Valve Opening

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

The transient is non-limiting; therefore, it was not evaluated for EPU (3,988 MWt).

15.1.4.1 Identification of Causes and Frequency Classification

15.1.4.1.1 Identification of Causes

The 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.

15.1.4.1.2 Frequency Classification

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

15.1.4.2 Sequence of Events and Systems Operation

15.1.4.2.1 Sequence of Events

The sequence of events for this event is listed in Table 15.1-5.

The plant Operator must reclose the valve as soon as possible and check that reactor and turbine generator output return to normal. If the valve cannot be closed, plant shutdown is initiated.

15.1.4.2.2 Systems Operation

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

15.1.4.2.3 Effect of Single Failures and Operator Errors

Single failure of additional components* (e.g., pressure regulator, feedwater flow controller) is also discussed in this chapter. In addition, a detailed discussion of such effects is given in Appendix 15A.

15.1.4.3 Core and System Performance

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(1). It was determined that this event is not limiting from a standpoint of core performance. Therefore, a qualitative presentation of results is described as follows.

15.1.4.3.2 Input Parameters and Initial Conditions

It is assumed that the reactor is operating at an initial power level corresponding to 105 percent of rated steam flow conditions when a SRV is inadvertently opened. Manual recirculation flow control is assumed. Flow through the valve at the normal plant operating conditions previously stated above is approximately 7 percent NBR steam flow.

15.1.4.3.3 Qualitative Results

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

The pressure regulator senses the nuclear system pressure decrease and within a few seconds closes the turbine control valve (TCV) 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

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* Leading to a more limiting transient.

transient. MCPR is essentially unchanged and, therefore, the safety limit margin is unaffected.

15.1.4.4 Barrier Performance

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

15.1.4.5 Radiological Consequences

While 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 (Section 15.1.2.5).

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

This event is not applicable to BWR plants.

15.1.6 Inadvertent RHR Shutdown Cooling Operation

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

The transient is non-limiting; therefore, it was not evaluated for EPU (3,988 MWt).

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 occurs before any thermal limits are reached if the Operator does not take action. The sequence of events for this event is shown in Table 15.1-6.

15.1.6.2.2 System Operation

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

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 is controlled in the normal manner. If no Operator action is taken, a scram terminates 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 previously, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are

designed; therefore, these barriers maintain their integrity and function as designed.

15.1.6.5 Radiological Consequences

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

15.1.7 References

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

2. Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154-A, Volumes 1-3, February 1986; NEDC-24154P-A Supplement 1, Volume 4, February 2000.

3. Licensing Topical Report, Power Uprate Licensing Evaluation for Nine Mile Point Nuclear Power Station, Unit 2, NEDC-31994P, Revision 1, May 1993.

4. GE Nuclear Energy, Constant Pressure Power Uprate, NEDC-33004P-A, Revision 4, Class III (Proprietary), July 2003; and NEDO-33004, Class I (Non-proprietary), July 2003.

TABLE 15.1-1

SEQUENCE OF EVENTS FOR FIGURE 15.1-1

Loss of Feedwater Heating - Automatic Recirculation Flow Control

Note: These results are for Cycle 1. This event is less severe than the case with the reactor in manual control, and is not reanalyzed for power uprate or for each reload cycle.

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 the automatic flow control system automatically reduces core flow to maintain initial steam flow.
200+	Reactor variables settle into new steady state.

TABLE 15.1-2

SEQUENCE OF EVENTS FOR FIGURE 15.1-2

Loss of Feedwater Heating - Manual Recirculation Flow Control

Note: These results are for Cycle 1. Cycle-specific results are presented in Appendix A.

Time (sec)	Event
0	Initiate a 100°F temperature reduction into the feedwater system.
~10	Initial effect of unheated feedwater starts to raise core power level and steam flow.
54	APRM initiates reactor scram on high thermal power.
~57	TCVs start to close to regulate pressure.
60	Narrow range water level exceeds L8 high water level setpoint, resulting in trip of the main turbine and feedwater pumps. Turbine bypass operation initiated.
60.01	RPT initiated when turbine stop valves reach the 10% closed position.
60.1	Turbine stop valves closed. Bypass valves do not open as turbine pressure falls below the pressure regulator setpoints.
60.14	Recirculation pump motor circuit breakers open, causing decrease in core flow to natural circulation.
62	Turbine inlet pressure exceeds pressure regulator setpoints, and bypass valves open to regain pressure control.
100+	Reactor variables settle into limit cycle.

TABLE 15.1-3

SEQUENCE OF EVENTS FOR FIGURE 15.1-3

Feedwater Controller Failure - Demand

Note: These results are for Cycle 1. Cycle-specific results are presented in Appendix A.

Time (sec)	Event
0	Initiate simulated failure of 145% upper limit on feedwater flow.
8.4	Turbine bypass valves start to open.
11.6	L8 vessel level setpoint trips main turbine and feedwater pumps.
11.6	Reactor scram trip actuated from main turbine stop valve position switches.
11.6	RPT actuated by stop valve position switches.
11.7	Main turbine stop valves closed, and turbine bypass valves start to open.
11.8	Recirculation pump motor circuit breakers open, causing decrease in core flow to natural circulation.
12.9	Relief valves actuated due to high pressure.
18.6	All relief groups closed.

TABLE 15.1-4

SEQUENCE OF EVENTS FOR FIGURE 15.1-4

Pressure Regulator Failure - Open

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	Simulate minimum limit on steam flow to main turbine.
0.19 (est)	TCVs open wide.
0.19	Main turbine bypass partially open.
3.95	Vessel water level (L8) trip initiates main turbine and feedwater pump motor trips.
3.95	Turbine trip indicates bypass operation to full flow.
3.96	Main turbine stop valves reach 10% closed position and initiate reactor scram trip and RPT.
4.05	Turbine stop valves closed. Turbine bypass valves opening to full flow.
4.09	Recirculation pump motor circuit breakers open, causing decrease in core flow to natural circulation.
6.8	Pressure relief valves actuated.
14 (est)	Relief valves closed.
46	Main steam line isolation on low turbine inlet pressure ($<$766 psig).
47	RCIC and HPCS systems initiation on low level (L2) (not simulated).
51	MSIVs closed. Bypass valves remain open, exhausting steam in steam lines downstream of isolation valves.

TABLE 15.1-5

SEQUENCE OF EVENTS FOR INADVERTENT
SAFETY/RELIEF VALVE OPENING

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	Initiate opening of one SRV.
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

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

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

15.2 INCREASE IN REACTOR PRESSURE

Cycle-specific reload analyses are discussed in Appendix A, Section A.15.2.

15.2.1 Pressure Regulator Failure - Closed

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

This transient is non-limiting; therefore, it was not evaluated for EPU (3,988 MWt).

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. The output of both regulators feeds into a high value gate. The regulator with the highest output controls the main TCVs. The lowest pressure setpoint gives the largest pressure error and thereby the largest regulator output. The backup regulator is set approximately 5 psi higher giving a slightly smaller error and a slightly smaller effective output of the controller.

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

15.2.1.1.2 Frequency Classification

This event is treated as a moderate frequency event.

15.2.1.2 Sequence of Events and System Operation

15.2.1.2.1 Sequence of Events

Postulating a failure of the primary or controlling pressure regulator in the closed mode (Section 15.2.1.1.1) causes the TCVs start to close. The pressure increases because the reactor is still generating the initial steam flow. The backup regulator reopens the valves and reestablishes steady-state operation above the initial pressure equal to the setpoint difference of approximately 5 psi.

No Operator action is required.

15.2.1.2.2 Systems Operation

One Pressure Regulator Failure - Closed

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

15.2.1.2.3 The Effect of Single Failures and Operator Errors

The nature of the first assumed failure produces a slight pressure increase in the reactor until the backup regulator gains control. No other action is significant in restoring normal operation. If the backup regulator fails at this time, the second assumed failure, the control valves start to close, raising reactor pressure to the point where a flux scram trip is initiated to shut down the reactor. At 100-percent power, this event is less severe than the turbine trip where stop valve closure occurs (Event 15.2.3) and shuts off steam flow at a faster rate. At less than 100-percent power, an evaluation has been performed for the reload cycles as specified in the Core Operating Limits Report source documents.

15.2.1.3 Core and System Performance

The disturbance is mild, similar to a pressure setpoint change, and no significant reductions in fuel thermal margins occur. This transient is much less severe than the generator and turbine trip transients described in Sections 15.2.2 and 15.2.3.

15.2.1.3.1 Mathematical Model

Qualitative evaluation only is provided.

15.2.1.3.2 Input Parameters and Initial Conditions

Qualitative evaluation only is provided.

15.2.1.3.3 Results

The response of the reactor during this regulator failure is such that pressure at the turbine inlet increases in less than 2 sec, due to the sharp closing action of the TCVs 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.

15.2.1.3.4 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 control rod worth characteristics). Plant behavior is therefore expected to reduce the actual severity of the transient.

15.2.1.4 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.

15.2.1.5 Radiological Consequences

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

15.2.2 Generator Load Rejection

This transient event is evaluated for each reload as a potentially limiting event and was one of the limiting cases evaluated for operation at uprated (3,467 MWt) power conditions[3] and evaluated for overpower results for EPU (3,988 MWt) in accordance with Reference 8. Results of the cycle-specific analysis at uprated power conditions are provided in Appendix A.

15.2.2.1 Identification of Causes and Frequency Classification

15.2.2.1.1 Identification of Causes

Fast closure of the TCVs is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The TCVs are required to close as rapidly as possible to prevent excessive overspeed of the turbine generator. Fast closure of the main TCVs causes a reactor scram and sudden reduction in steam flow which results in an increase in system pressure.

15.2.2.1.2 Frequency Classification

<u>Generator Load Rejection with Bypass</u>

This event is categorized as an incident of moderate frequency.

<u>Generator Load Rejection with Bypass Failure</u>

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

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

<u>Frequency Basis</u> 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, or a MTBE of 278 yr. Although this event is classified as infrequent, it has been considered in the evaluation of operating CPR limits in Table 15.0-2 and on Figure 15.0-1.

15.2.2.2 Sequence of Events and System Operation

15.2.2.2.1 Sequence of Events

Generator Load Rejection with Bypass

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

Generator Load Rejection with Bypass Failure

A loss of generator electrical load at high power with bypass failure produces the sequence of events listed in Table 15.2-2.

Identification of Operator Actions

The Operator should verify automatic functions and monitor all parameters.

No Operator actions are required.

15.2.2.2.2 System Operation

Generator Load Rejection with Bypass

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

TCV fast closure initiates a scram trip signal for power levels greater than 30 percent NBR. For analyses performed at EPU and thereafter, the TCV fast closure initiates a scram trip signal at 26 percent NBR. In addition, RPT is initiated. Both these trip signals satisfy single-failure criterion and credit is taken for these protection features.

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

Generator Load Rejection with Bypass Failure

Same as generator load rejection with bypass except that failure of the main turbine bypass valves is assumed for the entire transient.

15.2.2.2.3 The Effect of Single Failures and Operator Errors

Mitigation of pressure increase, the basic nature of this transient, is accomplished by the RPS functions. TCV trip scram and RPT are designed to satisfy the single-failure criterion. An evaluation of the most limiting single failure (i.e., failure of the bypass system) was considered in this event. Details of single-failure analysis can be found in 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 15.0-3. The EPU analysis was performed with plant conditions listed in Table 15.0-3a.

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

The closure characteristics of the TCVs are modeled as they actually operate, in partial arc mode (beginning in Cycle 14). This operating mode maintains three of the valves at full open position while the fourth valve is throttled at 47% $\pm$ 7%. The valves have a full stroke closure time, from fully open to fully closed, of 0.15 sec.

Auxiliary power is normally independent of any turbine generator overspeed effects and is continuously supplied at rated frequency, assuming automatic fast transfer to offsite power supplies. For the purpose of worst-case analysis, the recirculation pumps are assumed to remain powered from the main generator and thus increase in speed with the turbine generator overspeed until tripped by the RPT system.

The reactor is operating in the manual flow control mode when load rejection occurs. Results do not significantly differ if the plant is 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 closure of MSIVs and initiation of HPCS and RCIC system functions, are not included in the simulation. If these events occur, they will follow sometime after the primary concerns of fuel margin and overpressure effects have passed and are expected to result in effects less severe than those already experienced by the reactor system.

15.2.2.3.3 Results

Generator Load Rejection with Bypass

Figure 15.2-1 shows the results of the generator trip from 104.3 percent rated power. Peak neutron flux rises to 265 percent NBR conditions.

The average surface heat flux peaks at 107.3 percent of its initial value, and a ΔCPR in this transient is 0.10.

Generator Load Rejection with Bypass Failure

Figure 15.2-2 shows that, for the case of bypass failure, peak neutron flux reaches about 407 percent of rated and average surface heat flux reaches 113.2 percent of its initial value. The ΔCPR for this event is 0.16.

15.2.2.3.4 Consideration of Uncertainties

The full stroke closure time of the TCV of 0.15 sec is conservative. Typically, the actual closure time is closer to 0.20 sec; the less time it takes to close, the more severe the pressurization effect.

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). Anticipated plant behavior is, therefore, expected to reduce the actual severity of the transient.

15.2.2.4 Barrier Performance

Generator Load Rejection with Bypass

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

Generator Load Rejection with Bypass Failure

Peak pressure at the SRVs reaches 1,203 psig. The peak nuclear system pressure reaches 1,233 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1,375 psig.

15.2.2.5 Radiological Consequences

While the consequence of this event does not result in fuel failures, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation (Section 15.1.2.5).

15.2.3 Turbine Trip

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power). Evaluation of this event for operation at uprated (3,467 MWt) conditions confirmed that turbine trip events are similar to but bounded by the generator load rejection event.[3] The turbine trip event does not set reactor operating limits and is not reanalyzed for each reload cycle. For reload cores, an evaluation is performed to determine if the turbine trip with bypass failure could potentially alter the previous cycle MCPR operating limit. If it does, the results will be reported in the supplemental reload licensing report. The turbine trip with bypass failure was analyzed for EPU (3,988 MWt).

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 initiate a turbine trip. Some examples are moisture separator high level, turbine vibrations, Operator lockout, loss of EHC fluid pressure, low condenser vacuum, and reactor high water level.

15.2.3.1.2 Frequency Classification

<u>Turbine Trip</u>

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

<u>Turbine Trip with Bypass Failure</u>

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

Frequency	0.0064/plant year
MTBE	156 yr

<u>Frequency Basis</u> As discussed in Section 15.2.2.1.2, 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, or a MTBE of 156 yr. Although this event is classified as infrequent, it has been considered in the evaluation of operating CPR limits in Table 15.0-2 and on Figure 15.0-1.

15.2.3.2 Sequence of Events and Systems Operation

15.2.3.2.1 Sequence of Events

<u>Turbine Trip</u>

Turbine trip at high power produces the sequence of events listed in Table 15.2-3. The analysis assumes the availability of feedwater pump motors after successful fast transfer. Thereafter, the feedwater pump motor trips at Level 8. Even if the feedwater pump motor were to trip immediately after turbine trip, the analysis is still bounding. The timing of scram and RPT would remain unchanged and the subsequent HPCS and RCIC operation due to loss of feedwater would have no effect on all key transient parameters.

<u>Turbine Trip with Bypass Failure</u>

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

<u>Identification of Operator Actions</u>

The Operator should verify automatic functions and monitor all parameters.

No Operator action is required.

15.2.3.2.2 Systems Operation

<u>Turbine Trip</u>

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 RPS.

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

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

It should be noted that below approximately 30 percent NBR power level, 26 percent NBR for EPU and subsequent analyses, a main stop valve scram trip inhibit signal derived from the first-stage pressure of the turbine is activated. This is done to eliminate the stop valve scram trip signal from scramming the reactor when steam flow is within bypass valve capacity. All other protection system functions remain operational, and credit is taken for those protection system trips.

Turbine Trip with Bypass Failure

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

15.2.3.2.3 The Effect of Single Failures and Operator Errors

Turbine Trips at Power Levels Greater Than 26 Percent NBR

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

Turbine Trips at Power Levels Less Than 26 Percent NBR

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

15.2.3.3 Core and System Performance

15.2.3.3.1 Mathematical Model

The computer model 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 15.0-3.

The turbine stop valve full stroke closure time is 0.1 sec.

A reactor scram is initiated by position switches on the stop valves when the valves are more than 10 percent closed. This stop valve scram trip signal is automatically bypassed when the reactor is below approximately 26 percent NBR power level.

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

15.2.3.3.3 Results

Turbine Trip

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

Neutron flux increases because of the void reduction caused by the pressure increase. However, the increase is limited to 213 percent of rated flux level by the stop valve scram and the RPT system. Peak fuel surface heat flux does not exceed 104.5 percent of its initial value. The change in CPR during this transient is 0.07.

Turbine Trip with Bypass Failure

A turbine trip with failure of the bypass system is simulated at 105 percent NBR steam flow conditions as shown on Figure 15.2-4.

Peak neutron flux reaches 331 percent of its rated value, and peak surface heat flux increases to 111.1 percent of the initial value. The ΔCPR is 0.14.

Turbine Trip with Bypass Failure, Low Power

This transient is less severe than a similar one at high power. Below 26 percent of rated power, the turbine stop valve closure and TCV closure scrams and RPT are automatically bypassed. At these lower power levels, turbine first-stage pressure is used to initiate the scram logic bypass. The scram which terminates the transient is initiated by high neutron flux or high vessel pressure. The bypass valves are assumed to fail; therefore, system pressure increases until the pressure relief setpoints are reached. At this time, because of the relatively low power of this transient event, relatively few relief valves open to limit reactor pressure. Peak pressures are not expected to greatly exceed the lowest pressure relief valve setpoints and are significantly below the RCPB transient limit of 1,375 psig. The peak surface heat flux and peak fuel center temperature remain at relatively low values and the MCPR remains well above the GETAB safety limit.

15.2.3.3.4 Considerations of Uncertainties

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

1. Slowest allowable control rod scram motion is assumed.

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

3. Setpoints of the SRVs are 1 to 2 percent higher than the valves' nominal setpoints.

15.2.3.4 Barrier Performance

Turbine Trip

Peak pressure in the bottom of the vessel reaches 1,194 psig, which is below the ASME Code limit of 1,375 psig for the RCPB. Vessel dome pressure does not exceed 1,167 psig. The severity of turbine trips from lower initial power levels decreases to the point where a scram can be avoided if offsite power is available, and the power level is within the bypass capability.

Turbine Trip with Bypass Failure

The SRVs 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 1,232 psig at the vessel bottom; therefore, the overpressure transient is clearly below the RCPB transient pressure limit of 1,375 psig. Peak dome pressure does not exceed 1,203 psig.

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

A qualitative discussion of this event at low power is provided in Section 15.2.3.3.3, Turbine Trip with Bypass Failure, Low Power.

15.2.3.5 Radiological Consequences

While this event does not result in fuel failures, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation (Section 15.1.2.5).

15.2.4 Main Steam Isolation Valve Closures

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle. This event was not reanalyzed for EPU (3,988 MWt).

15.2.4.1 Identification of Causes and Frequency Classification

15.2.4.1.1 Identification of Causes

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

15.2.4.1.2 Frequency Classification

<u>Closure of All Main Steam Isolation Valves</u>

This event is categorized as an incident of moderate frequency. To define the frequency of this event as an initiating event and not the by-product of another transient, only the following contribute to the frequency: manual action (purposely or inadvertent); spurious signals such as low pressure, low reactor water level, low condenser vacuum; and equipment malfunctions such as faulty valves or operating mechanisms. A closure of one MSIV may cause an immediate closure of all the other MSIVs, depending on reactor conditions. If this occurs, it is also included in this category. During MSIV closure, position switches on the valves provide a reactor scram if the valves in three or more main steam lines are less than 85 percent open (except for bypasses which permit proper plant startup). Protection system logic, however, permits the test closure of one valve without initiating scram from the position switches.

<u>Closure of One Main Steam Isolation Valve</u>

This event is categorized as an incident of moderate frequency. The BWR is designed so that one MSIV may be manually closed at a time for testing purposes. Operator error or equipment malfunction may cause a single MSIV to be closed inadvertently. If reactor power is greater than about 75 percent when this occurs, a high flux scram or high steam line flow isolation may result. (If all MSIVs close as a result of the single closure, the event is considered a closure of all MSIVs.)

15.2.4.2 Sequence of Events and Systems Operation

15.2.4.2.1 Sequence of Events

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

<u>Identification of Operator Actions</u>

The Operator should verify automatic functions and monitor all parameters.

No Operator action is required.

15.2.4.2.2 Systems Operation

<u>Closure of All Main Steam Isolation Valves</u>

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

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

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

Closure of One Main Steam Isolation Valve

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

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

15.2.4.2.3 Effect of Single Failures and Operator Errors

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

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

15.2.4.3 Core and System Performance

15.2.4.3.1 Mathematical Model

The computer model described in 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 15.0-2.

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

Position switches on the valves initiate a reactor scram when the valves are less than 85 percent open.

15.2.4.3.3 Results

Closure of All Main Steam Isolation Valves

Figure 15.2-5 shows the changes in important nuclear system variables for the simultaneous isolation of all main steam lines while the reactor is operating at 105 percent of NBR steam flow. Peak neutron flux reaches 138 percent of rated after approximately 2.4 sec. At this time, the nonlinear valve closure becomes a strong effect and the conservative scram characteristic assumption has not yet allowed credit for the full shutdown of the reactor.

Closure of One Main Steam Isolation Valve

With a 3-sec closure of one MSIV during 105 percent rated steam flow conditions, the steam flow disturbance raises vessel pressure and reactor power enough to initiate a high neutron flux scram. This transient is considerably milder than closure of all MSIVs at full power. No quantitative analysis is furnished for this event; however, no significant change in thermal margins is experienced and no fuel damage occurs. Peak pressure remains below SRV pressure relief 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) produces no significant transient. Closures during plant heatup (operating state D) are less severe than the maximum power cases (maximum stored and decay heat) discussed in Section 15.2.4.3.3, Closure of All Main Steam Isolation Valves.

15.2.4.3.4 Considerations of Uncertainties

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

1. Slowest allowable control rod scram motion is assumed.

2. Nuclear characteristics for all-rods-out EOC conditions are assumed.

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

4. Pressure relief setpoints of the SRVs are assumed to be 1 to 2 percent higher than the valves' nominal setpoints.

15.2.4.4 Barrier Performance

Closure of All Main Steam Line Isolation Valves

The nuclear system relief valves begin to open at approximately 3.1 sec after the start of isolation. The relief valves close sequentially as the stored heat is dissipated but continue to discharge the decay heat intermittently. Peak pressure at the vessel bottom reaches 1,215 psig, which is below the pressure limits of the RCPB. Peak pressure in the main steam line is 1,177 psig.

Closure of One Main Steam Isolation Valve

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

15.2.4.5 Radiological Consequences

While this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation (Section 15.1.2.5).

15.2.5 Loss of Condenser Vacuum

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle. This event was not reanalyzed for EPU (3,988 MWt).

15.2.5.1 Identification of Causes and Frequency Classification

15.2.5.1.1 Identification of Causes

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

15.2.5.1.2 Frequency Classification

This event is categorized as an incident of moderate frequency.

15.2.5.2 Sequence of Events and Systems Operation

15.2.5.2.1 Sequence of Events

Table 15.2-7 lists the sequence of events for Figure 15.2-6.

For the analyzed event (vacuum loss at 2 in Hg/sec), no Operator action is assumed or required. The Operator should verify all automatic functions and monitor all parameters.

15.2.5.2.2 Systems Operation

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

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

15.2.5.2.3 Effect of Single Failures and Operator Errors

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

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

Single failures do not effect the vacuum monitoring and turbine trip devices which are redundant. The protective sequences of the anticipated operational transient are shown to be single-failure proof (see Appendix 15A for details).

15.2.5.3 Core and System Performance

15.2.5.3.1 Mathematical Model

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

15.2.5.3.2 Input Parameters and Initial Conditions

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

The turbine stop valve full stroke closure time is 0.1 sec.

A reactor scram is initiated by position switches on the stop valves when the valves are 10 percent closed. This stop valve scram trip signal is automatically bypassed when the reactor is below 26 percent NBR power level.

The analysis presented here is a hypothetical case with a conservative 2 in Hg/sec vacuum decay rate. The bypass system is available for several seconds since the bypass system is signaled to close at a vacuum level of approximately 15 in Hg abs.

15.2.5.3.3 Results

Under this hypothetical 2 in Hg/sec vacuum decay condition, the turbine bypass valve and MSIV closure would follow main turbine trip about 5 sec after they initiate the transient. This transient, therefore, is similar to a normal turbine trip with bypass. The effect of MSIV closure tends to be minimal since the closure of main turbine stop valves and subsequently the bypass valves have already shut off the main steam line flow. Figure 15.2-6 shows the transient expected for this event. It is assumed that the plant is initially operating at 105 percent of NBR steam flow conditions. Peak neutron flux reaches 215 percent of NBR power while average fuel surface heat flux reaches 104.7 percent of initial value. SRVs open to limit the pressure rise, then sequentially reclose as the stored energy is dissipated.

15.2.5.3.4 Considerations of Uncertainties

The reduction or loss of vacuum in the main turbine condenser sequentially trips the main turbine and closes the MSIVs and bypass valves. While these are the major events occurring, other resultant actions 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 SJAE problem produces a very slow rate of loss of vacuum (Table 15.2-6). If corrective actions by the Reactor Operators (ROs) are not successful, then trip of the main turbine, and eventual closure of the bypass valves (opened with the main turbine trip) and the MSIVs, occurs.

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:

1. Slowest allowable control rod scram motion is assumed.

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

3. Setpoints of the SRVs are assumed to be 1 to 2 percent higher than the valves' nominal setpoints.

15.2.5.4 Barrier Performance

Peak nuclear system pressure is 1,194 psig at the vessel bottom. The overpressure transient is below the RCPB transient pressure limit of 1,375 psig. Vessel dome pressure does not exceed 1,167 psig. A comparison of these values to those for turbine trip at high power shows the similarities between these two transients.

15.2.5.5 Radiological Consequences

While this event does not result in fuel failures, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation (Section 15.1.2.5).

15.2.6 Loss of Ac Power

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle. This event was not reanalyzed for EPU (3,988 MWt).

15.2.6.1 Identification of Causes and Frequency Classification

15.2.6.1.1 Identification of Causes

<u>Loss of Normal Station Service and Reserve Transformers</u>

Causes for interruption or loss of the normal Station service and reserve transformers can arise from external causes or malfunctioning of transformer protection circuitry. These can include high transformer oil temperature, reverse or high-current operation, and Operator error which trips the transformer breakers.

<u>Loss of All Grid Connections</u>

Loss of all grid connections can result from major shifts in electrical loads, loss of loads, lightning, storms, wind, etc., which contributed to electrical grid instabilities. These instabilities may 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

<u>Loss of Normal and Preferred Station Service Transformers</u>

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

<u>Loss of All Grid Connections</u>

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

Loss of Normal and Preferred Station Service Transformers

Table 15.2-9 lists the sequence of events for Figure 15.2-7.

Loss of All Grid Connections

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

Identification of Operator Actions

For either postulated event, no Operator actions are required. The Operator should verify automatic functions and monitor all parameters.

15.2.6.2.2 Systems Operation

Loss of Normal and Preferred Station Service Transformers

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

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

1. Recirculation pumps and feedwater pump motors are tripped at a reference time, t=0, with normal coastdown times.

2. At 2 sec, after the initial event, a reactor scram takes place due to loss of auxiliary electric power.

3. At 2 sec, after the initial event, the analysis assumes MSIV closure, whereas actual closure will take significantly longer due to low condenser vacuum or low main steam line turbine inlet pressure. The 2-sec MSIV closure time is a conservative assumption for the analysis, since closing MSIVs earlier will result in higher reactor pressure and a measured increase in the neutron flux.

4. Within 8 sec, the loss of main condenser circulating water pumps causes condenser vacuum to decrease to the turbine trip setpoint. However, MSIVs are assumed already closed and the feedwater system tripped.

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

<u>Loss of All Grid Connections</u>

Same as loss of normal and preferred Station service transformers, 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 foregoing sequence at reference time t=0. The load rejection immediately forces the TCVs closed, causes a scram, and initiates RPT (already tripped at reference time t=0).

15.2.6.2.3 Effect of Single Failures and Operator Errors

Loss of the normal and preferred Station service transformers 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. Additional failures of the other systems assumed to protect the reactor would not result in an effect different from those reported. Failures of the protection systems have been considered and satisfy single-failure criteria; therefore, 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 the "loss of auxiliary power transformers" event, and the model described in Section 15.1.2.3.1 was used to simulate the "loss of all grid connections" event. Operation of the RCIC or HPCS systems is not included in the simulation of this transient, since startup of these pumps does not permit flow in the time period of this simulation.

15.2.6.3.2 Input Parameters and Initial Conditions

<u>Loss of Normal and Preferred Station Service Transformers</u>

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-3 and under the assumed systems constraints described in Section 15.2.6.2.2.

<u>Loss of All Grid Connections</u>

Same as loss of normal and preferred Station service transformers.

15.2.6.3.3 Results

Loss of Normal and Preferred Station Service Transformers

Figure 15.2-7 shows the simulated transient. The initial portion of the transient is similar to the RPT transient. Scram and MSIV closure occur at T=2 sec due to loss of electric source power.

Sensed level drops to the RCIC and HPCS initiation setpoint at approximately 23 sec after loss of auxiliary power. Decay heat is dissipated by SRV actuation plus RHR system operation in the suppression pool cooling mode.

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

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 15.2-8 shows the simulated event. Peak neutron flux reaches 183 percent of NBR power while fuel surface heat flux peaks at 102.4 percent of initial value.

15.2.6.3.4 Consideration of Uncertainties

The most conservative characteristics of protection features are assumed. Anticipated plant performance is 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 sec 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.

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

15.2.6.4 Barrier Performance

Loss of Normal and Preferred Station Service Transformers

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.

Loss of All Grid Connections

SRVs 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 1,187 psig, well below the vessel pressure limit of 1,375 psig.

15.2.6.5 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 (Section 15.1.2.5).

15.2.7 Loss of Feedwater Flow

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power). This transient event does not pose any direct threat to the fuel in terms of a power increase from initial conditions, and thus is not reanalyzed for each reload cycle. Evaluation of this event for operation at uprated (3,467 MWt) conditions, and again at EPU (3,988 MWt) in accordance with Reference 8, confirmed that either the HPCS or RCIC system is capable of maintaining adequate core coverage when all normal feedwater flow is lost[3].

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

Table 15.2-11a lists the sequence of events for the EPU evaluation.

The Operator should verify automatic functions and monitor all parameters.

No Operator action is required.

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 RPS responds within 1 sec after this trip to scram the reactor. The low level (L3) scram trip function meets single-failure criterion.

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

Credit is taken for operation of the pressure relief valve (low setpoint) operation of the SRVs to remove decay heat since the bypass becomes ineffective due to main steam line isolation.

15.2.7.2.3 The Effect of Single Failures and Operator Errors

The nature of this event, as explained above, results in a decrease of vessel water level. Key corrective efforts to shut down the reactor are automatic and designed to satisfy single-failure criterion; therefore, any additional failure in these shutdown methods would not aggravate or change the simulated transient. The potential exists for a single relief valve failing to close once it is opened. This is discussed in Section 15.1.4. Either the HPCS or RCIC system is capable of maintaining adequate core coverage and will provide long-term inventory control. 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. The EPU analysis and any loss of feedwater event analyzed subsequent to EPU utilized the SAFER analytical model described in Section S.2 of GESTAR II[5] and in References 6 and 7.

15.2.7.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2. The EPU evaluation was performed using the plant conditions listed in Table 15.0-3a.

15.2.7.3.3 Results

The results of this transient simulation are shown on Figure 15.2-9. Feedwater flow terminates at approximately 5 sec into the transient. Subcooling decreases, causing a reduction in core power level and pressure. As power level is decreased, the turbine steam flow starts to drop off because the pressure regulator is attempting to maintain pressure for the first 15 sec or so. Water level continues to decrease until the vessel level

(L3) scram trip setpoint is reached, whereupon the reactor is shut down and the recirculation pumps are tripped. Main steam line isolation occurs at t=23 sec due to vessel water decreasing to the L2 trip.* At this time, the HPCS and RCIC operation is initiated. MCPR remains above the safety limit since increases in heat flux are not experienced.

The results of the EPU analysis are similar to the previous analysis; results are shown on Figures 15.2-9a and 15.2-9b. Table 15.2-11a provides a sequence of events for the EPU analysis.

15.2.7.3.4 Considerations of Uncertainties

EOC 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 sec of this transient since startup of these pumps occurs in the latter part of this time period and, therefore, these systems have no significant effects on the results of this transient, except as discussed in Section 15.2.7.2.3.

15.2.7.4 Barrier Performance

Peak pressure in the bottom of the vessel reaches 1,103 psig, which is below the ASME Code limit of 1,375 psig for the RCPB. Vessel dome pressure does not exceed 1,093 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

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 (Section 15.1.2.5).

15.2.8 Feedwater Line Break

Refer to Section 15.6.6.

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* For this event, MSIV closure was simulated at Level 2. Subsequent design modifications have lowered the closure setpoint to Level 1. However, HPCS and RCIC initiation at Level 2 will prevent the water level from dropping to L1 and no MSIV closure will occur. Since no safety limits are approached during this event, reanalysis for the setpoint change is not required.

15.2.9 Failure of RHR Shutdown Cooling

The discussion presented in the following sections represents the original event discussion and results. The analysis performed in support of Stretch Power Uprate is discussed in Section 15.2.9.3.5. The limiting case for this event was reanalyzed for EPU as discussed in Section 15.2.9.3.6.

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

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

1. Loss of all offsite ac power.

2. Utilization of safety shutdown equipment only.

3. Operator involvement only after 10 min after coincident assumptions.

These accident-type assumptions change the initial incident (malfunction of RHR suction valve) from a moderate frequency incident to a classification in the DBA status; however, the event is evaluated as a moderate frequency event with its subsequent limits.

15.2.9.1 Identification of Causes and Frequency Classification

15.2.9.1.1 Identification of Causes

The plant is operating at 100 percent NBR thermal power when a long-term loss of offsite power (LOOP) occurs, causing multiple SRV 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 LOOP, 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:

1. No RHR valves have failed in the shutdown cooling mode in total BWR operating experience.

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

15.2.9.2 Sequence of Events and System Operation

15.2.9.2.1 Sequence of Events

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

Identification of Operator Actions

For the early part of the postulated transient, no Operator actions are required. The Operator should verify automatic functions and monitor all parameters.

The discovery of a loss of shutdown cooling is based on the Operator depressurizing the reactor coolant system and attempting to place the shutdown cooling mode of the remaining division in service. Therefore, for the analysis, the following Operator actions are required:

1. The remaining RHR heat exchanger is utilized in the suppression pool cooling mode of RHR, rejecting heat to service water.

2. The vessel pressure is reduced, via SRV actuation, to approximately 100 psig at a rate appropriate to maintain cooldown to less than 100°F per hr.

3. After the inoperability of normal shutdown cooling is determined, an alternate shutdown cooling path is required.

4. All steam feeds are verified to be isolated, and suppression pool cooling flow is redirected to the LPCI injection path, increasing level.

5. A sufficient number of SRVs are power opened to maintain pressure low and establish a liquid flow path back to the suppression pool.

15.2.9.2.2 System Operation

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

15.2.9.2.3 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-min 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 NEDE-23014[1] and NEDO-20533[2].

15.2.9.3.3 Input Parameters and Initial Conditions

Table 15.2-13 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 simply reestablished using other, normal shutdown cooling equipment. In cases where one or both of the RHR shutdown cooling suction containment isolation valves cannot be opened, or neither of the pump inlet blocking valves from the shutdown cooling suction line can be opened, alternate paths are available to accomplish the shutdown cooling function (Figure 15.2-10). An evaluation has been performed assuming the worst single failure that could disable the RHR shutdown cooling valves.

The analysis demonstrates the capability to safely transfer fission product decay heat and other residual heat from the reactor core at such a rate that specified acceptable fuel design limits and the design conditions of the RCPB 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[3] (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 valve discharge piping also fails, the shutdown cooling function would eventually be accomplished as the cooling water would run directly out of the ADS or SRVs, flooding into the drywell.

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

The design evaluation is divided into two phases: 1) full power operation to approximately 100 psig vessel pressure, and 2) approximately 100 psig vessel pressure to cold shutdown (14.7 psig and 200°F) conditions.

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 the main condenser. In the case where the main condenser is not available, the RCIC/HPCS systems together with the nuclear boiler pressure relief system may be used.

For evaluation purposes, however, it is assumed that plant shutdown is initiated by a transient event (LOOP), 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 SRVs. 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 Technical Specification 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 LOOP, is capable of preventing reaching the 100 psig level.

Approximately 100 psig to Cold Shutdown

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

1. Vessel is at 100 psig and saturated conditions.
2. Worst-case single failure is assumed to have occurred (i.e., loss of a division of emergency power).
3. No offsite power is available.

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

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

1. ADS (Dc Divisions 1 and 2).
2. RHR LPCI Loop A (Division 1).
3. HPCS (Division 3).
4. RCIC (Dc Division 1).
5. LPCS (Division 1).

Since availability or failure of Division 3 equipment does not effect the normal shutdown mode, normal shutdown cooling is easily available through equipment powered from only Divisions 1

and 2. It should be noted that, conversely, the HPCS system is always available for coolant injections if either of the other two divisions fails. For failure of Divisions 1 or 2, the following systems are assumed functional:

1. Division 1 fails; Divisions 2 and 3 functional:

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

Assuming the single failure is a failure of Division 1 emergency power, the safety function is accomplished by establishing one of the cooling loops described in Activity C1 of Figure 15.2-11.

2. Division 2 fails; Divisions 1 and 3 functional:

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

Assuming the single failure is the failure of Division 2, the safety function is accomplished by establishing one of the cooling loops described in Activity C2 of Figure 15.2-11. Figures 15.2-12 through 15.2-15 show RHR Loops A, B, and/or C (simplified).

Using the preceding assumptions and following the depressurization rate shown on Figure 15.2-16, the suppression pool temperature is shown on Figure 15.2-17.

15.2.9.3.5 Pool Temperature Sensitivity Analysis

An additional analysis is performed with alternate shutdown cooling to demonstrate that the bulk pool thermal response is expected to be much less severe than that depicted on Figure 15.2-17.

As indicated in Table 15.2-13, Items 2 and 13, low water level and no passive heat sinks are assumed in the calculation that yields the peak pool temperature of 222°F shown on Figure 15.2-17.

The analysis of Case 2' of Section 6A.10 (MSIV closure/scram with alternate shutdown cooling), which assumes an initial core thermal power of 3,536 MWt (102 percent of rated thermal power), is extended to 50 hr to show compliance with design bulk pool temperature limits and to determine the time to cold shutdown (200°F) assuming alternate shutdown cooling is required (see

Section 6A.10.2.5). The alternate shutdown cooling mode assumed for this analysis is that shown on Figures 15.2-13 and 15.2-14. This mode produced the longer time to cold shutdown in the original analysis (see notes to Figure 15.2-11). Case 2' uses passive heat sinks in the suppression pool, a RHR heat exchanger K-factor of 240.2 Btu/sec-°F, and a service water temperature of 82°F (see Table 6A.10-1, Note ***).

Figures 15.2-18 and 15.2-19 show the reactor vessel pressure and temperature responses for Case 2' past the time of cold shutdown. Figure 15.2-20 shows the corresponding long-term suppression pool temperature response. The peak bulk pool temperature obtained for Case 2' of Section 6A.10 is less than the design value of 212°F and significantly less than the original value of 222°F shown on Figure 15.2-17. The analysis for Case 2' produces a time to cold shutdown less than the time (54 hr) given in the notes to Figure 15.2-11 for the same alternate shutdown cooldown mode.

15.2.9.3.6 Peak Suppression Pool Temperature at EPU

The analysis of Case 2’ of Section 6A.10 (MSIV closure/scram with alternate shutdown cooling), which assumes an initial core thermal power of 3,536 MWt (102 percent of the uprated (104.3 percent of OLTP) thermal power), was reanalyzed for EPU at an initial core thermal power of 4,068 MWt (102 percent of EPU rated thermal power). This case is designated as Case 2’’ (see Table 6A.10-1) and uses passive heat sinks in the containment, includes an increase in assumed RHR heat exchanger K-factor to 270 Btu/sec-°F, and an assumed service water temperature of 84°F. The resulting peak suppression pool temperature is 210°F which is slightly below the value predicted for EPU as reported in Table 6A.10-1 and below the design value of 212°F as identified in section 15.2.9.3.5 above. Figure 15.2-20a shows the long-term suppression pool temperature response. The time to cold shutdown (50.2 hr) at EPU remains less than the 54-hr criteria specified in section 15.2.9.3.5 above.

15.2.9.4 Barrier Performance

As noted previously, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Release of coolant to the containment occurs via SRV actuation. Release of radiation to the environment is described in the following section.

15.2.9.5 Radiological Consequences

While this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation (Section 15.1.2.5).

15.2.10 References

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

2. 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).

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

4. Licensing Topical Report, Power Uprate Licensing Evaluation for Nine Mile Point Nuclear Power Station Unit 2, NEDC-31994P, Revision 1, May 1993.

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

6. SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants, NEDE-30996P-A, October 1987.

7. GESTR LOCA and SAFER Models for the Evaluation of LOCA, NEDC-23785-1-P-A, October 1984.

8. GE Nuclear Energy, Constant Pressure Power Uprate, NEDC-33004P-A, Revision 4, Class III (Proprietary), July 2003; and NEDO-33004, Class I (Non-proprietary), July 2003.

TABLE 15.2-1

LOSS OF GENERATOR ELECTRICAL LOAD
SEQUENCE OF EVENTS FOR FIGURE 15.2-1

Note: These results are for Cycle 1. This event is less severe than the case with bypass failure and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
(-)0.015 (approx.)	Turbine generator detection of loss of electrical load.
0	Turbine generator load rejection sensing devices trip to initiate TCV fast closure and main turbine bypass system operation.
0	Fast control valve closure initiates scram trip and RPT.
0.07	TCVs closed.
0.10	Turbine bypass valves start to open.
0.19	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
1.5	Group 1 relief valves actuated.
1.6	Group 2 relief valves actuated.
1.7	Group 3 relief valves actuated.
1.9	Group 4 relief valves actuated.
2.0	Group 5 relief valves actuated.
5.4	Group 5 pressure relief valves start to close.
7.0	All relief valves are closed.

TABLE 15.2-2

LOSS OF GENERATOR ELECTRICAL LOAD WITH BYPASS FAILURE
SEQUENCE OF EVENTS FOR FIGURE 15.2-2

Note: These results are for Cycle 1. Cycle-specific results are presented in Appendix A.

Time (sec)	Event
(-)0.015 (approx.)	Turbine generator detection of loss of electrical load.
0	Turbine generator load rejection sensing devices trip to initiate TCV fast closure.
0	Turbine bypass valves fail to operate.
0	Fast control valve closure initiates scram trip and RPT.
0.07	TCVs closed.
0.19	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
1.2	Group 1 relief valves actuated.
1.3	Group 2 relief valves actuated.
1.4	Group 3 relief valves actuated.
1.5	Group 4 relief valves actuated.
1.6	Group 5 relief valves actuated.
7.0	Group 5 pressure relief valves start to close.
9.1	All relief groups closed.

TABLE 15.2-3

TURBINE TRIP
SEQUENCE OF EVENTS FOR FIGURE 15.2-3

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

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 percent open position and initiate reactor scram trip and a RPT.
0.10	Turbine stop valves closed.
0.10	Turbine bypass valves start to open to regulate pressure.
0.19	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
1.5	Group 1 relief valves actuated.
1.7	Group 2 relief valves actuated.
1.8	Group 3 relief valves actuated.
1.9	Group 4 relief valves actuated.
2.1	Group 5 relief valves actuated.
4.6	Feedwater pump motors trip on L8 high water level.
5.1	Group 5 relief valves start to close.

TABLE 15.2-4

TURBINE TRIP WITH BYPASS FAILURE
SEQUENCE OF EVENTS FOR FIGURE 15.2-4

Note: These results are for Cycle 1. For reload cores, an evaluation is performed to determine if this AOO could potentially alter the previous cycle MCPR operating limit. If it does, the results will be reported in the supplemental reload licensing report.

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 percent open position and initiate reactor scram trip and RPT.
0.10	Turbine stop valves close.
0.19	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
1.25	Group 1 relief valves actuated.
1.34	Group 2 relief valves actuated.
1.42	Group 3 relief valves actuated.
1.51	Group 4 relief valves actuated.
1.60	Group 5 relief valves actuated.
7.0	Group 5 relief valves start to close.
9.0	All relief groups closed.

TABLE 15.2-5

CLOSURE OF ALL MAIN STEAM ISOLATION VALVES
SEQUENCE OF EVENTS FOR FIGURE 15.2-5

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	Initiate closure of all MSIVs.
0.3	MSIVs reach 85 percent open.
0.3	MSIV position trip scram initiated.
2.4	Dome pressure reaches the set pressures of RPT.
2.45	Recirculation system starts coastdown.
2.7	Group 1 pressure relief valve opening setpoint is reached.
3.0	All MSIVs closed.
3.6	All five pressure relief valve groups open.
6.9 (est)	Group 5 pressure relief valves start to close.
8.9 (est)	All pressure relief valve groups closed.

TABLE 15.2-6

TYPICAL RATES OF DECAY FOR CONDENSER VACUUM

Cause	Estimated Vacuum Decay Rate (in Hg/min)
Failure or isolation of steam jet air ejectors	<1
Loss of sealing steam to shaft gland seals	Approximately 1 to 2
Opening of vacuum breaker valves	Approximately 2 to 12
Loss of one or more circulating water pumps	Approximately 4 to 24

TABLE 15.2-7

LOSS OF CONDENSER VACUUM
SEQUENCE OF EVENTS FOR FIGURE 15.2-6

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
-3.0	Initiate simulated loss of condenser vacuum at 2 in Hg/sec.
0	Low condenser vacuum main turbine trip and feedwater initiated.
0.0	Main turbine trip indicated turbine bypass operation.
0.01	Main turbine stop valves reach 90 percent open position and initiate reactor scram trip and RPT.
0.10	Turbine stop valves closed and turbine bypass valves start to open to regulate pressure.
0.19	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
1.5	Group 1 relief valves actuated.
1.6	Group 2 relief valves actuated.
1.8	Group 3 relief valves actuated.
1.9	Group 4 relief valves actuated.
2.1	Group 5 relief valves actuated.
4.6	Feedwater pump motors tripped on high level (L8).
5.0	Low condenser vacuum initiates turbine bypass valve closure and MSIV closure.
5.1	Group 5 relief valves start to close.
5.6	Turbine bypass valves closed.
6.8 (est)	Relief groups 1-5 closed.

TABLE 15.2-7 (Cont'd.)

Time (sec)	Event
8.0	MSIVs closed.
9.0	Group 1 relief valves reactuated on high pressure.
9.5	Group 2 relief valves reactuated on high pressure.
14.2	Group 2 relief valves start to close.

TABLE 15.2-8

TRIP SIGNALS ASSOCIATED WITH LOSS OF CONDENSER VACUUM

Vacuum (in Hg)	Protective Action Initiated
27 to 28	Normal vacuum range
20 to 23	Main turbine trip and feedwater turbine trip (stop valve closures)
7 to 10	MSIV closure and bypass valve closure

TABLE 15.2-9

LOSS OF ALL RESERVE AND NORMAL TRANSFORMER
SEQUENCE OF EVENTS FOR FIGURE 15.2-7

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	Loss of all reserve power transformer and the normal station service transformer occurs.
0	Recirculation system pump motors are tripped.
0	Condensate and booster pumps are tripped.
0	Condenser circulating water pumps are tripped.
0	Feedwater pump motors are tripped on loss of power.
2.0	Reactor scram initiated on loss of power.
2.0	MSIVs closure initiated.
5.0	Group 1 relief valves actuated.
5.2	Group 2 relief valves actuated.
5.3	Group 3 relief valves actuated.
5.4	Group 4 relief valves actuated.
5.7	Group 5 relief valves actuated.
8.2	Group 5 relief valves begin to close.
11.6 (est)	All relief groups closed.
12.4	Group 1 relief valves reactuated on high pressures.
13.3	Group 2 relief valves reactuated on high pressure.
20	Both relief groups closed.
23	RCIC and HPCS systems initiated on low water level (L2).
26+	Group 1 relief valves cycle open and close on pressure.

TABLE 15.2-10

LOSS OF ALL GRID CONNECTIONS
SEQUENCE OF EVENTS FOR FIGURE 15.2-8

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
(-)0.015 (approx.)	Loss of grid causes turbine generator to detect a loss of electrical load.
0	Turbine generator power-load unbalance devices trip to initiate TCV fast closure and turbine bypass system operation.
0	Condenser circulating water pumps are tripped.
0	Recirculation system pump motors are tripped.
0	Fast control valve closure initiates a reactor scram trip.
0	Feedwater condensate and booster pumps are tripped.
0.07	TCVs closed.
0.10	Turbine bypass valves start to open to regulate pressure.
1.5	Group 1 SRVs actuated.
1.6	Group 2 SRVs actuated.
1.8	Group 3 SRVs actuated.
2.0	MSIV closure initiated.
2.0	Group 4 SRVs actuated.
2.1	Group 5 SRVs actuated.
5.1	Group 5 SRVs start to close.
7.0 (est)	All relief groups closed.

TABLE 15.2-10 (Cont'd.)

Time (sec)	Event
8.4	Group 1 SRVs reactivated.
8.9	Group 2 SRVs reactivated.
14.2	Group 1 SRVs start to close.

TABLE 15.2-11

LOSS OF FEEDWATER FLOW
SEQUENCE OF EVENTS FOR FIGURE 15.2-9

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for each reload cycle. Evaluation of this event for power uprate is documented in Reference 4.

Time (sec)	Event
0	Trip of all feedwater pumps initiated.
3.7	Recirculation runback initiated with narrow range sensed level less than L4 and feedwater pumps off.
4.8	Feedwater flow decays to zero.
8.7	Vessel water level (L3) trip initiates scram trip.
8.7	Vessel water level (L3) trip initiates recirculation pump system trip.
23*	Vessel water level (L2) trip initiates main steam line isolation.
23	Vessel water level (L2) trip initiates RCIC and HPCS systems operation (not simulated).
26	MSIVs fully closed.
41	Group 1 pressure relief valves actuated.
48	Group 1 pressure valves closed.
50±	Group 1 pressure relief valves cycle open and close on pressure.

* For this event, MSIV closure was simulated at Level 2. Subsequent design modifications have lowered the closure setpoint to Level 1. However, HPCS and RCIC initiation at Level 2 will prevent the water level from falling to Level 1 and no MSIV closure will occur. Since no safety limits are approached during this event, a reanalysis for the setpoint change is not required.

TABLE 15.2-11a

LOSS OF FEEDWATER FLOW
SEQUENCE OF EVENTS FOR FIGURES 15.2-9a and 15.2-9b

Note: These results are for EPU. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

Time (sec)	Event
0	Trip of all feedwater pumps initiated. Recirculation runback initiated, low-level scram initiated.
1.0	Feedwater flow decays to zero.
68	RCIC initiates
783	Minimum downcomer level achieved.
1007	Minimum upper plenum level achieved.

TABLE 15.2-12

SEQUENCE OF EVENTS FOR FAILURE OF RHR SHUTDOWN COOLING

Note: These results are not fuel cycle-dependent. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

Approximate Elapsed Time (min)	Event
0	Reactor is operating at 100 percent rated core thermal power when LOOP occurs initiating plant shutdown.
0	Concurrently, loss of Division power (i.e., loss of one diesel generator) occurs.
10	Suppression pool cooling initiated*.
33	Controlled depressurization initiated (100°F/hr) using selected SRVs.
167	Blowdown to approximately 100 psig completed.
167	RHR shutdown cooling fails to be initiated.
184	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.

* See Table 15.2-9 for detailed sequence of events for loss of ac power transient.

TABLE 15.2-13

INPUT PARAMETERS FOR EVALUATION OF FAILURE OF RHR SHUTDOWN COOLING

1.	Initial power	Corresponds to 100% rated core thermal power*
2.	Suppression pool mass (low water level), lbm	8,944,655
3.	RHR (KHX value), Btu/sec/°F	199.2*
4.	Initial vessel conditions	
	Pressure, psia	1,055
	Temperature, °F	551
5.	Initial primary fluid inventory, lbm	646,801
6.	Initial pool temperature, °F	90
7.	Service water temperature, °F	77*
8.	Vessel heat capacity, Btu/lbm/°F	0.123
9.	HPCS on-off water level, ft	
	On	40.8
	Off	48.6
10.	HPCS flow rate, lbm/sec	869
11.	LPCI flow rate per loop, lbm/sec	982
12.	LPCS flow rate, lbm/sec	869
13.	Passive containment heat sink	None

* A design evaluation has been performed with an initial power of 102 percent of rated core thermal power, a heat exchanger K-factor of 240.2 and an increased service water temperature of 82°F (see Table 6A.10-1, Note ***). This evaluation concluded that the time to cold shutdown will be significantly reduced with the use of the 20 percent excess capacity available to the RHR heat exchangers.

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

15.3.1 Recirculation Pump Trip

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle. This event is not considered limiting up to a 20-percent uprate; therefore, it was not analyzed for EPU (3,988 MWt).

15.3.1.1 Identification of Causes and Frequency Classification

15.3.1.1.1 Identification of Causes

The recirculation pump motor can be tripped by design for intended reduction of other transient core and RCPB effects as well as randomly by unpredictable operational failures. Intentional tripping occurs in response to:

1. Reactor vessel water level L2 setpoint trip.
2. TCV fast closure or stop valve closure.
3. High pressure setpoint trip.
4. Motor branch circuit overcurrent protection.
5. Motor overload protection.
6. Suction block valve not fully open.

Random tripping occurs in response to:

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

15.3.1.1.2 Frequency Classification

<u>Trip of One Recirculation Pump</u>

This transient event is categorized as one of moderate frequency.

<u>Trip of Two Recirculation Pumps</u>

This transient event is categorized as one of moderate frequency.

15.3.1.2 Sequence of Events and Systems Operation

15.3.1.2.1 Sequence of Events

Trip of One Recirculation Pump

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

Trip of Two Recirculation Pumps

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

Identification of Operator Actions

The Operator should verify automatic functions and monitor all parameters.

No Operator actions are required.

Trip of One or 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 HPCS or RCIC operation, monitoring reactor water level, and pressure control after shutdown. When both reactor pressure and level are under control, the Operator should terminate 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 System Operation

Trip of One or Two Recirculation Pumps

Analysis of this event assumes normal functioning of plant instrumentation and controls, and plant protection and RPSs. 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 the pressure relief valve system operation.

15.3.1.2.3 The Effect of Single Failures and Operator Errors

Trip of One or Two Recirculation Pumps

Tables 15.3-1 and 15.3-2 list the vessel level (L8) trip event as the first response to initiate corrective action in this transient. The level (L8) trip is intended to prohibit moisture carryover to the main turbine. Multiple level sensors are used to sense and detect when the water level reaches the L8 setpoint. At this point, a single failure will neither initiate nor impede a turbine trip signal. Turbine trip signal transmission circuitry, however, is not built to the single-failure criterion. The result of a failure at this point would have the effect of delaying the pressurization "signature." However, high moisture levels entering the turbine will be detected by high levels in the turbine's moisture separators, resulting in a trip of the unit. This scram trip signal is designed in such a way that a

single failure neither initiates nor impedes 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 15.0-2.

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

15.3.1.3.3 Results

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 10 sec. However, the ratio of diffuser mass flow to pump mass flow in the active jet pumps increases considerably and produces approximately 125 percent of normal diffuser flow and 62 percent of rated core flow. MCPR has no significant change, thus the fuel thermal limits are not violated.

Trip of Two Recirculation Pumps

Figure 15.3-2 shows graphically this transient with minimum specified rotating inertia. The MCPR remains unchanged at its initial operating value for this event. 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 the feed pump motors and scramming the reactor. Subsequent events, such as main steam line isolation and initiation of RCIC and HPCS systems occurring late in this event, have no significant effect on the results.

15.3.1.3.4 Consideration of Uncertainties

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

Actual pump and pump-motor drive line rotating inertias are expected to be somewhat greater than the minimum design values assumed in this simulation. Anticipated inertia values 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

Trip of One Recirculation Pump

Figure 15.3-1 results indicate a basic initial reduction in system pressures from the initial conditions. Upon high-level turbine trip and turbine stop valve closure, pressure rises to 1,093 psig at the vessel bottom which is well below the 1,375-psig limit allowable by the applicable code. Therefore, the RCPB barrier is not threatened.

Trip of Two Recirculation Pumps

The results shown on Figure 15.3-2 indicate peak pressures stay well below the 1,375-psig limit allowed by the applicable code. Therefore, the barrier pressure boundary is not threatened.

15.3.1.5 Radiological Consequences

While this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SVR operation. Since this activity is contained in the primary containment, there will be no exposures to operating personnel (Section 15.1.2.5).

15.3.2 Recirculation Flow Control Failure - Decreasing Flow

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle. This event is not considered limiting up to a 20-percent uprate; therefore, it was not analyzed for EPU (3,988 MWt).

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 recirculation flow controller or the flux controller generates 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/sec. 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 closing stroking rate of 60 percent/sec 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

Fast Closure of One Main Recirculation Valve

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

Fast Closure of Two Main Recirculation Valves

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

Identification of Operator Actions

The Operator should verify automatic functions and monitor all parameters.

No Operator action is required.

15.3.2.2.2 Systems Operation

Fast Closure of One Main Recirculation Valve

Normal plant instrumentation and control is assumed to function. Credit is taken for scram in response to vessel high water level (L8) trip.

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) turbine 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. The fast closure of two recirculation valves instead of one would be the envelope case for the additional SEF 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 15.0-2.

The less negative void coefficient in Table 15.0-2 was used for these analyses.

<u>Fast Closure of One Main Recirculation Valve</u>

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

<u>Fast Closure of Two Main Recirculation Valves</u>

A downscale failure of either the master recirculation flow controller or the flux controller generates 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/sec. Recirculation loop flow is allowed to decrease to approximately 25 percent of rated. This is the flow expected when the flow control valves (FCVs) are maintained at a minimum open position at high pump speed.

15.3.2.3.3 Results

<u>Fast Closure of One Recirculation Valve</u>

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. Fuel thermal limits are not threatened.

<u>Fast Closure of Two Recirculation Valves</u>

Figure 15.3-4 illustrates the expected transient which is similar to the two-pump trip transient described in Section 15.3.1.3.3. The rate of flow decrease is somewhat slower, however, because of the limiter operation; thus, the subsequent events occur later in time. The initial flow decrease is also slower than the case of closure of one valve at its maximum rate described above.

Water level swell results in high water level (L8) trip of the main turbine and feedwater pump motors which, in turn, results in reactor scram and RPT initiation and shutdown of the reactor. MCPR remains unchanged from the initial operating limit and fuel thermal limits are not threatened.

15.3.2.3.4 Consideration of Uncertainties

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

These analyses, unlike the pump trip series, are unaffected by deviations in pump/pump motor and drive line inertias since it is the main valve that causes rapid recirculation decreases.

15.3.2.4 Barrier Performance

Fast Closure of One Recirculation Valve

The high water level turbine trip at about 6.9 sec results in pressure increase to a value above the setpoint of the first group of SRVs. The pressure in the vessel dome is limited to 1,100 psig which is well below the vessel pressure limit. The event does not result in a temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed, and these barriers maintain their integrity and function as designed.

Fast Closure of Two Recirculation Valves

Two groups of SRVs open as vessel pressure exceeds the actuation setpoints after the high water level turbine trip and steam is briefly discharged to the suppression pool. Peak pressure at the vessel bottom reaches 1,120 psig, which is below the ASME Code limit for the RCPB. Peak pressure in the vessel dome is 1,109 psig.

15.3.2.5 Radiological Consequences

While this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposures to operating personnel (Section 15.1.2.5).

15.3.3 Recirculation Pump Seizure

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle. This event is not limiting up to a 20-percent uprate; therefore, it was not analyzed for EPU (3,988 MWt).

15.3.3.1 Identification of Causes and Frequency Classification

The seizure of a recirculation pump is considered as a DBA event[1]. It has been evaluated as being a very mild event in relation to other DBAs such as the LOCA.

Refer to Section 5.1 for specific mechanical considerations and Chapter 7 for electrical aspects.

The seizure event postulated 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.

Identification of Operator Actions

The Operator should verify automatic functions and monitor all parameters.

No Operator action is required.

15.3.3.2.2 Systems Operation

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

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, Trip of Two Recirculation Pumps. Refer to Appendix 15A for further details.

15.3.3.3 Core and System Performance

15.3.3.3.1 Mathematical Model

The nonlinear dynamic 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 15.0-2.

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 NBR steam flow. Also, the reactor is assumed to be operating at thermally limited conditions. The void coefficient is adjusted to the most conservative value, that is, the least negative value in Table 15.0-2.

15.3.3.3.3 Results

Figure 15.3-5 presents the results of the first accident. Core coolant flow drops rapidly, reaching its first minimum value in approximately 1.75 sec. CPR 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 pump motors and stop valve closure scram and RPT. The scram conditions impose no threat to thermal limits. Additionally, the momentary opening of the bypass valves and some of the SRVs limit the pressure well within the range allowed by the ASME Code. Therefore, the RCPB is not threatened by overpressure.

15.3.3.3.4 Considerations of Uncertainties

Considerations of uncertainties are included in the GETAB analysis.

15.3.3.4 Barrier Performance

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

15.3.3.5 Radiological Consequences

While this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment there will be no exposure to operating personnel (Section 15.1.2.5).

15.3.4 Recirculation Pump Shaft Break

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

This event is not considered limiting up to a 20-percent uprate; therefore, it was not analyzed for EPU (3,988 MWt).

15.3.4.1 Identification of Causes and Frequency Classification

The breaking of the shaft of a recirculation pump is considered a DBA event. It has been evaluated as a very mild event in relation to other DBAs such as the LOCA. The analysis has been conducted with consideration of single- or two-loop operation. Refer to Chapter 5 for specific mechanical considerations and Chapter 7 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

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

15.3.4.1.2 Frequency Classification

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

15.3.4.2 Sequence of Events and Systems Operations

15.3.4.2.1 Sequence of Events

A postulated instantaneous break of the pump motor shaft of one recirculation pump as discussed in Section 15.3.4.1.1 causes the core flow to decrease rapidly resulting in water level swell in the reactor vessel. When the vessel water level reaches the high water level setpoint (Level 8), main turbine trip and feedwater pump trip are initiated. Subsequently, reactor scram and the remaining recirculation pump trip will be initiated due to the turbine trip. Eventually, the vessel water level is controlled by HPCS and RCIC flow.

<u>Identification of Operator Actions</u>

The Operator should verify automatic functions and monitor all parameters.

No Operator action is required.

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 trip the main turbine and feedwater pumps and

RPS to scram the reactor on turbine stop valve closure. High system pressure is limited by the pressure relief system 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, Trip of One or Two Recirculation Pumps.

Assumption of a SEF or SOE in other equipment has been examined and this has led to the conclusion that no other credible failure exists for this event. Therefore, the bounding case has been considered. Refer to Appendix 15A for more details.

15.3.4.3 Core and System Performance

The severity of this pump shaft break event is bounded by the pump seizure event described in Section 15.3.3. This can be easily demonstrated by consideration of those two events discussed in the following paragraph. Since this event is less limiting than the event described in Section 15.3.3, only qualitative evaluation is provided. Therefore, no discussion of mathematical model, input parameters, and consideration of uncertainties, etc., is necessary.

If this extremely unlikely event occurs, core coolant flow drops rapidly. The level swell produces a reactor scram and trip of the main turbine and the feedwater pumps. Since heat flux decreases much more rapidly than the rate at which heat is removed by the coolant, there is no threat to thermal limits. Additionally, the bypass valves and momentary opening of some of the SRVs limit the pressures well within the range allowed by the ASME Boiler and Pressure Vessel Code. Therefore, the RCPB is not threatened by overpressure.

The severity of this pump shaft break event is bounded by the pump seizure event (Section 15.3.3). This can be demonstrated easily by consideration of these two events. In either of these two events, the recirculation drive flow of the affected loop decreases rapidly. In the case of the pump seizure event, the loop flow decreases faster than the normal flow coastdown as a result of the large hydraulic resistance introduced by the stopped rotor. For the pump shaft break event, the hydraulic resistance caused by the broken pump shaft is less than that of the stopped rotor for the pump seizure event. Therefore, the core flow decrease following a pump shaft break effect is slower than the pump seizure event. Thus, it can be concluded that the potential effects of the hypothetical pump shaft break accident are bounded by the effects of the pump seizure event.

15.3.4.4 Barrier Performance

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

15.3.4.5 Radiological Consequences

While this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation (Section 15.1.2.5).

15.3.5 Reference

1. General Electric Standard Application for Reactor Fuel, including 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

Trip One Recirculation Pump

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	Trip of one recirculation pump initiated.
10.49	Vessel water level (L8) trip initiates turbine trip and feedwater pump trip.
10.49	Recirculation pump trip is initiated.
10.5	Main turbine stop valves reach 10% closed position and initiate reactor scram trip.
10.6	Turbine trip initiates bypass operation.
16.0	Group 1 of pressure relief valves actuated.
20.5	Group 1 pressure relief valves start to close.
22.6	Relief valves closed.
36.0	Turbine bypass valves start to close.
37.3	Turbine bypass closed.
43.3	Turbine bypass reopens on pressure increase at turbine inlet.
45.1	RCIC and HPCS systems initiate on low level (L2).

TABLE 15.3-2

SEQUENCE OF EVENTS FOR FIGURE 15.3-2

Trip Both Recirculation Pumps

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	Trip of both recirculation pumps initiated.
5.6	Vessel water level (L8) trip initiates turbine trip and feedwater pump trip.
5.6	Turbine trip initiates bypass operation.
5.61	Main turbine stop valves reach 10% closed position and initiate reactor scram trip.
5.7	Turbine stop valves closed and turbine bypass valves start to open to regulate pressure.
8.7	Group 1 relief valves actuated.
9.1	Group 2 pressure relief valves actuated.
12.6	Group 2 pressure relief valves start to close.
15.2	Both relief groups closed.
30	Turbine bypass valves start to close.
31	Turbine bypass closed.
38	Turbine bypass reopens on pressure increase at turbine inlet.
39	RCIC and HPCS systems initiate on low level (L2).

TABLE 15.3-3

SEQUENCE OF EVENTS FOR FIGURE 15.3-3

Fast Closure of One Main Recirculation Valve

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	Initiate fast closure of one main recirculation valve.
1.8	Recirculation control valve at minimum flow position.
6.9	High vessel water level (L8) trip initiates main turbine and feedwater pump motor trips. Bypass operation is initiated.
6.9	Main turbine stop valves reach 10% closed position and initiate reactor scram trip and RPT.
7.0	Main turbine stop valves closed and turbine bypass valves start to open to regulate pressure.
7.0	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
11	Group 1 pressure relief valves actuated.
19	Group 1 relief valves closed.
33	Turbine bypass starts to close.
34	Turbine bypass closed.
40	Turbine bypass reopens on pressure increase at turbine inlet.
43	RCIC and HPCS systems initiation on low level (L2).

TABLE 15.3-4

SEQUENCE OF EVENTS FOR FIGURE 15.3-4

Fast Closure of Both Main Recirculation Valves

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	Initiate fast closure of both main recirculation valves.
8.58	High vessel water level (L8) trip initiates main turbine and feedwater pump motor trips.
8.58	Main turbine trip initiates bypass operation.
8.59	Main turbine stop valves reach 10% closed position and initiate reactor scram trip and RPT.
8.69	Turbine stop valves closed and turbine bypass valves start to open to regulate pressure.
8.72	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
9.3	Both recirculation control valves at minimum flow position.
11.4	Group 1 pressure relief valves actuated.
11.7	Group 2 pressure relief valves actuated.
16.0	Group 2 pressure relief valves start to close.
18.4	Both relief groups closed.
33	Turbine bypass valves start to close.
34	Turbine bypass closed.
41	Turbine bypass reopens on pressure increase at turbine inlet.
50+	Turbine bypass closed. RCIC and HPCS systems initiation on low level (L2).

TABLE 15.3-5

SEQUENCE OF EVENTS FOR FIGURE 15.3-5

Recirculation Pump Seizure

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	Single pump seizure was initiated.
0.90	Jet pump diffuser flow reverses in seized loop.
4.04	High vessel water level (L8) trip initiates main turbine and feedwater turbine trips.
4.04	Main turbine trip initiates bypass operation.
4.05	Main turbine stop valves reach 10% closed position and initiate reactor scram trip and RPT.
4.14	Turbine stop valves closed and turbine bypass valves start to open to regulate pressure.
4.18	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
7.0 (est)	Group 1 and 2 pressure relief valves actuated.
12 (est)	Group 2 pressure relief valves start to close.
14	Both relief groups closed.
29	Turbine bypass starts to close.
31	Turbine bypass closed.
36	Turbine bypass reopens on pressure increase at turbine inlet.
40	RCIC and HPCS systems initiation on low level (L2).

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Cycle-specific analyses are discussed in Appendix A, Section A.15.4.

15.4.1 Rod Withdrawal Error - Low Power

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. This event was evaluated for the EPU (3,988 MWt). It was determined that the licensing basis criterion for fuel failure is still satisfied. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

15.4.1.1 Control Rod Withdrawal 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 result in an inadvertent rod withdrawal error (RWE) while in the REFUEL mode.

15.4.1.1.2 Sequence of Events and Systems Operation

Initial Control Rod Withdrawal

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

Fuel Insertion With Control Rod Withdrawn

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that prior to loading into a cell, the control rod for that cell be fully inserted. This requirement is backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the REFUEL position, the interlocks prevent the platform from being moved over the core if a control rod is withdrawn and fuel is on the hoist. Likewise, if the refueling platform is over the core and fuel is on the hoist, control rod motion is blocked by the interlocks.

Second Control Rod 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.

<u>Control Rod Withdrawal Without Fuel</u>

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

<u>Identification of Operator Actions</u>

No supplementary Operator actions are required to preclude this event.

<u>Effect of Single Failure and Operator Errors</u>

If any one of the operations involved in initial failure or error is followed by any other SEF or SOE, the necessary safety actions (e.g., rod block or scram) are taken automatically. 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 does 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 7.7.1.5). As a result, no radioactive material is ever released from the fuel, making it unnecessary to assess any radiological consequences.

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

15.4.1.1.4 Barrier Performance

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

15.4.1.2 Continuous Rod Withdrawal During Reactor Startup

15.4.1.2.1 Identification of Causes and Frequency Classification

The probability of initial causes of errors for this event alone is considered low enough to warrant its being categorized as an infrequent incident. The probability of further development of this event is extremely low because it is contingent upon the failure of the nuclear measurement analysis and control rod worth minimizer (NUMAC RWM) system concurrent with a high worth rod, out-of-sequence rod selection contrary to procedures, plus Operator nonacknowledgement of continuous alarm annunciations prior to safety system actuation.

15.4.1.2.2 Sequence of Events and Systems Operation

Sequence of Events

The NUMAC RWM prevents the Operator from selecting and withdrawing an out-of-sequence control rod. A special analysis described in Section 15.4.1.3 shows that, even for the unlikely event where the NUMAC RWM fails to block the continuous withdrawal of an out-of-sequence rod, the licensing basis criterion for fuel failure is still satisfied.

Identification of Operator Actions

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

Effects of Single Failure and Operator Errors

If any one of the operations involved the initial failure or error and is followed by another SEF or SOE, the necessary safety actions (e.g., rod blocks) are automatically taken prior to any limit violation. Refer to Appendix 15A for details.

15.4.1.2.3 Core and System Performance

See Section 15.4.1.3.

15.4.1.2.4 Barrier Performance

See Section 15.4.1.3.

15.4.1.2.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel. (See Section 15.4.1.3.)

15.4.1.3 Special Analysis

The NUMAC RWM constraints on rod sequence will prevent the continuous withdrawal of an out-of-sequence rod. This analysis was performed to demonstrate that, even for the unlikely event where an out-of-sequence control rod is withdrawn at the maximum allowable normal drive speed, the licensing basis criterion for fuel failure will not be exceeded.

This analysis is extracted for Unit 2 application from a generic study (Reference 7) previously docketed for Hatch 2, Fermi 2, and LaSalle. The analysis concludes that as a result of continuous withdrawal of an out-of-sequence rod in the startup range, the reactor is shut down and peak power is limited to 23 percent of rated thermal power, with the peak fuel enthalpy well below the licensing basis fuel failure threshold of 170 cal/gm. Therefore, evaluation of barrier performance and radiological consequences is not necessary.

15.4.2 Rod Withdrawal Error at Power

This transient event is evaluated for each reload as a potentially limiting event. This event is dependent on local power characteristics and the setting of the rod block instrumentation, rather than core average power conditions.[8] Results of the cycle-specific evaluation at EPU conditions are provided in Appendix A.

15.4.2.1 Identification of Causes and Frequency Classification

15.4.2.1.1 Identification of Causes

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

15.4.2.1.2 Frequency Classification

The probability of this event is considered low enough to warrant its being categorized as an infrequent incident. However, because of the lack of a sufficient frequency data base, this transient disturbance is analyzed as an incident of moderate frequency until the frequency can be further evaluated and justified.

15.4.2.2 Sequence of Events and Systems Operation

15.4.2.2.1 Sequence of Events

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

15.4.2.2.2 System Operations

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

While operating in the power range in a normal mode of operation (except as noted in Section 15.4.2.3.2), the RO makes a procedural error and withdraws the maximum worth control rod until the RBM system inhibits further withdrawal. Under most normal operating conditions no Operator action is required since the transient that will occur will be very mild. Should the peak linear power design limits be exceeded, the nearest local power range monitor (LPRM) will detect this phenomenon and alarm. The Operator should acknowledge this alarm and take appropriate action to rectify the situation.

If the RWE is severe enough, the RBM system will alarm, at which time the Operator should acknowledge the alarm and take corrective action. Even for extremely severe conditions (i.e., highly abnormal control rod patterns and operating conditions, and assuming that the Operator ignores all alarms and warnings and continues to withdraw the control rod), the RBM system will block further withdrawal of the control rod before the fuel reaches the point of boiling transition or the 1-percent plastic strain limit imposed on the clad.

15.4.2.2.3 Effect of Single Failures and Operator Errors

The effect of Operator errors has been discussed previously. It was shown that Operator errors (which initiated this transient) cannot impact the consequences of this event due to the RBM system. The RBM system is designed to be single-failure proof; therefore, termination of this transient is assured (see Appendix 15A for details).

15.4.2.3 Core and System Performance

15.4.2.3.1 Mathematical Model

For this transient the reactivity insertion rate is very slow; therefore, it is adequate to assume that the core has sufficient time to equilibrate (i.e., both the neutron flux and heat flux are in phase). Using this assumption, this transient is calculated using a steady-state, three-dimensional, coupled nuclear-thermal-hydraulics computer program. All spatial effects are included in the calculation. This program is described in Section 3 of GESTAR II.

The primary output from this code, in addition to the basic nuclear parameters, is: the variation of the linear heat rate (LHGR); the variation of the MCPR; the total reactor power; and the variation of the in-core instruments during the transient. A detector response code uses the instrument responses to predict the RBM action under the specified condition for the RWE.

The analytical methods and assumptions used in evaluating the consequences of this accident are considered to provide a realistic yet conservative assessment of the consequences.

15.4.2.3.2 Input Parameters and Initial Conditions

The number of possible RWE transients is extremely large due to the number of control rods and the wide range of exposures and power levels. In order to encompass all possible RWEs that could conceivably occur, a limiting analysis is defined that provides a conservative assessment of the consequences.

The conservative assumptions are:

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

2. The core is operating at rated conditions.

3. The reactor is in its most reactive state and devoid of all xenon. This ensures that the amount of excess reactivity that must be controlled by the movable control rods is maximum.

4. The Operator has fully inserted the maximum worth rod prior to its removal and selected the remaining control rod pattern in such a way as to approach thermal limits in the fuel bundles in the vicinity of the rod to be withdrawn (Figure 15.4-1). It should be emphasized that this control rod configuration would be highly abnormal and could only be achieved by deliberate Operator action or by numerous Operator errors.

5. The Operator has ignored all warnings during the transient.

6. Of the four LPRM strings nearest to the control rod being withdrawn, the two highest-reading LPRMs during the transient have failed.

7. One of the two instrument channels is bypassed and OOS. The A and C LPRM chambers input to one channel while the B and D chambers input to the other. The channel with the greatest response is bypassed.

These conservative assumptions provide a high degree of assurance that the transient, as analyzed, bounds all RWEs that could possibly occur. Table 15.4-2 presents the other parameters used in the analysis of this event.

<u>Rod Block Monitor System Operation</u>

The RBM system minimizes the consequences of a RWE by blocking the motion of the control rod before the safety limits are exceeded.

The RBM has three trip levels (rod withdrawal permissive removed). The upscale trip levels are set at a fixed level above the reference simulated thermal power (STP) and will automatically vary as step functions of the reference STP.

15.4.2.3.3 Results

The consequences of this transient are relatively mild and neither localized nor gross occurrence of boiling transition or violation of the 1-percent plastic strain limit on the cladding occur. The variation in the MCPR and MLHGR, as a function of withdrawal of the highest worth rod, is presented on Figures 15.4-2 and 15.4-3, respectively. The bundles presented on Figures 15.4-2 and 15.4-3 represent the envelope of the MCPR and the MLHGR for each 2-ft interval during the transient. Variation in the total reactor power is also shown on these figures. Although these figures show the change in thermal limits from the fully inserted to the fully withdrawn position, the control rod is automatically blocked at 5.0 ft, even under the worst set of assumptions. The variations in the signal response of the two independent channels is shown on Figures 15.4-4 and 15.4-5. With a setpoint of 106 percent, the rod is shown to block at 5.0 ft resulting in a CPR of 0.18 and MLHGR of 14.84 kW/ft.

15.4.2.3.4 Considerations of Uncertainties

The conservative assumptions assuring that this event has been conservatively analyzed are discussed in Section 15.4.2.3.2.

15.4.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since this is a localized event with very little change in the gross core characteristics. Typically, an increase in total core power is less than 5 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 evaluation provided in Sections 15.4.1 and 15.4.2.

15.4.4 Abnormal Startup of Idle Recirculation Pump

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle. This event is non-limiting for EPU (3,988 MWt); therefore, it was not analyzed.

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.

<u>Normal Restart of Recirculation Pump at Power</u>

This transient is categorized as an incident of moderate frequency.

<u>Abnormal Startup of Idle Recirculation Pump</u>

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 15.4-6.

An assumed Operator error initiates the event. No further Operator action is required in the analysis.

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 results in a reactor scram on flux. See Appendix 15A for details.

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 15.0-2. 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 active recirculation loop is operating with the FCV position that produces about 92 percent of normal rated jet pump diffuser flow in the active jet pumps.

The core is receiving 40 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 FCV 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.) If this event has been analyzed with the recirculation FCV opened at a larger position, the resultant flux spike would have exceeded the APRM scram setpoint causing a reactor scram. Due to the scram early in the event, the surface heat flux would peak at a lower value than if the scram had not occurred. Therefore, the case, as analyzed in this section which did not result in a scram, bounds those events with the FCV opened at a larger position.

15.4.4.3.3 Results

The transient response to the incorrect startup of a cold, idle recirculation loop is shown on Figure 15.4-6. Shortly after the pump begins to operate, a surge in flow from the started jet pump diffusers causes the core inlet flow to rise sharply. The motor approaches rated speed in approximately 4 sec because of the assumed minimum pump and motor inertia. The diffuser flows on the started loop side of the reactor, initially slightly reversed, increase to about 21 percent of rated. The resulting cold loop and hot downcomer flows to the core raising the core inlet subcooling.

A short-duration neutron flux peak is produced to just above 122 percent of NBR as the colder, increasing core flow reduces the void volume. Surface heat flux follows the slower response of the fuel and peaks at 87 percent of rated before decreasing after the cold water flows out of the loop in about 16 sec. No damage occurs to the fuel barrier and MCPR remains above the safety limit. The transient approaches within approximately 4 percent of the APRM thermal power setpoint.

15.4.4.3.4 Consideration of Uncertainties

This transient is analyzed for a much higher initial power level 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 power range, 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 15.4-6).

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

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle. This event is non-limiting for EPU (3,988 MWt); therefore, it was not analyzed.

15.4.5.1 Identification of Causes and Frequency Classification

15.4.5.1.1 Identification of Causes

Failure of the master recirculation flow controller or 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

<u>Fast Opening of One Recirculation Valve</u>

Table 15.4-4 lists the sequence of events for Figure 15.4-7.

<u>Fast Opening of Two Recirculation Valves</u>

Table 15.4-5 lists the sequence of events for Figure 15.4-8.

Identification of Operator Actions

The Operator should transfer flow control to manual, reduce flow to minimum, and identify the cause of failure.

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 RPS. Operation of engineered safeguards is not expected.

15.4.5.2.3 The Effect of Single Failures and Operator Errors

Both 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 RPS, 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 15.0-3.

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 55 percent NBR power and 35.7 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/sec.

Maximum stroking rate of a single recirculation loop valve for a loop controller failure is limited by hydraulics to 30 percent/sec.

15.4.5.3.3 Results

Fast Opening of One Recirculation Valve

Figure 15.4-7 shows the analysis of a failure where one recirculation loop main valve is opened at its maximum valve opening stroking rate of 30 percent/sec.

The rapid increase in core flow causes a sharp rise in neutron flux initiating a reactor scram at approximately 1.1 sec. The peak neutron flux reached is 308 percent of NBR value, while the accompanying average fuel surface heat flux reaches 78 percent of NBR at approximately 2.0 sec. MCPR remains considerably above the safety limit and fuel center temperature increases only 387°F. Reactor pressure is discussed in Section 15.4.5.4.

Fast Opening of Two Recirculation Valves

Figure 15.4-8 illustrates the failure where both recirculation loop main valves are opened at a maximum valve opening stroking rate of 11 percent/sec. It is similar to the fast opening of one recirculation valve. Flux scram occurs at approximately 1.2 sec, peaking at 241 percent of NBR while the average surface heat flux reaches 74.1 percent of NBR at approximately 2.2 sec. MCPR remains considerably above the safety limit, and fuel center temperature increases 330°F.

As indicated above, this is the most severe set of conditions under which this transient may occur. The results expected from an actual occurrence of this transient are less severe than those calculated.

15.4.5.3.4 Considerations of Uncertainties

Void reactivity characteristics, scram time, and worth are expected to be more favorable and, therefore, lead to reducing the actual severity from that which is discussed above.

15.4.5.4 Barrier Performance

15.4.5.4.1 Fast Opening of One Recirculation Valve

This transient results in a very slight increase in reactor vessel pressure as shown on Figure 15.4-7 and, therefore, represents no threat to the RCPB.

15.4.5.4.2 Fast Opening of Two Recirculation Valves

This transient results in a very slight increase in reactor vessel pressure as shown on Figure 15.4-8 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. This is a pressurized water reactor (PWR) event.

15.4.7 Misplaced Bundle Accident

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power), and was not reanalyzed for rated 3,467 MWt operation. Analysis of the mislocated bundle accident is performed for reload cores where the resultant CPR response may establish the operating limit MCPR.

The necessity to perform this analysis is determined on a cycle-specific basis; therefore, it was not analyzed for EPU (3,988 MWt).

15.4.7.1 Identification of Causes and Frequency Classification

15.4.7.1.1 Identification of Causes

This event is the improper loading of a fuel bundle and subsequent operation of the core. Three errors must occur for this event to take place in the initial core loading. First, a bundle must be loaded into a wrong location in the core. Second, the bundle that should have been loaded where the mislocation occurred would have to be overlooked and also put into 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.

The above number is based upon past experience. The only misloading events that have occurred in the past were in reload cores where only two errors are necessary. Therefore, the frequency of occurrence for initial cores is even lower since three errors must occur concurrently.

15.4.7.2 Sequence of Events and 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; therefore, no corrective Operator action or automatic protection system functioning occurs.

This analysis represents the worst case (i.e., operation of a misplaced bundle with three SEFs or SOEs).

15.4.7.3 Core and System Performance

This event is discussed in Section S.2.2.1.8 of GESTAR II[2].

15.4.7.3.1 Analysis

The initial core consists of three bundle types with average enrichments that are high, medium, or low with correspondingly different gadolinia concentrations. The fuel bundle loading error involves interchanging a bundle of one enrichment with another bundle of a different enrichment. The following fuel loading errors are possible in an initial core.

1. High-enrichment bundle misloaded into low-enrichment bundle location.

2. Medium-enrichment bundle misloaded into low-enrichment bundle location.

3. Low-enrichment bundle misloaded into high-enrichment bundle location.

4. Low-enrichment bundle misloaded into medium-enrichment bundle location.

5. Medium-enrichment bundle misloaded into high-enrichment bundle location.

6. High-enrichment bundle misloaded into medium-enrichment bundle location.

Since all low-enrichment bundles are located on the core periphery, the misloading of high- or medium-enrichment bundles into a low-enrichment bundle location, misloading errors 1 or 2, is not significant. In these cases, the higher reactivity bundles are moved to a region of low reactivity and power resulting in an overall improvement in performance and no impact on thermal margin.

The third type (3) of fuel loading error results in the largest enrichment mismatch. However, it does not result in an unacceptable operating consequence. Consider a fuel bundle loading error at beginning-of-cycle (BOC) with the low-enrichment bundle interchanged with a high-enrichment bundle located adjacent to the LPRM and predicted to have the highest LHGR and/or lowest CPR in the core. After the loading error has occurred and has gone undetected, assume, for purposes of conservatism, the Operator uses a control pattern that places the limiting bundle in the four-bundle array containing the misplaced bundle on thermal limits as recorded by the LPRM. As a result of loading the low-enrichment bundle in an improper location, the average power of the four bundles decreases. Normally, the reading of the LPRM will show a decrease in thermal flux due to

the decreased power; however, in this case an increase in the thermal flux occurs due to decreased neutron absorption in the low-enrichment bundle. The effect of the decreased thermal absorption is larger than the effect of power depression resulting in a net increase in the instrument reading. Thus, detected reductions in thermal margins during power operations will indicate a fuel loading error of this kind.

The fourth and fifth types (4 and 5) of fuel loading errors are similar to the third type and also result in conservative operating errors.

The fuel bundle loading error with greatest impact on thermal margin is of the sixth type (6), which occurs when a high-enrichment bundle is interchanged with a medium-enrichment bundle located away from a LPRM. Since the medium- and high-enrichment bundles have corresponding medium and high gadolinia contents, the maximum reactivity difference occurs at the EOC when the gadolinia has burned out. If the loading errors were made and have gone undetected, the Operator would assume that the mislocated bundle would operate at the same power as the instrumented bundle in the mirror-image location and would operate the plant until EOC. For the purpose of conservatism, it is assumed the mirror-image bundle is on thermal limits as recorded by the LPRM. As a result of placing the instrumented bundle on limits, the mislocated bundle violates the Technical Specification operating MCPR limit.

A summary of input parameters for the loading error type 6 analysis is listed in Table 15.4-7.

15.4.7.3.2 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 safety limit, the point where boiling transition would be expected to occur. Therefore, no violation of fuel design limits occurs as a result of this event.

15.4.7.4 Barrier Performance

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

15.4.7.5 Radiological Consequences

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

15.4.8 Spectrum of Rod Ejection Assemblies

Not applicable to BWRs. This is a PWR event.

The BWR has precluded this event by incorporating into its design mechanical equipment that restricts any movement of the CRD system assemblies. The CRD housing support assemblies are described in Chapter 4.

15.4.9 Control Rod Drop Accident

This event is described in Section S.2.2.3.1 of GESTAR II[2].

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power) and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

However, the post-accident radiological consequences have been recalculated based on operation at 102 percent of the uprated power corresponding to 120 percent of the original licensed thermal power level (4,067 MWt).

15.4.9.1 Evaluation of Results

The radiological evaluations are based on the assumed failure of 1154 GNF2 fuel rods. The number of rods that exceed the damage threshold is less than 1154 GNF2 for all plant operating conditions or core exposure, provided the peak enthalpy is less than the 170 cal/gm design limit. The results of the compliance check calculation (Table 15.4-9) indicate that the maximum incremental rod worth is well below the worth required to cause a control rod drop accident (CRDA) which would result in 280 cal/gm peak fuel enthalpy. The conclusion is that the 280 cal/gm design limit is not exceeded and the assumed failure of 1154 GNF2 pins for the radiological evaluation is conservative. The fraction of core damaged resulting from the CRDA involving GNF2 fuel rods bounds GE14.

15.4.9.2 Barrier Performance

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

15.4.9.3 Radiological Consequences

Two separate radiological analyses are provided for this accident:

One 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 10CFR50.67 criteria. This analysis is referred to as the design basis analysis. The second analysis is based on assumptions considered to provide a more realistic, but still conservative, estimate

of radiological consequences. This analysis is referred to as the realistic analysis.

15.4.9.3.1 Design Basis Analysis

The design basis analysis is based on an AST as described in NRC SRP 15.0.1[3] and RG 1.183. Specific parametric values used in the evaluation are presented in Table 15.4-10.

Fission Product Release from Fuel

The failure of 1154 GNF2 fuel rods is used for this analysis. Those fuel rods presumed to fail are assumed to have operated at power levels 1.8 times that of the average power level of the core. The mass fraction of the fuel in the damaged rods which reaches or exceeds the initiation temperature of fuel melting is assumed to be less than 0.0077. The release fractions for the different radionuclide groups from the gap and from melted fuel are listed in Table 15.4-10.

A maximum equilibrium inventory of fission products in the core is based on 1,400 days of continuous GE14 operation at 4,067 MWt. GNF2 fuel was evaluated for 1,315 days of operation and determined to be bounded by the GE14 core inventory from a consequence perspective. No delay time is considered between departure from the above power condition and the initiation of the accident.

Fission Product Transport to the Environment

Case 1 - Condenser Leakage

The transport pathway consists of carryover with steam to the turbine condenser prior to MSIV closure and leakage from the condenser to the environment. No credit is taken for the turbine building.

Of the activity released from the fuel, 100 percent of the noble gases, 10 percent of the iodines, and 1 percent of the remaining radionuclides are assumed to be carried to the condenser before MSIV closure is complete. Of the activity reaching the condenser, 100 percent of the noble gases, 10 percent of the iodines, and 1 percent of the remaining 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/day for the design basis calculation. Radioactive decay is neglected during residence in the turbine and condensers and after release to the environment.

Case 2 - Mechanical Vacuum Pump

This case addresses the condition where the mechanical vacuum pump is operating and the activity in the steam line is below the vacuum pump isolation setpoint. The transport pathway consists

of carryover with steam to the turbine condenser prior to MSIV closure and then exhaust from the condenser through the vacuum pump to the main stack.

The assumptions concerning release fractions and depletion due to partitioning and plateout are the same as Case 1. The activity is exhausted from the condenser at a rate determined by the vacuum pump flow rate. Doses are calculated using the computer code RADTRAD[9]. Radioactive decay is accounted for during residence in the turbine and condenser; however, it is neglected after release to the environment.

Activity Release

The activity airborne in the turbine and condensers is presented in Table 15.4-11.

The cumulative release of activity to the environment is presented in Table 15.4-12.

Results

The calculated exposures from the design basis analysis are presented in Table 15.4-13 and are a small fraction of the criteria of 10CFR50.67. Control room doses are calculated without credit for the control room emergency ventilation system (CREVS) and are within the criteria of 10CFR50.67.

Result of Analysis for Removing MSIV Isolation

Potential pathways to the environment include the recirculation system sample valves, the mechanical vacuum pump, and the offgas (OFG) system. The following is an evaluation of each pathway and the potential impact on releases following a CRDA.

Recirculation System Sample Valves The SRP 15.4.9 design basis source term assumption of transporting the activity from the CRDA directly to the condenser is nonmechanistic, conservative, and encompasses smaller, more mechanistic pathways that may exist. These valves are opened only as part of a procedure for taking a sample. Any activity release from the sampling lines following a CRDA would be small in comparison to CRDA activity postulated to be in the condenser under the SRP 15.4.9 assumption. Additionally, the activity in the sample lines would not be released directly to the environment, but would be exhausted at the sampling station hood to the reactor building ventilation system, where it is monitored for any high radiation level. The relatively small amount of sample line activity is encompassed by the conservative amount of activity postulated to be in the condenser. Therefore, the potential release of CRDA activity from the sample valve pathway need not be considered in the dose analysis.

Mechanical Vacuum Pump (MVP) The MVPs maintain the condenser at a negative pressure during startup and shutdown. In the event of a CRDA, the main steam line radiation monitor isolates the MVP flow path to the main plant stack. With the MVPs isolated, there are no releases to the environment from this pathway. The release path would be from the condenser. Therefore, the Case 1 analysis addresses the offsite doses with the MVP isolated (i.e., directly from the condenser to atmosphere). Case 2 addresses the condition where the activity in the steam lines is below the setpoint of the main steam line monitor and the release path is through the MVP.

Offgas System The CRDA assumed the fission product transport pathway consists of carryover with steam to the turbine condenser prior to MSIV closure and leakage from the condenser to the environment. Of the activity released from the fuel, 100 percent of the noble gases and 10 percent of the halogens are assumed to be carried to the condenser before Group 1 valve isolation.

The main steam line drain valves discharge to the main condenser, as do the MSIVs. Since the exhaust from the drain path is minimal compared to the MSIVs, and since these valves do not involve any significant difference in fission product pathway from that of the MSIVs, they can be taken as one pathway to the condenser. Therefore, the radiological contribution of the main steam line drain valve pathways to the condenser is already taken into account.

Assuming there is enough steam pressure available, the SJAE will draw the airborne gaseous activity from the condenser. The OFG system is designed to process the airborne activity before it is released to the environment through the main plant stack.

The following system configurations were considered for the offsite dose evaluation:

1. Offgas flow through the charcoal filter bypass line with the pretreatment monitor delay pipe isolated.

   At low power and during the power ascension, the OFG system charcoal beds may be bypassed, resulting in a direct release to the environment from the condenser through the main plant stack. This pathway is monitored for high radiation by the pretreatment monitor, which will isolate the pathway on high activity. The offgas pretreatment radiation monitor delay pipe is isolated during this mode of operation. The plant operating procedure administratively controls this. The pretreatment radiation monitor delay pipe is valved in only when the offgas charcoal beds are in service. With this pathway, there is no holdup of noble gases or adsorption of iodines. If there is enough steam, the SJAE releases the activity airborne

in the condenser to the environment through the main plant stack following a CRDA. Since the pretreatment radiation monitor is required by the Technical Specifications, and its use for isolating the pathway on high radiation has been reviewed and accepted by the NRC, a release through this pathway after a CRDA would be limited to the time required to isolate the pathway. The quantity released in this time frame is small when compared to the activity released from the condenser using SRP 15.4.9 assumptions, and the dose contribution would not affect the results of the design basis analysis.

2. Offgas flow through the charcoal beds with the pretreatment monitor delay pipe in service.

   During this process, noble gases are delayed for a significant period of time, and iodine is adsorbed on the charcoal beds for the duration of the accident. Due to the noble gas delay and the iodine adsorption on the offgas charcoal beds, doses resulting from releases through the OFG system are less than releases due to leakage from the condenser, as assumed in the DBA.

   This pathway is monitored for high radiation by the pretreatment monitor, which will isolate the pathway on high activity. Since the pretreatment monitor is required by the Technical Specifications, and its use for isolating the pathway on high radiation has been reviewed and accepted by the NRC, a release through this pathway after a CRDA would be limited to the time required to isolate the pathway. The quantity released in this time frame is small when compared to the activity released from the condenser using SRP 15.4.9 assumptions, and the dose contributions would be small. Therefore, the pathways to the environment, other than condenser leakage, are not significant contributors to the dose following a CRDA.

15.4.9.3.2 Realistic Analysis

The realistic CRDA analysis is provided to illustrate the conservatism of the design basis analysis. The realistic analysis is presented here as it appeared in Amendment 28 of the Final Safety Analysis Report (FSAR), and will not be updated.

The realistic analysis is based on a realistic but still conservative assessment of this accident. The specific models, assumptions, DRAGON, and the program used for computer evaluation are described in Reference 4. Specific values of parameters used in the evaluation are presented in Table 15.4-10.

<u>Fission Product Release from Fuel</u>

The following assumptions are used in calculating the fission product activity released from the fuel:

1. The reactor has been operating at design power for 1,000 days until 30 min prior to the accident. When translated into actual plant operation, this assumption means that the reactor was shut down from design power, taken critical, and brought to the initial temperature conditions within 30 min of the departure from design power. The 30-min time represents a conservative estimate of the shortest period in which the required plant changes could be accomplished and defines the decay time to be applied to the fission product inventory calculations.

2. The failure of 770 fuel rods is used for this analysis. Those fuel rods presumed to fail are assumed to have operated at power levels 1.5 times that of the core average power level. The mass fraction of the fuel in the damaged rods which reaches or exceeds the initiation temperature of fuel melting is estimated to be 0.0077.

3. Fuel that reaches melt conditions is assumed to release 100 percent of the noble gas inventory and 50 percent of the halogen inventory.

4. An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity in a failed fuel rod is assumed to be released. These percentages are consistent with actual measurements made during defective fuel experiments(5).

Fission Product Transport to the Environment

The following assumptions are used in calculating the amount of fission product activity transported from the reactor vessel to the main condenser:

1. The recirculation flow rate is 25 percent of rated, and the steam flow to the condenser is 5 percent of rated. The 25-percent recirculation flow and 5-percent steam flow are the maximum flow rates compatible with the maximum fuel damage. The 5-percent steam flow rate is greater than that which would be in effect at the reactor power level assumed in the initial conditions for the accident. This assumption is conservative because it results in the transport of more fission products through the steam lines than would be expected. Because of the relatively long fuel-to-coolant heat transfer time constant, steam flow is not significantly affected by the increased core heat generation within the time required for the MSIVs to achieve full closure.

2. The MSIVs are assumed to receive an automatic closure signal 0.5 sec after detection of high radiation in the main steam lines and to be fully closed at 5 sec from the receipt of the closure signal. The signal originates from the main steam line radiation monitors. The total amount of fission product activity transported to the condenser before the steam lines are isolated is, therefore, governed by the 5.5-sec isolation time and the conditions in Item 1 above.

3. All of the noble gas activity is assumed to be released to the steam space of the reactor vessel.

4. The mass ratio of the halogen concentration in steam to that of the water is assumed to be 2 percent.

5. Fission product plateout is neglected in the reactor vessel, main steam lines, turbine, and condenser.

Of those fission products released from the fuel and transferred to the condenser, it is assumed that 100 percent of the noble gases are airborne in the condenser. The halogen activity airborne in the condenser is a function of the partition factor, volume of air, and volume of water. The partition factor assumed applicable is 100, while the ratio of air volume to water volume is taken as 5.0. The activity airborne in the condenser, based on the preceding conditions, is presented in Table 15.4-14.

The following assumption and condition is used to evaluate the activity released to the environment:

> The leak rate out of the condenser is 0.5 percent of the combined condenser and turbine free volume (670 cu ft) per day.

The integrated fission product release to the environment, based on the preceding assumption, is presented in Table 15.4-15.

<u>Results</u>

The calculated offsite exposures for the realistic analysis are presented in Table 15.4-16 and demonstrate the margin of conservatism in the design basis analysis.

15.4.10 References

1. Paone, C. J. Banked Position Withdrawal Sequence. NEDO-21231, September 1976.

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

3. USNRC Standard Review Plan, NUREG-0800, Washington, DC, July 2000.

4. DRAGON Computer Code, Dose and Radioactivity from Nuclear Facility Gaseous Outflows, NU-115, Version 5, Level 0.

5. Horton, N. R.; Williams, W. A.; and Holtzclaw, K. W. Analytical Methods for Evaluating the Radiological Aspects of General Electric Boiling Water Reactors. APED-5756, March 1969.

6. Reactivity Control Data Book, GE Document No. 383HA587AI, Revision 1.

7. Stien, R. C. and Klapproth, J. F. Continuous Control Rod Withdrawal Transient in the Startup Range. NEDM-23842, Class 1, April 1978.

8. Licensing Topical Report, Power Uprate Licensing Evaluation for Nine Mile Point Nuclear Power Station, Unit 2, NEDC-31994P, Revision 1, May 1993.

9. Arcieri, W. C. RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation, NUREG/CR-6604, Supplement 2, October 2002.

TABLE 15.4-1

SEQUENCE OF EVENTS - ROD WITHDRAWAL ERROR IN POWER RANGE

Note: These results are for Cycle 1. Cycle-specific results are presented in Appendix A.

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

TABLE 15.4-2

INPUT PARAMETERS AND INITIAL CONDITIONS
FOR ROD WITHDRAWAL TRANSIENT

Note: These results are for Cycle 1. Cycle-specific results are presented in Appendix A.

Neutron power, MWt	3323
Average core exposure, MWd/t	6000
Xenon state	None
Average linear heat generation rate, kW/ft	5.39
Maximum linear heat generation rate, kW/ft	13.48
Location of maximum LHGR bundle	21-38
Minimum CPR	1.24
Location of minimum CPR bundle	21-38
Maximum worth control rod	26-35
Rod withdrawal speed, in/sec	3.6
Core coolant flow rate million/hr x 10^6	108.5
Core coolant inlet enthalpy, Btu/lb	527.6
Core average steam volume fraction	0.371
Reactor coolant pressure, average, psia	1035
Control rod pattern	Figure 15.4-1
RBM trip setpoint, %	106

TABLE 15.4-3

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

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	Start pump motor.
2.3	Jet pump diffuser flows on started pump side become positive.
3.9	Pump motor at full speed and drive flow at about 30% of rated.
14 (est)	Last of cold water leaves recirculation drive loop.
14.6	Peak value of core inlet subcooling.
15.8	Peak thermal power. Estimated APRM thermal power approximately 4% below the APRM thermal power setpoint.
50.0+	Reactor variables settle into new steady state.

TABLE 15.4-4

SEQUENCE OF EVENTS FOR FAST OPENING OF ONE RECIRCULATION VALVE (FIGURE 15.4-7)

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	Simulate failure of single loop control.
1.1	Reactor APRM high flux scram trip initiated.
4.6	TCVs start to close upon falling turbine pressure.
18	Feedwater decreases upon rising water level.
26.0	TCVs closed; turbine pressure below pressure regulator setpoints.
50+	Reactor variables settle into new steady state.

TABLE 15.4-5

SEQUENCE OF EVENTS FOR FAST OPENING OF TWO RECIRCULATION VALVES (FIGURE 15.4-8)

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	Initiate failure of master controller.
1.2	Reactor APRM high flux scram trip initiated.
4.5	TCVs start to close upon falling turbine pressure.
21	Feedwater decreases upon rising water level.
29	TCVs closed; turbine pressure below pressure regulator setpoints.
50+	Reactor variables settle into new steady state.

TABLE 15.4-6

SEQUENCE OF EVENTS FOR MISPLACED BUNDLE ACCIDENT

Note: These results are for Cycle 1. Analysis of the mislocated bundle accident is performed for reload cores where the resultant CPR response may establish the operating limit MCPR.

1. During core loading operation, a 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, the two errors are 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 - CYCLE 1

Input Parameters	Initial Conditions
1. Power, % rated	100
2. Flow, % rated	100
3. MCPR operating limit	1.24
4. MLHGR operating limit, kW/ft	13.4
5. Average core exposure, MWD/t	EOC
6. Control rod pattern	All rods out

NOTES:

1. Core conditions are assumed to be normal for a hot operating core at EOC.
2. Analysis of the mislocated bundle accident is performed for reload cores where the resultant CPR response may establish the operating limit MCPR.

TABLE 15.4-8

RESULTS OF FUEL LOADING ERROR ANALYSIS - CYCLE 1

Note: Analysis of the mislocated bundle accident is performed for reload cores where the resultant CPR response may establish the operating limit MCPR.

1.	MCPR limit	1.24
2.	MCPR with misplaced bundle	1.13
3.	ΔCPR for event	0.11
4.	ΔLHGR limit, kW/ft	13.4
5.	LHGR with misplaced bundle, kW/ft	14.7
6.	LHGR for event, kW/ft	1.3

TABLE 15.4-9

INCREMENT WORTH OF THE MOST REACTIVE ROD USING A BANK POSITION WITHDRAWAL SEQUENCE - CYCLE 1[1]

Core Condition (MWD/T)	Control Rod Group[2]	Banked at Notch	Control Rod (I, J)	Drops From-To	Increase in K_{eff}
BOC-1 Sequence A G1 through G4 W/D. All others at position 0, except Group 7	7	12	(26-35)	0-48	0.004658

[1] The following assumptions were made to ensure that the rod worths were conservatively high for the BPWS:
a. BOC
b. Hot startup
c. No xenon

[2] For definition of rod groups, see Reference 6.

TABLE 15.4-10

CONTROL ROD DROP ACCIDENT EVALUATION PARAMETERS

	Design Basis Assumptions		Realistic Basis Assumptions
1. Data and assumptions used to estimate radioactive source from postulated accidents			
a. Power level	4,067 MWt		3,489 MWt
b. Fuel damaged	1154 GNF2 rods		770 rods
c. Release of activity by nuclide	Table 15.4-12		Table 15.4-15
d. Peaking factor	1.8		1.5
2. Data and assumptions used to estimate activity released			
a. Condenser leak rate, %/day	1.0 (Case 1)		0.5
b. Valve closure time, sec	N/A		5.5
c. All other pertinent data and assumptions			
(1) Control building volume	3.81+05 ft^3		4.8+05 ft^3
(2) Control building air intake rate	N/A		1500 cfm
(3) Control building recirculation rate	N/A		750 cfm
(4) Control building filter efficiencies	N/A		99%
(5) Condenser volume	97,000 ft^3		N/A
(6) Vacuum pump flow rate	5,000 cfm (Case 2)		N/A
3. Dispersion data			
a. X/Qs (sec/m^3) for time intervals of			
	Case 1 Ground	Case 2 Stack	
(1) 0-2 hr - EAB	1.19-04	2.96-05	2.19-05
(2) 0-8 hr - LPZ	1.62-05	1.42-05	6.48-06
(3) 8-24 hr - LPZ	N/A	5.41-07	4.58-06
(4) 0-2 hr - control room	1.47-03	8.03-05	2.13-04
(5) 2-8 hr - control room	N/A	4.48-05	2.13-04
(6) 8-24 hr - control room	N/A	1.68-05	1.66-04
b. Breathing rates (m^3/sec)			
(1) 0-2 hr EAB and 0-8 hr LPZ	3.5-04		3.47-04
(2) 8-24 hr EAB and LPZ	1.8-04		1.75-04
(3) Control room	3.5-04		3.47-04

TABLE 15.4-10 (Cont'd.)

	Design Basis Assumptions	Realistic Basis Assumptions
4. Release Fractions	Gap / Melted Fuel	
a. Noble gas	10% / 100%	
b. Iodine	10% / 50%	
c. Br	5% / 30%	
d. Cs, Rb	12% / 25%	
5. Dose data	Table 15.4-13	Table 15.4-16

NOTE: 3.81+05 = 3.81×10^5

TABLE 15.4-11

CONTROL ROD DROP ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY AIRBORNE IN TURBINE AND CONDENSER (CURIES)

	Total Curies	
Isotope	Case 1	Case 2
Kr-83m	4.57E+04	9.51E+02
Kr-85m	9.51E+04	2.67E+03
Kr-85	5.48E+03	1.89E+02
Kr-87	1.81E+05	3.03E+03
Kr-88	2.55E+05	6.36E+03
Kr-89	3.10E+05	3.02E-04
Xe-131m	4.23E+03	1.46E+02
Xe-133m	2.27E+04	7.70E+02
Xe-133	7.35E+05	2.52E+04
Xe-135m	1.52E+05	1.40E+02
Xe-135	2.67E+05	8.30E+03
Xe-137	6.69E+05	1.62E-02
Xe-138	6.30E+05	8.34E+02
I-131	3.65E+04	1.25E+03
I-132	5.28E+04	1.22E+03
I-133	7.41E+04	2.44E+03
I-134	8.11E+04	9.75E+02
I-135	6.94E+04	2.08E+03
Rb-86	1.15E+01	3.96E-01
Cs-134	1.15E+03	3.96E+01
Cs-136	3.61E+02	1.24E+01
Cs-137	6.89E+02	2.37E+01

NOTE: 4.57E+04 = $4.57 \times 10^{+4}$

TABLE 15.4-12

CONTROL ROD DROP ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT (CURIES)

Isotope	Total Curies Case 1	Total Curies Case 2
Kr-83m	4.57E+02	1.03E+04
Kr-85m	9.51E+02	4.12E+04
Kr-85	5.48E+01	5.29E+03
Kr-87	1.81E+03	2.93E+04
Kr-88	2.55E+03	8.15E+04
Kr-89	3.09E+03	5.60E+02
Xe-131m	4.23E+01	4.03E+03
Xe-133m	2.27E+02	2.01E+04
Xe-133	7.35E+03	6.83E+05
Xe-135m	1.52E+03	3.61E+03
Xe-135	2.67E+03	1.63E+05
Xe-137	6.69E+03	1.75E+03
Xe-138	6.30E+03	1.73E+04
I-131	3.65E+01	3.42E+03
I-132	5.28E+01	1.43E+03
I-133	7.41E+01	5.76E+03
I-134	8.11E+01	8.97E+02
I-135	6.94E+01	3.69E+03
Rb-86	1.15E-03	1.10E-01
Cs-134	1.15E-01	1.11E+01
Cs-136	3.61E-02	3.42E+00
Cs-137	6.88E-02	6.62E+00

NOTE: 4.57E+02 = $4.57 \times 10^{+2}$

TABLE 15.4-13

CONTROL ROD DROP ACCIDENT (DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

	Total Effective Dose Equivalent (Rem)	
	Case 1	Case 2
Exclusion area (2 hr)	6.17-01	1.12+00
Low population zone (24 hr)	8.39-02	1.27+00
Main control room*	1.37+00	2.51+00

NOTE: 5.7-01 = 5.7×10^{-1}

* Control room doses listed are the maximum values calculated for time periods for and beyond the release duration of 24 hr.

TABLE 15.4-14

CONTROL ROD DROP ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN CONDENSER (CURIES)

Isotope	Total Curies
Kr-83m	3.39+02
Kr-85m	6.88+02
Kr-85	3.29+01
Kr-87	1.08+03
Kr-88	1.77+03
Kr-89	3.48+00
Xe-131m	1.72+01
Xe-133m	2.51+02
Xe-133	6.03+03
Xe-135m	8.92+02
Xe-135	9.62+02
Xe-137	2.49+01
Xe-138	1.16+03
I-129	2.61-08
I-131	7.90-01
I-132	1.15+00
I-133	1.64+00
I-134	1.58+00
I-135	1.48+00
I-136	2.63-07
Br-83	8.45-02
Br-84	9.60-02
Br-85	1.68-04
Br-87	7.00-11

NOTE: 3.39+02 = 3.39×10^{2}

TABLE 15.4-15

CONTROL ROD DROP ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT

Isotope	Total Curies Released
I-129	8.67-11
I-131	2.48-03
I-132	6.65-05
I-133	3.19-03
I-134	6.87-07
I-135	1.04-03
I-136	0.0
Br-83	5.98-06
Br-84	4.40-10
Br-85	0.0
Br-87	0.0
Kr-83m	1.01-02
Kr-85m	2.36-01
Kr-85	1.09-01
Kr-87	4.97-03
Kr-88	2.01-01
Kr-89	0.0
Xe-131m	5.50-02
Xe-133m	6.81-01
Xe-133	1.84+01
Xe-135m	6.54-03
Xe-135	1.07+00
Xe-137	0.0
Xe-138	5.59-12

NOTE: 8.67-11 = 8.67×10^{-11}

TABLE 15.4-16

CONTROL ROD DROP ACCIDENT (REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole-Body Dose (Rem)	Thyroid Dose (Rem)	Beta Dose (Rem)
Exclusion area (2 hr)	1.00-05	6.42-06	4.99-06
Low population zone (24 hr)	9.13-06	1.11-05	6.97-06
Main control room (24 hr)	8.27-06	2.94-06	1.63-04

15.5 INCREASE IN REACTOR COOLANT INVENTORY

15.5.1 Inadvertent HPCS Startup

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power) and was not reanalyzed for rated 3,467 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

This event is non-limiting; therefore, it was not evaluated for EPU (3,988 MWt).

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.

Regardless of the mode of recirculation system operation, relatively small changes would be experienced in plant conditions. The analyzed event requires no Operator action. Subsequent to the event, however, the Operator would terminate HPCS flow to the reactor vessel.

15.5.1.2.2 System 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 those described is not expected for this transient event.

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

15.5.1.2.3 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 aggravates the transient depending upon the nature of the failure. Pressure regulator failures are discussed in Sections 15.1.3 and 15.2.1.

The effect of a single failure in the level control system has 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. This trip response is described in the failure of feedwater controller with increasing flow (Section 15.1). Decreasing level automatically initiates a scram at the L3 level trip and has a response similar to loss of feedwater control - decreasing flow.

15.5.1.3 Core and System Performance

15.5.1.3.1 Mathematical Model

The detailed nonlinear dynamic model (Section 15.1.1.3.1) is used to simulate this transient.

15.5.1.3.2 Input Parameter and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-3. 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 recirculation flow control mode. It begins with the introduction of cold water into the upper core plenum. Within 1.22 sec the full HPCS flow is established at approximately 7.6 percent of the rated feedwater flow rate. This flow is nearly 137 percent the HPCS flow at rated pressure. No delays were considered because they are not relevant to the analysis.

Addition of cooler water to the upper plenum causes a reduction in steam flow, which results in some depressurization as the pressure regulator responds to the event. In the automatic flow control mode, following a momentary decrease, neutron power stabilizes out at a level slightly above operating level. In manual mode, the flux level stabilizes out slightly below operating level. In either case, pressure and thermal variations

are relatively small, and no significant consequences are experienced. MCPR remains unchanged from the initial operating value; therefore, fuel thermal margins are maintained.

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

15.5.1.4 Barrier Performance

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

15.5.1.5 Radiological Consequences

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

15.5.2 Chemical Volume Control System Malfunction (or Operator Error)

This section is not applicable to BWRs. This is of PWR interest.

15.5.3 BWR Transients That Increase Reactor Coolant Inventory

These events are discussed in Sections 15.1 and 15.2.

TABLE 15.5-1

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

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	Simulate HPCS cold water injection into the reactor vessel.
1.2	Full flow established for HPCS.
7	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 or outside the primary containment, but within a controlled release structure. 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 able to be isolated and where detection is not automatic or apparent. This event is less limiting than the postulated events in Sections 15.6.4, 15.6.5, and 15.6.6.

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

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power) and was not reanalyzed for EPU operation (3,988 MWt). This event does not set reactor operating limits and is not reanalyzed for each reload cycle. The post-accident radiological consequences have been considered at 3,988 MWt.

15.6.2.1 Identification of Causes and Frequency Classification

15.6.2.1.1 Identification of Causes

There is no identified specific event or circumstance 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.

<u>Event Description</u>

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 reactor building. This event could conceivably occur also in the drywell. However, the associated effects would not be as significant as those from a failure in the reactor building.

15.6.2.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.2.2 Sequence of Events and Operator Actions

Identification of Operator Actions

Termination of the analyzed event is dependent on Operator action. The action is initiated with the discovery of the unisolatable leak. The action consists of verification of secondary containment isolation and SGTS initiation, and the orderly shutdown and depressurization of the reactor vessel.

Discovery of the leak will be the result of noticeable increases in radiation, temperature, humidity, or noise levels in the secondary containment or abnormal indications of actuations caused by the affected instrument.

15.6.2.3 Core and System Performance

Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint than the events examined in Sections 15.6.4, 15.6.5, and 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.

No fuel damage or core uncovering occurs as a result of this accident. Similarly, instrument line breaks are within the spectrum considered in emergency core cooling system (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.

15.6.2.4 Barrier Performance

15.6.2.4.1 General

The following assumptions and conditions are the basis for the mass loss during the 2-hr release period of this event:

1. The instrument line releases coolant at normal operating temperature and pressure to the reactor building.

2. The coolant is released at a constant rate for 2 hr.

3. The reactor remains at full power for the 2-hr duration of the event.

4. The instrument line contains a 1/4-in diameter flow restricting orifice inside the drywell.

5. The Moody critical blowdown flow model is applicable, and flow is critical at the orifice[1].

The total integrated mass of fluid released into the reactor building via the break is 19,800 lb. Of this total, 7,583 lb flash to steam.

15.6.2.4.2 Containment Effects

The energy released by this coolant loss would cause the reactor building's siding pressure to be exceeded. No benefit from the secondary containment or the standby gas treatment system is to be assumed.

15.6.2.5 Radiological Consequences

15.6.2.5.1 Design Basis Analysis

While the NRC has developed a standard review plan for this event, a specific regulatory guide calculation method has not been issued to specify unique design basis assumptions. For this reason, only the realistic bases will be provided.

15.6.2.5.2 Realistic Analysis

The realistic analysis is based on a realistic, but still conservative, assessment of this accident. Specific values of parameters used in the evaluation are presented in Table 15.6-1.

Several Chapter 15 accident sections include both a design basis and a realistic basis analysis. The realistic basis analysis for such SAR sections is not updated beyond FSAR Amendment 28. However, because the only instrument line break analysis presented is a realistic analysis, it is and will continue to be updated as necessary.

Fission Product Release from Fuel

The iodine activity in the coolant is assumed to be at the maximum Technical Specification limit for continued operation. No iodine spike occurs because the reactor remains at normal temperature and pressure throughout the accident.

All of the iodine in the released coolant that flashes to steam becomes airborne and available for release to the environment. All other activity which may become airborne is negligible when compared to the iodine concentration. Table 15.6-2 lists the iodine activities before the accident.

Fission Product Release to the Environment

No credit is taken for operation of the SGTS, and an instantaneous, ground level release is assumed. Table 15.6-3 presents the activity released to the environment.

Results

The calculated exposures for the realistic analysis are presented in Table 15.6-4, and are a small fraction of the guidelines of 10CFR100. Control room doses for exposures for and beyond the 2-hr release duration are a small fraction of the GDC 19 limit.

15.6.3 Steam Generator Tube Failure

This section is not applicable to the direct-cycle BWR, but is a PWR-related event.

15.6.4 Steam System Piping Break Outside Containment

This event involves the postulation of a main steam line pipe break outside primary containment. It is assumed that the largest steam line instantaneously and circumferentially breaks at a location downstream of the outside isolation valve. The plant is designed to detect such an occurrence immediately, initiate isolation of the main steam lines, and actuate the necessary protective features. This postulated event represents the envelope evaluation of steam line failures outside primary containment.

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of original rated power) and was not reanalyzed for uprated 3,988 MWt operation. This event does not set reactor operating limits and is not reanalyzed for each reload cycle. The post-accident radiological consequences have been considered at 3,988 MWt.

15.6.4.1 Identification of Causes and Frequency Classification

15.6.4.1.1 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 primary containment are the result of postulated breaches in the RCPB or the steam power conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault

event for breaks outside the containment is a complete severance of one of the four main steam lines. The sequence of events and approximate time are given in Table 15.6-5.

The Operator should verify automatic functions and monitor all parameters.

No Operator action is required.

15.6.4.2.2 Systems Operation

A postulated guillotine break of one of the four main steam lines outside the primary 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 inside isolation valve. Flow from the downstream side is initially limited by the total area of flow restrictors in three unbroken lines. Subsequent closure of the MSIVs further limits the flow when the valve area becomes less than the limiter area and finally terminates the mass loss when full closure is reached.

A discussion of plant and RPS action and ESF action is given in Sections 6.3, 7.3, and 7.6.

15.6.4.2.3 Effect of Single Failures and Operator Errors

The effect of single failures has been considered in analyzing this event. The ECCS aspects are covered in Section 6.3. The break detection and isolation considerations are defined in Sections 7.3 and 7.6. All the protective sequences for this event are capable of SCF and SOE accommodation with completion of the necessary safety action (refer to Appendix 15A for further details).

15.6.4.3 Core and System Performance

Quantitative results (including math models, input parameters, and consideration of uncertainties) for this event are given in Section 6.3. 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 Considerations of Uncertainties

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

15.6.4.4 Barrier Performance

Since this break occurs outside the containment, barrier performance within the containment envelope is not applicable. Details of the results of this event can be found in Section 6.2.3.2.3 (Bypass Leakage Paths). Release of fission products to the environment is discussed in Section 15.6.4.5.

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

1. The reactor is operating at full power.

2. Nuclear steam supply system (NSSS) pressure remains constant during closure.

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

4. Isolation valves start to close at 0.5 sec on high flow signal and are fully closed at 5.5 sec.

5. The Moody critical flow model is applicable[1].

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

The mass released from the break is equal to the mass of steam in the broken line and connecting lines, plus the steam and coolant that passes through the MSIVs prior to closure. Rapid depressurization of the reactor pressure vessel (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 4.8+07 gm, of which 4.1+07 gm is liquid and 7.1+06 gm is steam.

15.6.4.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

1. 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 10CFR50.67 as the design basis analysis.

2. The second is based on assumptions considered to provide a realistic, but still conservative, estimate of the radiological consequences. This analysis is referred to as the realistic analysis.

15.6.4.5.1 Design Basis Analysis

The design basis analysis is based on an alternative source term (AST) as described in NRC SRP 15.0.1 and RG 1.183. Specific values of parameters used in the evaluation are presented in Table 15.6-6.

15.6.4.5.2 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. This level of activity is consistent with an offgas release rate of 350,000 uCi/sec after 30-min delay.

The concentrations in the reactor coolant are presented in Table 15.6-7, where the iodine has been normalized to the maximum Technical Specification limit of 4 uCi/gm I-131 dose equivalent.

Because of its short half-life, N-16 is not considered in the analysis.

15.6.4.5.3 Fission Product Transport to the Environment

The MSIV detection and closure time of 5.5 sec results in a discharge of 7.1+06 gm of steam and 4.1+07 gm of liquid from the break. This release of steam causes the main steam tunnel blowout panels to open, resulting in an unfiltered, instantaneous ground level release to the environment. Assuming all the activity in this discharge becomes airborne, the release of activity to the environment is presented in Table 15.6-8.

15.6.4.5.4 Results

The calculated exposures for the design basis analysis are presented in Table 15.6-9. The exclusion area boundary (EAB) and low population zone (LPZ) doses are a small fraction of the criteria of 10CFR50.67. Control room doses calculated assuming a puff release with no credit for the control room emergency ventilation system (CREVS) are within the criteria of 10CFR50.67.

15.6.4.5.5 Realistic Analysis

The realistic main steam line break accident analysis is provided to illustrate the conservatism of the design basis analysis. The realistic analysis is presented here as it appeared in Amendment 28 of the FSAR, and will not be updated.

The realistic analysis is based on a realistic, but still conservative, assessment of this accident. Specific values of parameters used in the evaluation are presented in Table 15.6-6.

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 associated with the steam and liquid discharged from the break.

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 release path to the environment is an instantaneous ground level release via the main steam tunnel blowout panels.

The following assumptions are used in calculating quantity and types of radioactive material released from the RCPB.

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

2. The expected concentrations contained in the primary coolant are presented in Table 15.6-10. This level of activity is consistent with an offgas release rate of 50,000 uCi/sec after a 30-min delay.

   Measurements made on current generation BWRs show that the activity ratio between the main turbine condensate and reactor coolant is approximately 0.5 percent to 2 percent. For the purpose of this evaluation, it is conservatively assumed that the activity per pound of steam is equal to 2 percent of the activity per pound of reactor water.

3. The noble gas concentration in the steam and coolant is assumed to be that of the expected value in the steam.

4. Because of the short half-life of N-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 available for atmospheric dispersion is presented in Table 15.6-11.

15.6.4.5.6 Results

Calculated exposures for this event are presented in Table 15.6-12. As noted, these values are a small fraction of 10CFR100.

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

This event involves the postulation of a spectrum of piping breaks inside containment varying in size, type, and location. The break type includes steam and/or liquid process system lines. This event is also assumed to be coincident with a safe shutdown earthquake (SSE). The occurrence of this event while SGTS is operating in the pressure control mode is also postulated.

The event has been analyzed quantitatively in Sections 6.2, 6.3, 7.1, 7.3, and 8.3. Therefore, the following discussion provides 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.

The LOCA analysis was performed at 3,988 MWt (120 percent of original rated power and up to 105 percent steam flow). The full spectrum LOCA analysis was performed for operation at 102 percent of the uprated power (4,068 MWt) using the GE SAFER/GESTR-LOCA methodology.[13] The results of the analysis confirmed that all LOCA-ECCS performance requirements are met. The results, as stated in 15.6.5.3, are given in detail in Section 6.3. The post-accident radiological consequences have also been considered at 3,988 MWt.

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 LOCA coincident with SSE plus SACF criteria requirements. The subject piping is designed to high quality and strict industry code and standard criteria and severe seismic and environmental conditions. However, since such an accident provides an upper limit estimate to the resultant effects for this category of pipe breaks, it is evaluated without the causes being identified.

15.6.5.1.2 Frequency Classification

This event is 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 core system performance, and Table 6.2-8 for barrier (containment) performance.

Following the pipe break and scram, the low-low water level signal initiates HPCS and RCIC systems, and high drywell pressure initiates HPCS system, at time 0, plus approximately 30 sec.

The low-low-low water level or high drywell pressure signal initiates both the low-pressure core spray (LPCS) and LPCI systems at time 0, plus approximately 40 sec, and low-low-low water level initiates MSIV closure at time 0.

Since automatic actuation and operation of the ECCS is a system design basis, no Operator actions are required for the accident.

15.6.5.2.2 Systems Operations

Accidents that could result in the release of radioactive fission products directly into the containment are the results of postulated nuclear system primary coolant pressure boundary pipe breaks. Possibilities for all pipe break sizes and locations are examined in Sections 6.2 and 6.3, including the severance of small process system lines, the main steam lines upstream of the flow restrictors, and the recirculation loop pipelines. The most severe nuclear system effects and the greatest release of radioactive material to the containment result from a complete circumferential break of one of the two recirculation loop pipelines. The minimum required functions of any reactor and plant protection system are discussed in Sections 6.2, 6.3, 7.3, 7.6, and 8.3, and Appendix 15A.

15.6.5.2.3 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 considerations for 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 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 Sections 6.3, 7.3, 7.6, 8.3, and Appendix 15A.

15.6.5.3.2 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the analysis of this event are given in Table 6.3-1.

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. Post-accident tracking instrumentation and control is assured. Continued long-term core cooling is demonstrated. Radiological input is minimized and within limits. Continued Operator control and surveillance is examined and guaranteed.

15.6.5.3.4 Consideration of Uncertainties

This event was conservatively analyzed (see Sections 6.3, 7.3, 7.6, 8.3, and Appendix 15A for details).

15.6.5.4 Barrier Performance

The design basis for the containment is to maintain its integrity, and experience acceptable stresses after the instantaneous rupture of the largest single primary system piping within the structure, while also accommodating the dynamic effects of the pipe break at the same time a SSE is also occurring. Therefore, any postulated LOCA does not result in exceeding the containment design limit. For details and results of the analyses, see Sections 3.8, 3.9, and 6.2.

15.6.5.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

1. The first analysis applies the isentropic flow model (see Section 6.2.3.2) to the design basis analysis. The results are used in determining the adequacy of the plant design to meet 10CFR50.67 criteria.

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

15.6.5.5.1 Design Basis Analyses

The methods, assumptions, and conditions used to evaluate this accident are in accordance with those guidelines for an AST as described in SRP 15.0.1 and RG 1.183.

Containment sprays are credited for reduction in airborne activity in containment. The effect of natural deposition of particulates inside containment is not considered. Specific values of parameters used in this evaluation are presented in Table 15.6-13.

15.6.5.5.2 Fission Product Release from Fuel

It is assumed that the GE14 fueled reactor is operating at a power level of 4,067 MWt for 1,400 days prior to the accident. GNF2 fuel was evaluated for 1,315 days of operation and determined to be bounded by the GE14 core inventory from a consequence perspective. The airborne source available for release from primary containment is released from the core in two phases: gap release, which occurs between 2 min and 32 min following reactor shutdown (0.5-hr duration), and early in-vessel release, which occurs between 32 min and 122 min (1.5-hr duration). The suppression pool source contains no noble gases and 50 percent of the core halogen inventory. While not specifically stated in RG 1.183 or SRP 15.0.1, the assumed release implies fuel damage approaching melt conditions. Even though this condition is inconsistent with operation of the ECCS system (Section 6.3), it is assumed applicable for the evaluation of this accident. The airborne activity available for release from the primary containment is presented in Table 15.6-14.

15.6.5.5.3 Fission Product Transport to the Environment

The transport pathways consist of leakage from the primary containment to the environment through several different mechanisms. Where applicable, the SGTS filter efficiency for halogen removal is assessed as 99 percent. The mechanisms for leakage from the primary containment are discussed in the following paragraphs:

1. Containment leakage - The Technical Specification leak rate of the primary containment and its penetrations (excluding the bypass leakage paths) is initially 1.1 percent per day. Additional primary containment leakage occurs through one of the TIPs at 0.12 percent per day of drywell volume into secondary containment. The leakage rate is reduced by a factor of two beginning one day following the LOCA. All this leakage is to the reactor building, and from there to the environment via SGTS. Credit is taken for 50-percent mixing within the reactor building.

2. Leakage from ESF components outside primary containment - 1 gpm total leakage of suppression pool fluid into the reactor building is assumed to occur for the duration of the accident. This leak rate is increased by a factor of two as per RG 1.183. An additional 60 gpm of ESF component leakage of suppression pool fluid into secondary containment is also considered. Ten

percent of the iodines in this leakage become airborne and available for release. The ESF leakage is released unfiltered and at ground level during the drawdown period. When secondary containment integrity is reestablished, the ESF leakage is released via the main stack, filtered by the SGTS.

3. Reactor building pressurization (containment leakage released during the drawdown period) - During the time when the pressure in the reactor building is greater than negative 1/4-in W.G. compared to the environment leakage from the primary containment, TIP, and ESF systems travels into secondary containment. Credit is taken for 50-percent mixing within the reactor building; however, no credit is taken for filtration.

4. SGTS operating in pressure control mode - Occurrence of a LOCA while the SGTS is operating in the pressure control mode would result in primary containment air being released via the SGTS filters and the main stack to the environment. This release through the 2-in pressure control line is terminated within 5 sec by closure of the primary containment purge isolation valves.

5. Bypass leakage - The piping paths listed below provide potential routes for post-LOCA primary containment atmosphere to bypass the reactor building and the SGTS and be released directly to the environment.

   a. Main steam lines (4)

   b. Feedwater lines (2)

   c. Post-accident sampling lines (4) (until such time as a modification eliminates the potential leakage paths)

   d. Main steam drain lines (2)

   e. RWCU line

   f. Drywell equipment drain and vent lines (2)

   g. Drywell floor drain and vent lines (2)

   h. Primary containment purge lines (4)

   i. Primary containment purge lines (2)

   Instrument air lines (3)

   Nitrogen inerting system line (1)

j. Instrument air lines to ADS valve accumulators (2)

Section 6.2.3 describes in detail the method used to determine the leak rates through the isolation valve(s) for each path. Since the isentropic rather than the isothermal flow process results in a higher leakage rate for each bypass path, the isentropic model defines the more conservative approach for use in the design basis analysis, and is the only analysis described in Section 15.6.5 of the FSAR.

Pre-release holdup based on slug-flow is credited for the main steam lines and certain bypass lines that originate in the drywell (feedwater lines, containment purge line and RWCU line). Delay is neglected for all other bypass lines.

Credit is taken for the deposition of elemental and particulate iodines on the walls of the piping between the two closed isolation valves for select bypass leakage paths (see Table 15.6-13). No credit is taken for deposition of organic iodine.

Data used to determine the radiological consequences of a design basis LOCA for isentropic conditions is presented in Table 15.6-13.

Fission product releases to the environment are given in Table 15.6-15b.

15.6.5.5.4 Results

The calculated exposures for the design basis analysis are presented in Table 15.6-16b and are within the criteria of 10CFR50.67.

15.6.5.5.5 Realistic Analysis

The realistic LOCA analysis is provided to illustrate the conservatism of the design basis analysis. The realistic analysis is presented here as it appeared in Amendment 28 of the FSAR, and will not be updated.

The realistic analysis is based on a realistic, but still conservative, assessment of this accident. Specific values of parameters used in the evaluation are presented in Table 15.6-13.

<u>Fission Product Release from Fuel</u>

Since this accident does not result in any fuel damage, the only activity released to the primary containment is that activity contained in the reactor coolant, plus any additional activity which may be released as a consequence of reactor scram and vessel depressurization.

While there are various activation and corrosion products contained in the reactor coolant, the products of primary importance are the iodine isotopes I-131 to I-135. The design reactor coolant iodine activities are normalized to the maximum Technical Specification[5] limits. The coolant concentrations and the normalized concentrations for these isotopes are presented in Table 15.6-17.

Considering that approximately 40 percent of the released liquid flashes to steam, it is conservatively assumed that 40 percent of the released iodine activity is airborne initially.

As a consequence of reactor scram and depressurization, additional iodine activity is released from those rods which experienced cladding perforation during normal operation. The reactor coolant iodine activities that are normalized to the maximum Technical Specification[4] limits are multiplied by 500 to account for iodine spiking[5].

While no measurements have been obtained during a pressure transient as rapid as the LOCA, it is difficult to predict the actual release rate from the fuel as a consequence of iodine spiking. It is, therefore, arbitrarily assumed that 100 percent of the spiking source term is released during the time period that 40 percent of the discharged coolant is flashing to steam. The initial airborne iodine inventories resulting are presented in Table 15.6-18.

Since it is also assumed that plateout and condensation remove 50 percent of the airborne iodine activity, the resultant activity airborne in the primary containment from spiking is presented in Table 15.6-18.

<u>Fission Product Transport to the Environment</u>

A large reactor coolant pipe fails and instantaneously releases the entire mass of coolant in the reactor vessel and recirculation system to the primary containment. The activity airborne in the primary containment and available for release leaks to the reactor building at a constant rate for 30 days. From 0 to 360 sec post-LOCA, the reactor building cannot maintain a pressure less than -1/4 in W.G. During this period, the activity leaking from the primary containment is assumed to be released directly to the environment. The SGTS begins operation within 30 sec after the LOCA signal. After 360 sec, the reactor building returns to a pressure less than -1/4 in W.G. and for the remaining 30 days the activity airborne in the reactor building is removed and filtered by the SGTS and exhausted through the main stack.

The leak rate from the primary containment to the reactor building is 1.1 percent/day, where 50 percent mixing is assumed to occur. Release from the reactor building to the environment

through a 99 percent iodine-efficient SGTS is at a rate of 3,500 cfm. The integrated isotopic activity released to the environment is presented in Table 15.6-19.

Results

The calculated radiological exposures for this event are presented in Table 15.6-20 and, as shown, are a small fraction of 10CFR100.

15.6.6 Feedwater Line Break - Outside Containment

In order to evaluate large liquid process line pipe breaks outside containment, the failure of a feedwater line is assumed to evaluate the response of the plant design to this postulated event. The postulated break of the feedwater line, representing the largest liquid line outside the containment, provides the envelope evaluation relative to this type of occurrence. The break is assumed to be instantaneous, circumferential, and downstream of the isolation valve.

A more limiting event from a core performance evaluation standpoint (feedwater line break inside containment) has been quantitatively analyzed in Section 6.3. Therefore, the following discussion provides information not presented in Section 6.3.

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of originally rated power) and was not reanalyzed for uprated 3,988 MWt power conditions. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

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, strict engineering codes and standards, and environmental requirements.

15.6.6.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.6.2 Sequence of Events and Systems Operation

15.6.6.2.1 Sequence of Events

The sequence of events is shown in Table 15.6-21.

Since automatic actuation and operation of the ECCS is a system design basis, no Operator actions are required for this accident.

15.6.6.2.2 Systems Operations

It is assumed that the normally operating plant instrument and controls are functioning. Credit is taken for the actuation of the reactor isolation system and ECCS. The RPS (SRVs, ECCS, and CRD) and plant protection system (RHR heat exchangers) are assumed to function properly to assure a safe shutdown.

The ESF and RCIC/HPCS systems are assumed to operate normally.

15.6.6.2.3 Effect of Single Failures and Operator Errors

The feedwater line break outside the containment is a special case of the general LOCA break spectrum considered in detail in Section 6.3. The general single-failure analysis for a LOCA is discussed in detail in Section 6.3.3.3. For the feedwater line break outside the containment, since the break is isolatable, either the RCIC or the HPCS can provide adequate flow to the vessel to maintain core cooling and prevent fuel rod clad failure. A single failure of either the HPCS or the RCIC would still provide sufficient flow to keep the core covered with water (see Section 6.3 and Appendix 15A for detailed description of analysis).

15.6.6.3 Core and System Performance

15.6.6.3.1 Qualitative Summary

The accident evaluation qualitatively considered in this section is considered to be a conservative and envelope assessment of the consequences of the postulated failure (i.e., severance) of one of the feedwater piping lines external to the containment.

The accident is postulated to occur at the input parameters and initial conditions as given in Table 6.3-1.

15.6.6.3.2 Qualitative Results

The feedwater line break outside the containment is less limiting than either the steam line break outside the containment (analysis presented in Sections 6.3.3 and 15.6.4), or the feedwater line break inside the containment (analysis presented in Sections 6.3.3 and 15.6.5). It is far less limiting than the DBA (the recirculation line break analysis presented in Sections 6.3.3 and 15.6.5).

The reactor scrams on low water level and the RCIC and HPCS restore the reactor water level to the normal elevation. The fuel is covered throughout the transient, and there are no pressure or temperature transients sufficient to cause fuel damage.

15.6.6.3.3 Consideration of Uncertainties

This event was conservatively analyzed and uncertainties were adequately considered (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 RCPB or the steam power conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the main steam lines as described in Section 15.6.4. The feedwater system piping break is less severe than the main steam line break. Results of analysis of this event can be found in Sections 6.2.3 or 6.2.4.

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 specific design basis analysis is presented. However, the radiological consequences of this event are enveloped by the results of analyses for the main steam line break (presented in Section 15.6.4.5). This is considered justified since the feedwater line check valves isolate the reactor from the downstream side of the break at time Ø+ sec after accident initiation. Therefore, there is no reactor coolant backflow via the feedwater lines connected to the RPV. The only contribution is from main steam, which must first pass through the turbines, condenser, and other condensate and feedwater system components, which reduce isotopic concentrations due to decay and demineralization. In the main steam line break analysis, a coincident iodine spike is assumed based on a compound spiking sequence giving 4 uCi/gm dose equivalent I-131, as described in Section 15.6.4.5.2.

15.6.7 References

1. Moody, F. J. Maximum Two-Phase Vessel Blowdown From Pipes. ASME Paper Number 65-WA/HT-1, March 15, 1965.

2. Nguyen, D., et al. Radiological Accident Evaluation - The CONCAC03 Code, NEDO-21143-1, December 1981.

3. DRAGON Code, Dose and Radioactivity From Nuclear Facility Gaseous Outflows, NU-115, Version 5, Level 0, October 1987.

4. USNRC Standard Technical Specifications for General Electric Boiling Water Reactors. NUREG-0123, Rev. 2, Washington, D.C., August 1979.

5. USNRC Standard Review Plan, Radiological Consequences of a Small Line Carrying Primary Coolant Outside Containment, 15.6.2, Rev. 2, July 1981.

6. NUREG/CR-6604, RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation, April 1998 (with Supplements 1 and 2).

7. Not Used.

8. Not Used.

9. Not Used.

10. Not Used.

11. Not Used.

12. Licensing Topical Report, Power Uprate Licensing Evaluation for Nine Mile Point Unit 2, NEDC-31994P, Revision 1, May 1993.

13. Nine Mile Point Nuclear Power Station Unit 2, SAFER/GESTR-LOCA Loss-of-Coolant Analysis, NEDC-31830P, Revision 1, November 1990.

TABLE 15.6-1

INSTRUMENT LINE BREAK ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Realistic Basis Assumptions
1. Data and assumptions used to estimate radioactive source from postulated accidents	
a. Power level	3,536 MWt
b. Maximum Technical Specification limit for iodine	4.0 uCi/gm
c. Design coolant iodine activity before accident	Table 15.6-2
d. Release of activity by isotope	Table 15.6-3
2. Data and assumptions used to estimate activity released	
a. Duration of event	2 hr
b. Total mass of coolant released	19,800 lb (8.99+06 g)
c. Mass of coolant flashing to steam	7,583.41 lb (3.44+06 g)
3. Other pertinent data and assumptions	
a. Control room free air volume	3.81+05 ft^3
b. Control room air intake rate	0-30 sec 1650 cfm 30 sec - 20 min 2750 cfm 20 min - 720 hr 1650 cfm
c. Control room recirculation rate	675 cfm
d. Intake and recirculation filter halogen removal efficiencies	0%
4. Dispersion data	
a. $\frac{X}{Q}$ - EAB 0-2 hr	1.90-04 sec/m^3
b. $\frac{X}{Q}$ - LPZ 0-2 hr	1.78-05 sec/m^3
c. $\frac{X}{Q}$ - Control room 0-8 hr	1.29-03 sec/m^3
d. Breathing rate - all locations	3.47-04 m^3/sec
5. Dose data	Table 15.6-4

TABLE 15.6-2

INSTRUMENT LINE FAILURE (REALISTIC ANALYSIS)
DESIGN COOLANT IODINE ACTIVITY BEFORE ACCIDENT

Isotope	Design Coolant Iodine Activity (uCi/gm)	Activity Normalized to Technical Specification Limit (uCi/gm)
I-131	1.3-02	6.12-01
I-132	2.2-01	1.04+01
I-133	1.6-01	7.54+00
I-134	4.0-01	1.88+01
I-135	1.7-01	8.01+00

TABLE 15.6-3

INSTRUMENT LINE FAILURE (REALISTIC ANALYSIS) ACTIVITY RELEASED TO THE ENVIRONMENT

Isotope	Ci
I-131	2.1+00
I-132	3.6+01
I-133	2.6+01
I-134	6.4+01
I-135	2.8+01
Total	1.56+02

NOTE: 2.1+00 = 2.1×10^{0}

TABLE 15.6-4

INSTRUMENT LINE FAILURE RADIOLOGICAL EFFECTS
(REALISTIC ANALYSIS)

	Whole-Body Dose (Rem)	Thyroid Dose (Rem)	Beta Dose (Rem)
Exclusion area (2 hr)	1.43-2	1.36+0	3.27-3
Low population zone (2 hr)	1.34-3	1.27-1	3.07-4
Control room*	2.55-3	7.16+0	1.09-2

NOTE: 1.43-2 = 1.43×10^{-2}

* Control room quoted doses are the maximum values calculated for time periods for and beyond the release duration of 2 hr.

TABLE 15.6-5

SEQUENCE OF EVENTS FOR STEAM LINE BREAK
OUTSIDE CONTAINMENT

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	Guillotine break of one main steam line outside primary containment.
0.5	High steam line flow signal initiates closure of MSIVs.
<1.25	Reactor begins scram.
≤5.5	MSIVs fully closed.
⍰~30	RCIC and HPCS would initiate on low-low water level (RCIC considered unavailable; HPCS assumed single failure and therefore not available).
~70	SRVs open on high vessel pressure. The valves open and close to maintain vessel pressure at approximately 1,100 psi.
~100	Reactor water level begins to drop slowly due to loss of steam through the SRVs; reactor pressure still at approximately 1,100 psi.
~720	ADS would signal to initiate on low-low-low water level.
~840	ADS initiated. Vessel depressurizes rapidly.
~1,040	Low pressure ECCS initiated with reactor fuel partially uncovered.
~1,135*	Core effectively reflooded and clad temperature heatup terminated; no fuel rod failure.

* See Section 6.3.3.

TABLE 15.6-6

STEAM LINE BREAK ACCIDENT - PARAMETERS
TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Basis Assumptions
1. Data and assumptions used to estimate radioactive source from postulated accidents		
a. Power level[1]	4,067 MWt	3,489 MWt
b. Release of activity by nuclide	Table 15.6-8	Table 15.6-11
c. Reactor coolant activity before the accident	Table 15.6-7	Table 15.6-10
d. Maximum Technical Specification limit for iodine	4.0 uCi/gm Dose equivalent I-131	N/A
2. Data and assumptions used to estimate activity released		
a. Isolation valve closure time	5.5 sec	5.5 sec
b. Total mass released	4.8+07 gm	4.8+07 gm
(1) Mass of steam released	7.1+06 gm	7.1+06 gm
(2) Mass of coolant released	4.1+07 gm	4.1+07 gm
(a) Mass of coolant released in liquid form	2.56+07 gm	-
(b) Mass of coolant that flashes and is released as steam	1.58+07 gm	-
c. Control room		
(1) Pressure envelope volume	3.8+05 ft^3	4.8+05 ft^3
(2) Air intake flow rate	N/A	1.5+03 cfm
(3) Recirculation flow rate	N/A	7.5+02 cfm
(4) Intake and recirculation halogen filter efficiency		
0 - 30 sec	N/A	99%
30 sec - 30 day	N/A	99%
d. Breathing rate	3.5-04 m^3/sec	3.47-04 m^3/sec

TABLE 15.6-6 (Cont'd.)

	Design Basis Assumptions	Realistic Basis Assumptions
3. Dispersion data		
a. $\frac{X}{Q}$ - EAB 0-2 hr	1.19-04 sec/m^3	2.19-05 sec/m^3
b. $\frac{X}{Q}$ - LPZ 0-2 hr	1.62-05 sec/m^3	6.48-06 sec/m^3
c. $\frac{X}{Q}$ - Control room	1.47-03 sec/m^3	1.29-03 sec/m^3
4. Dose data		
a. Computer code for dose calculations	N/A	DRAGON4
b. Doses	Table 15.6-9	Table 15.6-12

NOTE: 3.8+05 = $3.8x10^5$

(1) Power level is that which associates with the maximum mass loss.

TABLE 15.6-7

STEAM LINE BREAK ACCIDENT (DESIGN BASIS ANALYSIS)
DESIGN COOLANT IODINE ACTIVITY BEFORE ACCIDENT

Isotope	Design Coolant Iodine Activity (uCi/gm)	Activity Normalized to Technical Specification Limits (uCi/gm)
I-131	1.3-02	1.00+00
I-132	2.2-01	1.69+01
I-133	1.6-01	1.23+01
I-134	4.0-01	3.08+01
I-135	1.7-01	1.31+01

NOTE: 1.3-02 = 1.3×10^{-2}

TABLE 15.6-8

STEAM LINE BREAK ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT

Isotope	Activity (Ci)
I-131	4.16E+01
I-132	7.03E+02
I-133	5.11E+02
I-134	1.28E+03
I-135	5.43E+02
Total halogens	3.08E+03
Kr-83m	1.74E-01
Kr-85m	2.99E-01
Kr-85	9.54E-04
Kr-87	1.06E+00
Kr-88	1.06E+00
Xe-131m	7.64E-04
Xe-133m	1.45E-02
Xe-133	4.09E-01
Xe-135m	1.36E+00
Xe-135	1.14E+00
Xe-138	4.34E+00
Total noble gases	9.87E+00
Cs-134	2.71E-01
Cs-136	1.75E-01
Cs-137	7.01E-01
Cs-138	5.10E+02
Total alkali metals	5.11E+02

NOTE: 4.16+01 = 4.16×10^{1}

TABLE 15.6-9

STEAM LINE BREAK ACCIDENT (DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

	Total Effective Dose Equivalent (Rem)
Exclusion area (2 hr)	3.9-01
Low population zone (2 hr)	5.3-02
Control room	3.0+00

NOTE: 3.9-01 = 3.9×10^{-1}

TABLE 15.6-10

STEAM LINE BREAK ACCIDENT (REALISTIC ANALYSIS)
EXPECTED COOLANT IODINE ACTIVITY BEFORE ACCIDENT

Isotope	Activity (uCi/gm)
I-131	1.7-03
I-132	3.1-02
I-133	2.3-02
I-134	5.6-02
I-135	2.4-02

NOTE: 1.7-03 = 1.7×10^{-3}

TABLE 15.6-11

STEAM LINE BREAK ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT

Isotope	Activity (Ci)
I-131	6.80-02
I-132	1.20+00
I-133	9.20-01
I-134	2.20+00
I-135	9.60-01
Br-83	1.30-01
Br-84	1.70-01
Br-85	7.59-02
Total halogens	5.72+00
Kr-83m	6.51-03
Kr-85m	1.10-02
Kr-85	3.60-05
Kr-87	3.90-02
Kr-88	3.90-02
Kr-89	2.40-01
Xe-131m	2.80-05
Xe-133m	5.30-04
Xe-133	1.50-02
Xe-135m	5.01-02
Xe-135	4.30-02
Xe-137	2.80-01
Xe-138	1.60-01
Total noble gases	8.84-01

NOTE: 6.80-02 = 6.8×10^{-2}

TABLE 15.6-12

STEAM LINE BREAK ACCIDENT (REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole-Body Dose (Rem)	Thyroid Dose (Rem)	Beta Dose (Rem)
Exclusion area	6.28-05	5.37-03	1.83-05
Low population zone	1.86-05	1.59-03	5.40-06
Control room	2.30-06	8.29-04	1.49-05

NOTE: 6.28-05 = 6.28×10^{-5}

TABLE 15.6-13

LOSS-OF-COOLANT ACCIDENT - PARAMETERS TABULATED
FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions			Realistic Basis Assumptions
1. Data and assumptions used to estimate radioactive source from postulated accidents				
a. Power level	4,067 MWt			3,489 MWt
b. Release of activity to containment air		Gap Release Phase	Early In-Vessel Phase	100% of iodines in coolant flashing to steam
	Onset	2 min	32 min	
	Duration	0.5 hr	1.5 hr	
	Group Release Fraction			
	Noble Gas	0.05	0.95	
	Halogens	0.05	0.25	
	Alkali Metals	0.05	0.20	
	Tellurium Metals	0.00	0.05	
	Ba, Sr	0.00	0.02	
	Noble Metals	0.00	0.0025	
	Cerium Group	0.00	0.0005	
	Lanthanides	0.00	0.0002	
c. Release of activity to suppression pool	30% core iodine inventory			N/A
d. Iodine fractions	Containment	Pool		
(1) Organic	0.015	0.03		0.0
(2) Elemental	0.0485	0.97		1.0
(3) Particulate	0.95	0		0.0
e. Computer code used(1)	RADTRAD			DRAGON
f. Single active failure	Loss of diesel or failure of one MSIV(2)			N/A

TABLE 15.6-13 (Cont'd.)

	Design Basis Assumptions	Realistic Basis Assumptions
2. Data and assumptions used to estimate activity released		
a. Total mass of coolant released	N/A	2.72+8g
b. Four main steam lines		
Bypass leakage rates per line (scfh) (main steam tunnel release)	24	N/A
Horizontal area between isolation valves (IV) (ft^2)		N/A
MSLA	121.41	
MSLB	133.8	
MSLC	134.28	
MSLD	121.15	
Horizontal distance between IV (ft)		N/A
MSLA	19.75	
MSLB	21.77	
MSLC	21.84	
MSLD	19.71	
Bypass leakage delay time (hr)		N/A
1 valve closed (1 line)	5.26	
2 valves closed (3 lines)	7.11	
Iodine deposition applied		N/A
1 valve closed (1 line)	No	
2 valves closed (3 lines)	Yes	
c. Inboard main steam drain line		
Bypass leakage rate (scfh) (main steam tunnel release)	1.875	0
Horizontal area between IV (ft^2)	42.4	N/A
Horizontal distance between IV (ft)	28.11	N/A
Bypass leakage delay time (hr)	0	N/A
Iodine deposition applied	Yes	N/A

TABLE 15.6-13 (Cont'd.)

	Design Basis Assumptions	Realistic Basis Assumptions
d. Four post-accident sampling lines		
Bypass leakage rate per line (scfh) (PASS sample panel release)	0.2344	0
Horizontal area between IV (ft^2)		N/A
PASS Sample A	0.22	
PASS Sample B	0.22	
PASS Return A	0.22	
PASS Return B	0.25	
Horizontal distance between IV (ft)		N/A
PASS Sample A	1.0	
PASS Sample B	1.0	
PASS Return A	1.0	
PASS Return B	1.17	
Bypass leakage delay time (hr)	0	N/A
Iodine deposition applied	Yes	N/A
e. Two feedwater lines		
Bypass leakage rate per line (scfh) (main steam tunnel release)	12.	0
Horizontal area between IV (ft^2)		N/A
Feedwater Line A	72.01	
Feedwater Line B	71.9	
Horizontal distance between IV (ft)		N/A
Feedwater Line A	13.84	
Feedwater Line B	13.82	
Bypass leakage delay time (hr)	2.45	N/A
Iodine deposition applied	Yes	N/A
f. Outboard main steam drain line		
Bypass leakage rate (scfh) (main steam tunnel release)	0.625	0
Horizontal area between IV (ft^2)	0	N/A
Horizontal distance between IV (ft)	0	N/A
Bypass leakage delay time (hr)	0	N/A
Iodine deposition applied	No	N/A

TABLE 15.6-13 (Cont'd.)

	Design Basis Assumptions	Realistic Basis Assumptions
g. Reactor water cleanup line		
Bypass leakage rate (scfh) (main steam tunnel release)	2.5	0
Horizontal area between IV (ft^2)	26.28	N/A
Horizontal distance between IV (ft)	13.17	N/A
Bypass leakage delay time (hr)	2.45	N/A
Iodine deposition applied	Yes	N/A
h. Drywell equipment drain (DER) line		
Bypass leakage rate (scfh) (combined radwaste and reactor building release)	1.25	0
Horizontal area between IV (ft^2)	14.67	N/A
Horizontal distance between IV (ft)	13.92	N/A
Bypass leakage delay time (hr)	0	N/A
Iodine deposition applied	Yes	N/A
i. Drywell equipment drain (DER) vent line		
Bypass leakage rate (scfh) (combined radwaste and reactor building release)	0.625	0
Horizontal area between IV (ft^2)	4.54	N/A
Horizontal distance between IV (ft)	8.39	N/A
Bypass leakage delay time (hr)	0	N/A
Iodine deposition applied	Yes	N/A
j. Drywell floor drain (DFR) line		
Bypass leakage rate (scfh) (combined radwaste and reactor building release)	1.875	0
Horizontal area between IV (ft^2)	13.41	N/A
Horizontal distance between IV (ft)	8.44	N/A
Bypass leakage delay time (hr)	0	N/A
Iodine deposition applied	Yes	N/A

TABLE 15.6-13 (Cont'd.)

	Design Basis Assumptions	Realistic Basis Assumptions
k. Drywell floor drain (DFR) vent line		
Bypass leakage rate (scfh) (combined radwaste and reactor building release)	0.9375	0
Horizontal area between IV (ft^2)	15.44	N/A
Horizontal distance between IV (ft)	19.22	N/A
Bypass leakage delay time (hr)	0	N/A
Iodine deposition applied	Yes	N/A
l. Drywell purge inlet line		
Bypass leakage rate (scfh) (SGTS building release)	4.38	0
Horizontal area between IV (ft^2)	40.4	N/A
Horizontal distance between IV (ft)	11.65	N/A
Bypass leakage delay time (hr)	2.45	N/A
Iodine deposition applied	Yes	N/A
m. Wetwell purge inlet line		
Bypass leakage rate (scfh) (SGTS building release)	3.75	0
Horizontal area between IV (ft^2)	34.43	N/A
Horizontal distance between IV (ft)	10.96	N/A
Bypass leakage delay time (hr)	0	N/A
Iodine deposition applied	Yes	N/A
n. Drywell purge makeup line		
Bypass leakage rate (scfh) (SGTS building release)	0.625	0
Horizontal area between IV (ft^2)	4.57	N/A
Horizontal distance between IV (ft)	9.01	N/A
Bypass leakage delay time (hr)	0	N/A
Iodine deposition applied	Yes	N/A

TABLE 15.6-13 (Cont'd.)

	Design Basis Assumptions	Realistic Basis Assumptions
o. Wetwell purge makeup line		
Bypass leakage rate (scfh) (SGTS building release)	0.625	0
Horizontal area between IV (ft^2)	9.4	N/A
Horizontal distance between IV (ft)	17.38	N/A
Bypass leakage delay time (hr)	0	N/A
Iodine deposition applied	Yes	N/A
p. Not Used		
q. 3 instrument air lines 2 CPS lines 1 GSN line		
Bypass leakage rate, total for all lines (scfh) (SGTS building release)	3.6	0
Horizontal area between IV (ft^2)		N/A
IAS line 3	1.43	
IAS line 4	8.78	
IAS line 5	1.99	
GSN line	1.96	
CPS line 1	4.17	
CPS line 2	1.42	
Horizontal distance between IV (ft)		N/A
IAS line 3	3.39	
IAS line 4	20.83	
IAS line 5	4.73	
GSN line	12.04	
CPS line 1	15.19	
CPS line 2	5.17	
Bypass leakage delay time (hr)	0	N/A
Iodine deposition applied	Yes	N/A

TABLE 15.6-13 (Cont'd.)

	Design Basis Assumptions	Realistic Basis Assumptions
r. 2 instrument air lines to ADS accumulator		
Bypass leakage rate per line (scfh) (SGTS building release)	0.9375	0
Horizontal area between IV (ft^2)		N/A
IAS line 1	1.98	
IAS line 2	1.15	
Horizontal distance between IV (ft)		N/A
IAS line 1	4.71	
IAS line 2	2.72	
Bypass leakage delay time (hr)	0	N/A
Iodine deposition applied	Yes	N/A
s. Containment leakage rate (main stack release t=60 min to t=720 hr) (radwaste/reactor building release from t=0 to t=60 min)	1.1% per day of primary containment volume reduced by 50% at t=24 hr	1.1% per day of primary containment volume for duration of accident
Drywell spray removal applied	Yes	
t. TIP leakage rate (main stack release from t=60 min to t=720 hr) (radwaste/reactor building vent release from t=0 to t=60 min)	0.12% per day of drywell volume reduced by 50% at t=24 hr	N/A
Drywell spray removal applied		
u. Reactor building leak rate (main stack release)	4,000 cfm through SGTS	3,500 cfm through SGTS
v. Percentage mixing in reactor building air	50%	50%
w. Reactor building pressurization time (radwaste/reactor building vent release)	60 min	129 sec
x. SGTS halogen filtration efficiency	99%	99%

TABLE 15.6-13 (Cont'd.)

	Design Basis Assumptions		Realistic Basis Assumptions
y. ESF leakage to reactor building (main stack release from t=60 min to 720 hr) (radwaste/reactor building vent release from t=0 to t=60 min)			
(1) Leak rate	62 gpm		0.0
(2) Iodine partition factor (air/water)	0.1		0.0
3. All other pertinent data	N/A		
a. Primary containment			
(1) Drywell free air volume	3.06+5 ft^{3}[9]		N/A
(2) Primary containment free air volume	4.97+5 ft^{3}[9]		4.73+5 ft^{3}[9]
(3) Suppression pool volume	1.45+5 ft^3		N/A
b. Reactor building			
(1) Free air volume	3.88+6 ft^3		3.88+6 ft^3
c. Control room			
(1) Free air volume	3.81+5 ft^3		3.81+05 ft^3
(2) Intake rate	<u>Time</u>	<u>Intake Rate (cfm)</u>	1.50+3 cfm
	0 sec - 1200 sec	2750	
	1200 sec - 720 hr	1650	
(3) Recirculation rate	625 cfm		7.50+2 cfm
(4) Intake/recirculation halogen filtration efficiency	0-80 sec	0%	99%
	80 sec - 720 hr	99%	
(5) Unfiltered inleakage rate	250 cfm		N/A
d. Drywell spray removal			N/A
(1) Manual initiation	20 min		
(2) Spray removal rate	19.8 per hr(reduced to 1.98 per hr at DF=50)		
(3) Maximum decontamination factor (DF) for elemental iodine	200		

TABLE 15.6-13 (Cont'd.)

	Design Basis Assumptions	Realistic Basis Assumptions
4. Dispersion data (s/m^3)		
a. Stack		
0-2 hr EAB	2.96-5	1.16-7
0-8 hr LPZ	1.42-5	4.32-7
8-24 hr LPZ	5.41-7	3.21-7
24-96 hr LPZ	2.31-7	1.69-7
96-720 hr LPZ	7.65-8	6.73-8
0-2 hr control room	8.03-5	8.10-5
2-8 hr control room	4.48-5	8.10-5
8-24 hr control room	1.68-5	2.44-8
24-96 hr control room	1.20-5	2.10-8
96-720 hr control room	8.83-6	1.69-8
b. Radwaste/reactor building vent[(8)]		
0-2 hr EAB	1.19-4	2.19-5
0-8 hr LPZ	1.62-5	6.48-6
8-24 hr LPZ	1.09-5	N/A
24-96 hr LPZ	4.59-6	N/A
96-720 hr LPZ	1.33-6	N/A
0-2 hr control room	1.09-3	2.13-4
2-8 hr control room	7.23-4	2.13-4
8-24 hr control room	2.50-4	N/A
24-96 hr control room	1.92-4	N/A
96-720 hr control room	1.47-4	N/A

TABLE 15.6-13 (Cont'd.)

	Design Basis Assumptions	Realistic Basis Assumptions
c. Main steam tunnel		
0-2 hr EAB	1.19-4	N/A
0-8 hr LPZ	1.62-5	N/A
8-24 hr LPZ	1.09-5	N/A
24-96 hr LPZ	4.59-6	N/A
96-720 hr LPZ	1.33-6	N/A
0-2 hr control room	1.47-3	N/A
2-8 hr control room	9.74-4	N/A
8-24 hr control room	3.63-4	N/A
24-96 hr control room	2.45-4	N/A
96-720 hr control room	1.90-4	N/A
d. Radwaste tunnel (PASS area)		
0-2 hr EAB	1.19-4	N/A
0-8 hr LPZ	1.62-5	N/A
8-24 hr LPZ	1.09-5	N/A
24-96 hr LPZ	4.59-6	N/A
96-720 hr LPZ	1.33-6	N/A
0-2 hr control room	3.84-4	N/A
2-8 hr control room	2.28-4	N/A
8-24 hr control room	8.23-5	N/A
24-96 hr control room	6.28-5	N/A
96-720 hr control room	4.57-5	N/A
e. SGTS building		
0-2 hr EAB	1.19-4	N/A
0-8 hr LPZ	1.62-5	N/A
8-24 hr LPZ	1.09-5	N/A
24-96 hr LPZ	4.59-6	N/A
96-720 hr LPZ	1.33-6	N/A
0-2 hr control room	5.33-4	N/A
2-8 hr control room	3.72-4	N/A
8-24 hr control room	1.36-4	N/A
24-96 hr control room	9.17-5	N/A
96-720 hr control room	6.72-5	N/A

TABLE 15.6-13 (Cont'd.)

NOTE: 1.45-4 = 1.45 x 10^{-4}

(1) Dragon Code, Dose and Radioactivity from Nuclear Facility Gaseous Outflows, NU-115. RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation, NUREG/CR-6604.

(2) Both the loss-of-diesel and failure-of-one-MSIV accident scenarios are analyzed. The highest dose is from the failure-of-one-MSIV and is presented in Table 15.6-16b.

(3) Not used.

(4) Not used.

(5) Not used.

(6) Not used.

(7) Not used.

(8) These values are used for the 60-min release directly to the environment during the period when reactor building pressure is above -0.25 in W.G.

(9) The calculated free air volumes are a drywell free air volume of 306,200 ft^3 and a primary containment free air volume of 496,800 ft^3. The radiological LOCA analysis that was performed using the values tabulated in the USAR is conservative when compared to the new calculated primary containment volumes.

TABLE 15.6-14

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY AVAILABLE FOR RELEASE FROM PRIMARY
CONTAINMENT (AND DEPOSITED) AT END OF RELEASE FROM VESSEL

Isotope	Activity* (curies)	Isotope	Activity* (curies)
Kr-83m	6.11E+06	Mo-99	5.00E+05
Kr-85m	1.99E+07	Tc-99m	3.95E+05
Kr-85	1.56E+06	Ru-103	4.27E+05
Kr-87	1.74E+07	Ru-105	2.20E+05
Kr-88	4.47E+07	Ru-106	1.75E+05
Kr-89	3.94E-04	Rh-105	2.75E+05
Xe-131m	1.20E+06	Y-90	4.43E+03
Xe-133m	6.31E+06	Y-91	2.55E+04
Xe-133	2.08E+08	Y-92	2.56E+05
Xe-135m	1.88E+05	Y-93	2.67E+04
Xe-135	6.86E+07	Zr-95	3.46E+04
Xe-137	1.06E-01	Zr-97	3.22E+04
Xe-138	1.34E+06	Nb-95	3.47E+04
I-131	3.23E+07	La-140	8.50E+04
I-132	2.72E+07	La-141	2.42E+04
I-133	6.19E+07	La-142	1.36E+04
I-134	1.48E+07	Pr-143	3.08E+04
I-135	5.02E+07	Nd-147	1.39E+04
Rb-86	7.25E+04	Am-241	5.80E+00
Cs-134	7.27E+06	Cm-242	1.43E+03
Cs-136	2.26E+06	Cm-244	9.54E+01
Cs-137	4.34E+06	Ce-141	8.78E+04
Sb-127	5.84E+05	Ce-143	7.76E+04
Sb-129	1.27E+06	Ce-144	7.28E+04
Te-127m	7.97E+04	Np-239	1.11E+06
Te-127	5.40E+05	Pu-238	2.86E+02
Te-129m	2.56E+05	Pu-239	2.64E+01
Te-129	9.77E+05	Pu-240	3.72E+01
Te-131m	7.47E+05	Pu-241	1.08E+04
Te-132	7.45E+06	Sr-89	1.94E+06
Ba-137m	4.19E+06	Sr-90	2.50E+05
Ba-139	1.42E+06	Sr-91	2.12E+06
Ba-140	3.72E+06	Sr-92	1.61E+06

* At t=2 hr.

TABLE 15.6-15

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TABLE 15.6-15a

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TABLE 15.6-15b

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT[1]

Isotope	Activity (curies)	Isotope	Activity (curies)
Kr-83m	3.41E+03	Mo-99	2.42E+00
Kr-85m	4.06E+04	Tc-99m	2.12E+00
Kr-85	3.21E+05	Ru-103	3.03E+00
Kr-87	5.88E+03	Ru-105	4.76E-01
Kr-88	4.74E+04	Ru-106	1.35E+00
Kr-89	5.01E-01	Rh-105	1.30E+00
Xe-131m	1.23E+05	Y-90	6.82E-01
Xe-133m	1.68E+05	Y-91	2.08E-01
Xe-133	1.58E+07	Y-92	2.81E-01
Xe-135m	4.12E+01	Y-93	8.77E-02
Xe-135	1.54E+06	Zr-95	2.54E-01
Xe-137	1.89E+00	Zr-97	1.23E-01
Xe-138	2.47E+02	Nb-95	2.67E-01
I-131	1.65E+04	La-140	5.93E+00
I-132	1.25E+04	La-141	4.85E-02
I-133	5.43E+03	La-142	1.39E-02
I-134	1.33E+02	Pr-143	2.05E-01
I-135	1.31E+03	Nd-147	8.41E-02
Rb-86	5.85E-01	Am-241	4.50E-05
Cs-134	6.57E+01	Cm-242	1.09E-02
Cs-136	1.77E+01	Cm-244	7.40E-04
Cs-137	3.93E+01	Ce-141	6.15E-01
Sb-127	2.97E+00	Ce-143	3.39E-01
Sb-129	2.72E+00	Ce-144	5.59E-01
Te-127m	5.97E-01	Np-239	5.27E+00
Te-127	2.87E+00	Pu-238	2.22E-03
Te-129m	1.78E+00	Pu-239	2.05E-04
Te-129	2.15E+00	Pu-240	2.89E-04
Te-131m	3.21E+00	Pu-241	8.40E-02
Te-132	3.71E+01	Sr-89	1.40E+01
Ba-137m	3.02E+01	Sr-90	1.94E+00
Ba-139	1.38E+00	Sr-91	6.81E+00
Ba-140	2.30E+01	Sr-92	2.44E+00
		Total	1.81E+07

[1] Total release for 30 days.

TABLE 15.6-16

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TABLE 15.6-16a

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TABLE 15.6-16b

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

	Total Effective Dose Equivalent (rem)
Exclusion area (2 hr)	0.66
Low population zone (30-day)	0.77
Control room (30-day)	1.65

TABLE 15.6-17

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
REACTOR COOLANT IODINE CONCENTRATIONS

Isotope	Design Reactor Coolant (uCi/gm)	Normalized Reactor Coolant (uCi/gm)
I-131	1.3-2	6.1-1
I-132	2.2-1	1.0+1
I-133	1.6-1	7.5+1
I-134	3.9-1	1.8+1
I-135	1.7-1	8.0+0

NOTE: 1.3-2 = 1.3×10^{-2}

* Design reactor coolant iodines normalized to maximum Technical Specification limit of 4 uCi/gm.

TABLE 15.6-18

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN CONTAINMENT DUE TO IODINE SPIKING

(Ci)

Isotope	Initial Airborne Activity	Airborne Activity Available for Release
I-131	8.33+4	1.67+4
I-132	1.41+6	2.82+5
I-133	1.02+6	2.04+5
I-134	2.50+6	5.00+5
I-135	1.90+6	2.18+5

NOTE: 8.33+4 = 8.33×10^4

TABLE 15.6-19

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT

(Ci)

Isotope	Activity Released
I-131	1.79+1
I-132	4.21+0
I-133	2.70+1
I-134	2.87+0
I-135	9.59+0
Xe-131m*	6.85+1
Xe-133m*	4.25+2
Xe-133*	1.28+4
Xe-135m*	8.72+3
Xe-135*	5.27+3

NOTE: 1.79+1 = 1.79×10^{1}

* The xenon isotopes were produced by the decay of the iodines released.

TABLE 15.6-20

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole-Body Dose (Rem)	Thyroid Dose (Rem)	Beta Dose (Rem)
Exclusion area (2 hr)	1.53-4	1.41-2	3.54-5
Low-population zone (30 d)	4.65-4	5.83-3	2.14-4
Control room (30 d)	3.75-4	1.48-3	4.42-3

NOTE: 1.53-4 = 1.53×10^{-4}

TABLE 15.6-21

SEQUENCE OF EVENTS FOR FEEDWATER
LINE BREAK OUTSIDE CONTAINMENT

Note: These results are for Cycle 1. This event does not set reactor operating limits and is not reanalyzed for power uprate or for each reload cycle.

Time (sec)	Event
0	One feedwater line breaks.
0+	Feedwater line check valves isolate the reactor from the break.
~5	Reactor scrams on low water level.
<30	HPCS and RCIC would initiate on low-low water level and maintain water level above the L1 trip and eventually restore it to the normal elevation.
1 to 2 hr	Normal reactor cooldown procedure established.

15.7 RADIOACTIVE RELEASE FROM SUBSYSTEMS AND COMPONENTS

15.7.1 Radioactive Gas Waste System Leak or Failure

The main condenser offgas treatment system is examined under severe failure mode conditions for effects on the plant safety profile.

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of originally rated power) and the radiological consequences have been recalculated for rated 3,467 MWt power conditions. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

15.7.1.1 Identification of Causes and Frequency Classification

15.7.1.1.1 Identification of Causes

Events that could cause a gross failure in the offgas treatment system are:

1. A seismic occurrence, greater than design basis.

2. A hydrogen detonation that ruptures the system pressure boundary.

3. Failure of spatially related equipment.

The offgas equipment and piping are designed to contain any hydrogen-oxygen detonation that has a reasonable probability of occurring. Therefore, a detonation is not considered a possible failure mode.

The offgas system is isolated from other systems or components that could cause any serious interaction or failure.

The seismic failure is the only conceivable event that could cause significant system damage and result in the release of significant activity to the environment.

Even though the offgas system is located in an area of the turbine building designed and analyzed for seismic conditions, an event more severe than the design requirements is arbitrarily assumed to occur, resulting in the failure of the offgas system.

The design basis, description, and performance evaluation of the offgas system are given in Section 11.3.

Fire suppression systems are described in Section 9.5.1.

15.7.1.1.2 Frequency Classification

The failure of the offgas system is categorized as a limiting fault.

15.7.1.2 Sequence of Events and System Operation

15.7.1.2.1 Sequence of Events

The sequence of events following the failure of the offgas system is shown in Table 15.7-1.

15.7.1.2.2 Identification of Operator Actions

Gross failure of the offgas system may require manual scram and isolation of the system from the main condenser.

15.7.1.2.3 Systems Operation

In analyzing the postulated offgas system failure, no credit is taken for the operation of plant and RPSs, or of ESFs. The break occurs just downstream of the SJAEs. Credit is taken for the functioning of normally operating plant instruments and controls, and other systems only in assuming the following:

1. Capability to detect the failure itself, indicated by an alarmed loss of flow in the offgas system and by an alarmed increase in activity at the main stack.

2. Capability to isolate the system and shut down the reactor.

3. Operational indicator and annunciators in the main control room.

15.7.1.2.4 Effect of Single Failures and Operator Errors

After the initial system gross failure, the failure of the Operator to actuate a system isolation could affect the analysis. However, the seismic event that is assumed to occur beyond the present plant design basis for nonsafety equipment would undoubtedly cause the tripping of the turbine or lead to a load rejection. This would initiate a scram and negate the need for the Operator to initiate a reactor shutdown; see Appendix 15A for a detailed discussion.

15.7.1.3 Core and System Performance

The postulated failure results in a manual system isolation necessitating reactor shutdown because of loss of vacuum in the main condenser. This transient is analyzed in Section 15.2.5.

15.7.1.4 Barrier Performance

The postulated failure is the rupture of the offgas system pressure boundary. No credit is taken for performance of secondary barriers, except to the extent inherent in the assumed equipment release fractions discussed in Section 15.7.1.5.

15.7.1.5 Radiological Consequences

The design basis analysis provided for this accident is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining the adequacy of the plant design to meet 10CFR100 guidelines.

The exposures calculated using design basis assumptions are so far below the allowable limits of 10CFR100 that no realistic analysis of the radiological consequences of this accident is provided.

15.7.1.5.1 Design Basis Analysis

Specific parametric values used in this evaluation are presented in Table 15.7-2. These data are in accordance with RG 1.98 and result in a conservative estimate of the radiological consequences.

<u>Fission Product Release</u>

<u>Initial Conditions</u> The activity in the offgas system is based on the following conditions:

1. 6 scfm condenser air in-leakage.

2. 353,634 uCi/sec noble gas activity after 30-min delay.

<u>Assumptions</u>

1. 100 percent of the noble gas activity in the charcoal adsorber beds and the piping and process equipment is released.

2. Iodines and activation gases are neglected.

3. The SJAE continues to operate for a period of 1 hr following the failure.

The activity in the offgas system at the time of the accident is presented in Table 15.7-3.

<u>Fission Product Transport to the Environment</u>

The transport pathway consists of direct release from the failed charcoal adsorber, the offgas piping and components, and the SJAE to the environment without credit for the building ventilation system. The release of activity to the environment is presented in Table 15.7-4.

<u>Results</u>

The calculated exposures for the design basis analysis postulated by NUREG-75/087 are presented in Table 15.7-5. These values are a small fraction of the guidelines of 10CFR100 and are within the Branch Technical Position (BTP) ETSB 11-5 guideline of 0.5 rem for total body exposure to an individual at the nearest EAB.

15.7.2 Radioactive Liquid Waste System Leak or Failure (Release to the Atmosphere)

The NRC has revised the SRP, Revision 2, for Section 15.7.2, so that analysis of this event is no longer required.

15.7.3 Postulated Radioactive Releases Due to Liquid Radwaste Equipment Failure

The accident postulated by NUREG-0800 for this analysis is that an unspecified event causes the release of the contents of the tank (or component) containing the largest inventory of radionuclides in the liquid radwaste system that is most easily transported to groundwater.

The liquid radwaste area of the radwaste building is surrounded by a steel liner to prevent spillage from flowing out of the building and being transported to groundwater.

Liners are also provided for any radwaste component, the rupture of which would result in groundwater radionuclide concentrations greater than those allowed by Appendix B of 10CFR20.

An analysis of postulated ruptures of the condensate storage tanks (CSTs) is provided as the release of the tanks' contents may be transported to surface water.

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of originally rated power), and the radiological consequences have been recalculated for 4,068 MWt (120 percent of original licensed thermal power, 3,988 MWt + 2 percent for instrument error) power conditions. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

15.7.3.1 Condensate Storage Tanks Rupture

The two CSTs are located in the CST building next to the radwaste building. These tanks contain relatively low concentrations of radioactive liquid. For the purposes of this analysis, an unspecified event is assumed to cause the release of the contents of both CSTs.

15.7.3.1.1 Identification of Causes and Frequency Classification

Postulated events that could cause the release of the radioactive inventory of the CSTs are cracks in the vessels and Operator error.

The possibility of cracks and consequent low-level release rates receives primary consideration in system and component design, so the possibility of a failure is considered small.

A CST release caused by Operator error is also considered a remote possibility because operating techniques and administrative procedures emphasize detailed system and equipment operating instructions.

The probability of a complete rupture or malfunction accident is considered even lower than that for small cracks and Operator error. Although not analyzed for the requirements of Category I equipment, the CSTs are constructed in accordance with sound engineering principles. This accident is, therefore, expected to occur with the frequency of a limiting fault.

15.7.3.1.2 Sequence of Events and Systems Operation

1. Event begins; CSTs fail, and the contents are released into the condensate storage building.

2. Level alarms in the CSTs alert personnel.

3. Operator actions begin.

The rupture of the CSTs would leave little recourse to the Operator. No method of containing the discharge is available.

No credit for any Operator action has been taken in evaluating this event.

15.7.3.1.3 Core and System Performance

The failure of the CSTs does not directly affect the NSSS.

15.7.3.1.4 Barrier Performance

This event does not involve any containment barrier integrity.

15.7.3.1.5 Radiological Consequences

The radiological analysis provided for this accident is based on realistic conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10CFR100 guidelines.

The analysis is based on SRP 15.7.3, Revision 2, although no specific regulatory guideline requirements are established.

Specific values of parameters used in the evaluation are presented in Table 15.7-6.

Several Chapter 15 accident sections include both a design basis and a realistic basis analysis. The realistic basis analysis for such SAR sections is not updated beyond FSAR Amendment 28. However, because the only CSTs rupture analysis presented is a realistic analysis, it is and will continue to be updated as necessary.

Fission Product Release

The CST inventory corresponding to the design reactor steam activities in Table 11.1-1 is shown in Table 15.7-7. To determine the CST inventory that corresponds to the accident, the inventory in Table 15.7-7 is adjusted as an intermediate step based on the expected reactor steam activities in Table 11.1-1. The CSTs are assumed to fail simultaneously, releasing the entire contents of both tanks to the CST building. (This is conservative with respect to the SRP 15.7.3 assumption that 80 percent of the tanks' liquid volume is released.) The spillage proceeds, as surface water, directly to Lake Ontario.

Fission Product Transport to the Environment

The travel time to the Oswego & Metropolitan Water Board Intake is 29.6 hr, and the dilution factor at this intake is 45.3. Table 15.7-8 presents the fixed concentrations at the water intake and the fraction of maximum permissible concentration (MPC).

15.7.4 Fuel Handling Accident

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of originally rated power) and the design basis radiological consequences have been recalculated for 4,067 MWt (120% of original licensed thermal power, 3,988 MWt + 2 percent for instrument error) power conditions. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

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. A variety of events that qualify for the class of accidents has been investigated. The accident which produces the largest number of failed spent fuel rods is the drop of a spent fuel bundle onto the reactor core when the reactor vessel head is off.

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 viewpoint is the dropping of a fuel assembly onto the top of the core. The sequence of events is as follows:

Event	Approximate Elapsed Time
1. A fuel assembly is being handled by refueling equipment; assembly drops onto the top of the core.	0
2. Some of the fuel rods in both the drop assembly and reactor core are damaged, resulting in the release of gaseous fission products to the reactor coolant and eventually to the reactor building atmosphere.	0
3. The reactor building ventilation radiation monitoring system alarms to alert plant personnel, isolates the ventilation system, and starts operation of the SGTS.	<1 min
4. Operator actions begin.	<5 min

15.7.4.2.2 Identification of Operator Actions

Operation of other plant or RPSs, or ESF systems, is not expected.

15.7.4.2.3 System Operation

Normally, operating plant instrumentation and controls are assumed to function, although credit is taken only for the isolation of the normal ventilation system and the operation of the SGTS in the realistic fuel handling accident analyses presented in Section 15.7.4.5.2. Conservatively, no credit for isolation of the normal ventilation system and operation of SGTS is taken in the fuel handling accident analyses presented in Sections 15.7.4.5.1 and 15.7.4.5.3.

15.7.4.2.4 Effects of Single Failures and Operator Errors

The automatic ventilation isolation system includes: 1) the radiation monitoring detectors, 2) isolation dampers, and 3) the

SGTS, and is designed to single-failure criteria and safety requirements.

Refer to Sections 7.6 and 9.4 and Appendix 15A for further details.

15.7.4.3 Fuel Assembly Drop Accident Evaluation

15.7.4.3.1 Mathematical Model

The analytical methods and associated assumptions used to evaluate the consequences of this accident are considered to provide a realistic, yet conservative assessment of the consequences.

The kinetic energy acquired by a falling fuel assembly may be dissipated in one or more impacts.

To estimate the expected number of failed fuel rods in each impact, an energy approach is used.

The dropped assembly is considered to impact at a small angle, subjecting all the fuel rods in the dropped assembly to bending moments. The fuel rods are expected to absorb little energy prior to failure as a result of bending. For this reason it is assumed that all the rods in the dropped assembly fail. The total energy to be dissipated by the first impact is:

$$E = \left(W_B + W_G\right)\left(D\right) \qquad (15.7\text{-}1)$$

Where:

W_B = Weight of the bundle

W_G = Weight of the grapple

D = Drop distance

The rods in the impacted assemblies are assumed to fail in compression. The energy available for clad deformation is considered to be proportional to the mass ratio:

$$M_R = \frac{mass\ of\ cladding}{\left(mass\ of\ assembly - mass\ of\ fuel\ pellets\right)} \qquad (15.7\text{-}2)$$

Where:

M_R = Mass ratio

One half of the energy is considered to be absorbed by the falling assembly and one half by the assemblies which are

impacted. Therefore, the energy absorbed by the cladding of the impacted assemblies is:

$$E_c = \frac{1}{2}(E)(M_R) \qquad (15.7\text{-}3)$$

Where:

E_C = Energy absorbed by impacted assemblies

E = Total energy

Each rod that fails as a result of gross compression distortion is expected to absorb approximately 250 ft-lb before cladding failure, based on uniform 1-percent plastic deformation of the cladding. Therefore, the number of failed rods from the initial impact of the impacted assemblies is calculated by:

$$N_F = \frac{E_c}{250\ ft-lb} \qquad (15.7\text{-}4)$$

Where:

N_F = Number of failed rods

After the initial impact, the assembly is assumed to tip over and impact horizontally on the top of the core. The remaining available energy is used to predict the number of additional rod failures. The available energy was calculated by assuming a linear weight distribution in the assembly with a point load at the top of the assembly to represent the fuel grapple weight. The energy of this impact is calculated by:

$$E = W_G H_G + \int_O^{Hg} \frac{W_B}{H_B} y\ dy = W_G H_G + \frac{1}{2} W_B H_B \qquad (15.7\text{-}5)$$

Where:

W_G = Weight of the grapple

H_G = Height of the grapple

W_B = Weight of the bundle

H_B = Height of the bundle

As before, the energy is considered to be absorbed equally by the falling assembly and the impacted assemblies. Therefore, equations 15.7-3 and 15.7-4 are used again to determine the number of failures for the second impact.

15.7.4.3.2 Input Parameters and Initial Conditions for GE 9 (8x8) Fuel Using GESTAR II, NEDE-24011-P-A-10, Methodology

The following assumptions are used in the analysis of this accident:

1. The fuel assembly is dropped from the maximum height allowed by the fuel handling equipment (less than 32.95 ft).

2. The entire amount of potential energy, referenced to the top of the reactor core, is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the core and requires that the grapple cable breaks, allowing the grapple head and three sections of the telescoping mast to remain attached to the falling assembly.

3. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

4. All fuel rods, including tie rods, were assumed to fail by 1-percent strain in compression, the same mode as ordinary fuel rods. For the fuel designs considered here, there is no propensity for preferential failure of tie rods.

15.7.4.3.3 Results

<u>Energy Available</u>

Dropping a fuel assembly from the maximum height allowed by the refueling equipment, less than 32.95 ft, results in a kinetic energy acquired by the falling fuel assembly of less than 31,870 ft-lb and is dissipated in one or more impacts.

<u>Energy Loss Per Impact</u>

Based on the fuel geometry in the reactor core, a maximum of four fuel assemblies are struck by the impacting assembly. One half of the energy is considered to be absorbed by the falling assembly and one half by the four impacted assemblies.

The second impact is expected to be less direct. The assembly is assumed to tip over and impact horizontally on the top of the core. The remaining available energy is used to predict the number of additional rod failures. The available energy is calculated by assuming a linear weight distribution in the assembly with a point load at the top of the assembly to represent the fuel grapple weight. The energy is considered to

be absorbed equally by the falling assembly and the impacted assemblies.

Fuel Rods and Fuel Rod Failures

Each fuel assembly consists of an 8x8 arrangement of rods. Two water rods and 62 fuel rods make up an assembly. In the fuel rod failure analyses that follow, only damage to the 62 fuel rods is of concern. Eight of the fuel rods are tie rods which support the assembly during fuel handling. The tie rods are no more susceptible to bending failure than the other 54 fuel rods. Therefore, the tie rods are assumed to fail by 1-percent strain in compression, the same mode as ordinary fuel rods.

First Impact Failures The first impact dissipates 31,870 ft-lb of energy. It is assumed that 50 percent of this energy is absorbed by the dropped fuel assembly and that the remaining 50 percent is absorbed by the struck fuel assemblies in the core. Because the fuel rods of the dropped fuel assembly are susceptible to the bending mode of failure and because 1 ft-lb of energy is sufficient to cause cladding failure as a result of bending, all 62 rods of the dropped fuel assembly are assumed to fail.

Because the fuel rods of the struck assemblies are held rigidly in place in the core, they are susceptible only to the compression mode of failure. To cause cladding failure of one fuel rod as a result of compression, 250 ft-lb of energy is required. To cause failure of all the rods of the four struck assemblies, 250 x 62 x 4 or 62,000 ft-lb of energy would have to be absorbed in cladding alone. Thus, it is clear that not all the fuel rods of the struck assemblies can fail on the first impact. The number of fuel rod failures caused by compression is computed as follows.

The energy available for clad deformation is considered to be proportional to the mass ratio:

$$M_R = \frac{mass\ of\ cladding}{(mass\ of\ assembly - mass\ of\ fuel\ pellets)} \qquad (15.7\text{-}2)$$

and is equal to a maximum of .519 for the fuel designs considered here.

The energy absorbed by the cladding of the four impacted assemblies is:

$$E_c = (0.5)(31{,}870\ ft-lb)(.519) = 8270\ ft-lb \qquad (15.7\text{-}3)$$

Each rod that fails is expected to absorb approximately 250 ft-lb before cladding failure, based on uniform 1-percent plastic deformation of the cladding.

The number of rods failed in the impacted assemblies is:

$$N_F = \frac{(8270\ ft-lb)}{(250\ ft-lb)} = 33\,rods \qquad (15.7\text{-}4)$$

Thus, during the first impact, fuel rod failures are as follows:

Dropped assembly	62 rods (bending)
Struck assemblies	33 rods (compression)
	95 failed rods

Second Impact Failures The second impact dissipates 8,780 ft-lb of energy. It is assumed that 50 percent of this energy is absorbed by the dropped fuel assembly and that the remaining 50 percent is absorbed by the struck fuel assemblies in the core. Because all of the fuel rods in the dropped assembly were assumed to fail on the initial impact, all of the failures in the second impact are in the impacted assemblies. The number of fuel rod failures caused by compression on the second impact is computed as follows.

The available energy is calculated by assuming a linear weight distribution in the assembly with a point load at the top of the assembly to represent the fuel grapple weight.

$$\begin{aligned} E &= W_G H_G + \int_O^{H_B} \frac{W_B}{H_B}\, y\, dy = W_G H_G + \frac{1}{2} W_B H_B \\ &= (350\,lb)\left(\frac{160}{12}\right) + \frac{1}{2}(617)\left(\frac{160}{12}\right) = 8780\ ft-lb \end{aligned} \qquad (15.7\text{-}5)$$

As before, the energy is considered to be absorbed equally by the falling assembly and the impacted assemblies and the fraction available for clad deformation is .519. The energy available to deform clad in the impacted assemblies is:

$$E_c = (0.5)(8780\ ft = lb)(.519) = 2278\ ft-lb \qquad (15.7\text{-}3)$$

and the number of failures in the impacted assemblies is:

$$N_F = \frac{2278\ ft-lb}{250\ ft-lb} = 9 \qquad (15.7\text{-}4)$$

Thus, during the second impact the fuel rod failures are as follows:

Struck assemblies	9 rods (compression)
	9 failed rods

<u>Total Failures</u> The total number of failed rods resulting from the accident is as follows:

First impact	95	rods
Second impact	9	rods
	104	total failed rods

15.7.4.3.4 Input Parameters and Initial Conditions for GE 9 (8x8 Array), GE 11 (9x9 Array) GE14 and GNF2 (10 X 10 Arrays) Using Methodology in GSTAR II, NEDE-24011-P-A US Supplement.

An analysis for the GE 9 fuel, using the parameters and initial conditions in Section 15.7.4.3.2, with the exception that the weights associated with the new design of the refueling mast with grapple head (NF-500 with total maximum wet weight of 619 lb) were used, resulted in 117 failed rods instead of 104 failed rods. Therefore, the design basis analysis using the specific values of parameters as presented in Table 15.7-9 is bounding.

GESTAR II, NEDE-24011-P-A-11-US, analyzed the GE 11 fuel that is used at Unit 2 using the same methodology as in the analysis of the GE 9 fuel, with the following changes to the input parameters and initial conditions:

1. A more conservative threshold value before cladding failure (200 ft-lb).

2. NF-500 refueling mast with grapple head.

3. A drop height of 34 ft.

4. The fraction of energy available for clad deformation 0.510.

This resulted in a total of 140 failed fuel rods.

GESTAR II, NEDE-24011-P-A-14-US, analyzed the GE 14 fuel that is currently used at Unit 2 using the same methodology as in the analysis of the GE 9 fuel, with the following changes to the input parameters and initial conditions:

1. A more conservative threshold value before cladding failure (175 ft-lb).

2. NF-500 refueling mast with grapple head.

3. A drop height of 34 ft.

4. The fraction of energy available for clad deformation 0.526.

This resulted in a total of 172 failed fuel rods.

NEDC-33270P analyzed the GNF2 fuel using the same methodology as in the analysis of the GE9 fuel, with changes to the parameters associated with the GNF2 design and the NF-500 mast, resulting in a failure of 172 equivalent full-length rods.

15.7.4.4 Barrier Performance

The RCPB and primary containment are assumed to be open. The transport of fission products from the reactor building is discussed in Section 15.7.4.5.2.

15.7.4.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

1. The first, referred to as the design basis 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 10CFR50.67 criteria.

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

The fission product inventory in the fuel rods assumed to be damaged is based on 1,400 days of continuous GE14 fueled operation at 3,489 MWt for the realistic analysis and 4,067 MWt for the design basis analysis. GNF2 fuel was evaluated for 1,315 days of operation and determined to be bounded by the GE14 core inventory from a design basis consequence perspective. A 24-hr period for decay from the above power condition is assumed because it is not expected that fuel handling can begin within 24 hr following initiation of reactor shutdown.

15.7.4.5.1 Design Basis Analysis

The design basis analysis is based on an alternative source term (AST) as described in SRP 15.0.1 and RG 1.183. Specific values of parameters used in the evaluation are presented in Table 15.7-9. The design basis parameters listed in Table 15.7-9, the activity released to the reactor building listed in Table 15.7-10, the activity released to the environment listed in Table 15.7-11, and the doses presented in Table 15.7-12 are based on a design basis accident with the number of failed fuel rods equivalent to the failure of all the fuel rods in two assemblies. The radiological consequences given in those tables bound the fuel handling accident in which 124 fuel rods of 8x8 fuel, 140 fuel rods of 9x9 fuel, or 172 fuel rods of GE14 fuel are damaged. The radiological consequences given in those tables bound the fuel handling accident in which 172 equivalent full-length fuel rods of GNF2 fuel are damaged if the radial peaking factor of the GNF2 fuel remains below 1.79.

Fission Product Release from Fuel

The fission product inventory of a core average rod is adjusted by a peaking factor of 1.8 to establish the inventory of each damaged rod. Five percent of the noble gas inventory (10 percent for Kr-85) and 5 percent of the halogen inventory (8 percent for I-131) are assumed to be released to the fuel pool.

Fission Product Transport to the Environment

All of the noble gases and 0.57 percent (overall decontamination factor of 175) of the halogen inventory released from the fuel and mixing in the fuel pool are assumed to migrate from the pool and become airborne in the reactor building. The activity released to the fuel pool is presented in Table 15.7-10. The transport pathway to the environment consists of an instantaneous release to the environment of the activity that becomes airborne in the reactor building. Although the reactor building ventilation system would isolate on a high radiation signal, no credit for SGTS filtration/elevated release is taken. The activity released to the environment is presented in Table 15.7-11.

Results

The calculated exposures for the design basis analysis are presented in Table 15.7-12 and are a small fraction of the criteria of 10CFR50.67. Control room doses are calculated without credit for the control room emergency ventilation system (CREVS) and are less than the 10CFR50.67 criteria.

15.7.4.5.2 Realistic Analysis

The realistic fuel handling accident analysis is provided to illustrate the conservatism of the design basis analysis. The realistic analysis is presented here as it appeared in Amendment 28 of the FSAR and will not be updated.

The realistic analysis is based on a realistic but still conservative assessment of this accident. Specific values of parameters used in the evaluation are presented in Table 15.7-9.

Fission Product Release from Fuel

Fission product release estimates for the fuel handling accident are based on the following assumptions:

1. The reactor fuel has an average irradiation time of 1,000 days at 3,489 MWt up to 24 hr prior to the accident. This assumption results in an equilibrium fission product concentration at the time the reactor is shut down. Longer operating histories do not increase the concentration of biologically significant isotopes. The 24-hr decay period allows time to shut

down the reactor, depressurize the nuclear system, remove the reactor vessel head, and remove the reactor vessel upper internals. It is not expected that these operations could be accomplished in less than 24 hr and probably will require at least 48 hr.

2. An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity is in the fuel rod plena and available for release. This assumption is based on fission product release data from defective fuel experiments.

3. Because of the negligible particulate activity available for release from the fuel plena, none of the solid fission products are assumed to be released.

4. It is assumed that 124 fuel rods fail. This is considered to be conservative, because it is expected that significantly fewer than 124 rods would be damaged.

Fission Product Transport to the Environment

The transport pathway consists of mixing in the fuel pool, migration from the pool to the reactor building atmosphere, and release to the environment through the SGTS. All of the noble gases and 1 percent of the halogens in the pool are assumed to become airborne in the reactor building.

Based on these assumptions, the activity airborne in the reactor building is shown in Table 15.7-13. The release rate of activity under normal ventilation conditions is sufficient to cause a trip of the reactor building discharge plenum radiation monitors, resulting in reactor building isolation and SGTS startup. For 30 sec, until reactor building ventilation isolation is effected, release is through the reactor building vent. The remainder of the release (2 hr) is out the main stack via SGTS. Filtration is by the SGTS filters, with 99 percent removal efficiency for halogens.

The cumulative release to the environment is presented in Table 15.7-14.

Results

The calculated exposures for the realistic analysis are presented in Table 15.7-15 and demonstrate the margin of conservatism in the design basis analysis.

15.7.5 Spent Fuel Cask Drop Accident

The polar crane, which is used to lift the spent fuel shipping and storage casks, is equipped with a redundant lift mechanism and is single-failure proof.

As discussed in Section 9.1.4.2.2 the reactor building polar crane is designed as a Service Class A1 crane in accordance with CMAA Specification No. 70 and the design features described are consistent with the guidelines of NUREG-0554. The main hoist has a single-failure-proof design based on a maximum critical load of 132 tons, so that the single failure of any component in the hoist train does not result in loss of the lifted load. This also applies to the electrical components in the main hoist path. The transfer cask, which will be used for the NMPNS dry cask storage operations, will be within the capacity of the reactor building crane in for single-failure-proof handling. Consequently, for dry cask storage operations within the reactor building a drop of the transfer cask is not postulated. The following discussion for the drop of a shipping cask is considered to be historical and was not revised for a drop of a transfer cask to implement dry cask storage operations at the NMPNS.

Shipping cask handling procedures ensure that a postulated spent fuel drop height of 30 ft is not exceeded. If impact-limiting devices are removed during shipping cask handling within the plant, the 30-ft drop height is still not exceeded.

Therefore, the shipping cask drop accident is not believed to be a credible event, but the accident analysis has been performed.

The analysis of this event was initially performed at 3,467 MWt (104.3 percent of originally rated power) and the radiological consequences have been recalculated for rated 3,467 MWt power conditions. This event does not set reactor operating limits and is not reanalyzed for each reload cycle.

15.7.5.1 Identification of Causes and Frequency Classification

15.7.5.1.1 Identification of Causes

It is assumed that a spent fuel shipping cask containing irradiated fuel assemblies is in the process of being moved with the shipping cask suspended from the crane above the rail car. The fuel assemblies have been out of the reactor for at least 90 days. Through some unspecified failure, the shipping cask is released from the crane and falls 92 ft onto the rail car. Some of the coolant in the outer cask structure may leak from the shipping cask.

15.7.5.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.5.2 Sequence of Events and System Operation

15.7.5.2.1 Sequence of Events

15.7.5.2.2 Identification of Operator Actions

The Operator should ascertain the degree of shipping cask damage, initiate the necessary repairs, and refill the shipping cask coolant to its normal level if coolant has been lost. If unable to fill the shipping cask, water should be sprayed on the outside of the shipping cask to keep the fuel cool.

The Operator should initiate the SGTS to assure that any radioactive release will be filtered even though the consequences of an unfiltered, ground level release are shown in this analysis to be a small fraction of the guidelines of 10CFR100.

15.7.5.2.3 System Operation

It is assumed that if the coolant is lost from the external cask shield the Operator will establish forced cooling of the shipping cask by introducing water into the outer structure annulus or by spraying water on the shipping cask exterior surface. Maintaining the cask in a cool condition will, therefore, ensure no fuel damage as a result of a temperature increase due to decay heat.

15.7.5.2.4 Effects of Single Failures and Operator Errors

Certainly these systems are single-failure proof. Therefore, further failures will not negate indicated Operator actions. See Appendix 15A for further details.

15.7.5.3 Core and System Performance

The event considered will not have a direct impact on the continued normal operation of the reactor system. Containment integrity will not be affected by the postulated event. If the event does require reactor shutdown, normal shutdown cooling and auxiliary system cooling have been previously evaluated for greater demand and more severe situations.

15.7.5.4 Barrier Performance

15.7.5.4.1 Cask Damage

A fuel shipping cask drop of 92 ft would exceed the design basis drop of 30 ft. However, the 30-ft design basis drop is onto a nonyielding surface. The 92-ft drop onto the rail car is considered a drop onto a yielding surface.

Since this accident occurs outside the normal barriers (RCPB and drywell), this section is not directly applicable.

15.7.5.4.2 Releases From the Cask Handling Area

Releases from the cask handling area are conservatively assumed to be instantaneously exhausted to the environment via the reactor building vent, at the normal ventilation exhaust rate,

although the SGTS system is capable of providing filtration and an elevated release path.

15.7.5.5 Radiological Consequences

The radiological analysis provided for this accident is based on conservative assumptions considered to be acceptable to the NRC for the purposes of determining adequacy of the plant design to meet 10CFR100 guidelines.

Several Chapter 15 accident sections include both a design basis and a realistic basis analysis. The realistic basis analysis for such SAR sections is not updated beyond FSAR Amendment 28. The exposures calculated using design basis assumptions are so far below the allowable limits of 10CFR100, that no realistic analysis of the radiological consequences of this accident is provided.

The fission product inventory in the maximum number of fuel rods in the shipping cask is based on 1,000 days of continuous operation at 3,536 MWt to establish a maximum equilibrium inventory in the core. A 90-day period for decay from the above power condition is assumed to account for storage in the fuel pool before shipping cask loading. The isotopes Kr-85 and I-131 are the only ones considered, as the 90-day decay time reduces all other noble gases and iodines to insignificant levels.

15.7.5.5.1 Analysis

Specific parametric values used in this evaluation are presented in Table 15.7-16. This analysis is based on SRP 15.7.5, Revision 2.

<u>Fission Product Release from Fuel</u>

The drop of a loaded spent fuel shipping cask is assumed to cause cladding damage to all 24 assemblies in the shipping cask.

The fuel activity is adjusted by a peaking factor of 1.5 and by the ratio of fuel assemblies in the cask (24) to total fuel assemblies in the core (764). The result is the activity of the fuel in the shipping cask.

The activity released from the damaged fuel assemblies is assumed to be 30 percent of the Kr-85 activity and 12 percent I-131 activity in the shipping cask.

<u>Fission Product Transport to the Environment</u>

The transport pathway consists of leakage from the cask to the reactor building atmosphere and release to the environment via the radwaste/reactor building vent. All of the Kr-85 and 10 percent of the iodine released from the fuel is assumed to become airborne.

The instantaneous release of the Kr-85 activity does not yield a trip of the reactor building discharge plenum radiation monitors.

Therefore, no credit is taken for SGTS operation though manual initiation of SGTS would lower the releases.

The release of activity to the environment is 1.48+04 Ci of Kr-85 and 2.32+01 Ci of I-131.

<u>Results</u>

Calculated exposures for this analysis are presented in Table 15.7-17 and are a small fraction of the guidelines of 10CFR100.

Control room doses for exposures for and beyond the 2-hr release duration are a small fraction of the limit of GDC 19.

15.7.6 References

1. Technical Specification Amendment No. 101, dated February 11, 2002.
2. GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GSTAR II), NEDC-33270P, Revision 5, May 2013

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TABLE 15.7-1

SEQUENCE OF EVENTS FOR MAIN CONDENSER
GAS TREATMENT SYSTEM FAILURE

Approximate Elapsed Time	Event
0 sec	Event begins - system fails
0 sec	Noble gases are released
<1 min	OFG system low flow alarms alert plant personnel
<60 min	Operator actions begin with: 1. Initiation of appropriate system isolations 2. Manual scram actuation

TABLE 15.7-2

GASEOUS RADWASTE SYSTEM FAILURE
PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Assumptions
I. Data and assumptions used to estimate radioactive source for postulated accidents	
A. Power level	3,536 MWt
B. Activity before the accident	Table 15.7-3
II. Data and assumptions used to estimate activity released	
A. Percentage of noble gas activity released	100
B. Duration of release	2 hr
C. Charcoal adsorber bed hold-up times	
1. Xe	178 days
2. Kr	278 hr
D. Condenser air in-leakage	6 scfm
III. Dispersion Data	
A. X/Q for EAB	1.90-04 m/sec^3
IV. Dose Data	
A. Method of dose calculation	IONEXCHANGER and DRAGON codes
B. Breathing rate	3.47-04 m^3/sec
C. Time of exposure	2 hr
D. Doses	Table 15.7-5

NOTE: 1.90-04 = 1.90×10^{-4}

TABLE 15.7-3

ACTIVITY IN OFFGAS SYSTEM
AT TIME OF ACCIDENT

Isotope	Activity* (uCi/cc)
Kr-83m	3.50+00
Kr-85m	7.14+00
Kr-85	2.50-02
Kr-87	1.87+01
Kr-88	2.25+01
Kr-89	2.25-01
Xe-131m	1.87-02
Xe-133m	3.64-01
Xe-133	1.02+01
Xe-135m	8.37+00
Xe-135	2.62+01
Xe-137	8.37-01
Xe-138	2.62+01

* Activity per cc of the offgas after 30-min delay.

NOTE: 3.50+00 = 3.50×10^{0}

TABLE 15.7-4

GASEOUS RADWASTE SYSTEM FAILURE SYSTEM RUPTURE
(DESIGN BASIS ANALYSIS)
FISSION PRODUCT RELEASE TO ENVIRONMENT

Isotope	Activity Released (Ci)
Kr-83m	1.542+02
Kr-85m	5.834+02
Kr-85	8.543+01
Kr-87	6.897+02
Kr-88	1.304+03
Kr-89	5.547+02
Kr-90	5.716+00
Xe-131m	7.887+01
Xe-133m	2.87+02
Xe-133	1.932+04
Xe-135m	3.382+02
Xe-135	3.944+03
Xe-137	7.804+02
Xe-138	1.111+03
Xe-139	1.973+01

NOTE: 1.542+02 = $1.542 \times 10^{+2}$

TABLE 15.7-5

GASEOUS RADWASTE SYSTEM FAILURE SYSTEM RUPTURE
(DESIGN BASIS ANALYSIS)
OFFSITE RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Thyroid Dose (Rem)	Beta Dose (Rem)
Exclusion area boundary	3.92-01	0.0	3.12-01

NOTE: 3.92-01 = 3.92×10^{-1}

TABLE 15.7-6

LIQUID RADWASTE TANKS FAILURE
PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS

		Assumptions
I.	Data and assumptions used to estimate radioactive sources from postulated accidents	
	A. Power level	4,068 MWt
	B. Condensate storage tank inventory	Table 15.7-7
II.	All other pertinent data and assumptions	
	A. Dilution factor	45.3
	B. Transit time	29.6 hr
III.	Concentration data	
	A. Diluted/decayed activities (concentrations at water intake)	Table 15.7-8
	B. Fractions of MPC	Table 15.7-8

TABLE 15.7-7

LIQUID RADWASTE SYSTEM FAILURE ANALYSIS
CONDENSATE STORAGE TANK INVENTORY

Isotope	Activity (uCi/cc)	Isotope	Activity (uCi/cc)
Br-83	3.39-5	Te-131	7.27-10
Br-84	4.92-5	Te-132	5.03-6
Br-85	1.08-5	Ba-137m	1.64-8
I-131	2.60-5	Ba-139	1.56-5
I-132	3.12-4	Ba-140	9.00-7
I-133	2.59-4	Ba-141	1.43-5
I-134	7.21-4	Ba-142	1.23-5
I-135	2.75-4	La-140	1.29-9
Rb-89	8.75-6	La-141	2.32-7
Cs-134	8.00-8	La-142	1.71-6
Cs-136	5.57-8	Ce-141	8.41-9
Cs-137	1.20-7	Ce-143	8.39-9
Cs-138	8.69-5	Ce-144	3.50-9
Sr-89	3.10-7	Pr-143	1.10-8
Sr-90	2.39-8	Pr-144	6.33-10
Sr-91	6.86-6	Nd-147	1.40-9
Sr-92	1.10-5	Pm-147	3.51-15
Y-90	2.15-11	Na-24	4.08-7
Y-91m	2.73-7	P-32	7.90-9
Y-91	1.11-8	Cr-51	2.40-7
Y-92	2.05-6	Mn-54	4.15-9
Y-93	1.19-6	Mn-56	4.89-6
Zr-95	4.13-9	Fe-55	3.90-8
Zr-97	3.32-9	Fe-59	8.00-9
Nb-95m	5.45-14	Co-58	5.06-7
Nb-95	4.34-9	Co-60	5.00-8
Nb-97m	3.10-9	Ni-65	2.93-8
Nb-97	1.13-9	Cu-64	1.29-6
Mo-99	2.23-6	Zn-65	7.90-9
Tc-99m	2.81-5	Zn-69m	8.17-8
Tc-101	3.04-5	Ag-110m	6.00-9
Ru-103	5.47-9	Ag-110	1.20-10
Ru-105	6.19-7	W-187	2.99-7
Ru-106	8.40-10	Nb-98	1.40-6
Rh-103m	3.19-10	Tc-104	2.72-5
Rh-105m	6.15-7	Np-239	2.40-5
Rh-106	8.39-10		
Te-129m	1.10-8		
Te-129	5.39-10		
Te-131m	2.79-8		

NOTE: 8.39-10 = 8.39×10^{-10}

TABLE 15.7-8

LIQUID RADWASTE TANKS FAILURE
FINAL CONCENTRATION AND FRACTION OF MPC
AT WATER INTAKE

Isotope	Concentration (uCi/cc)	Fraction of MPC
Br-83	5.2-12	1.7-06
I-131	8.2-08	2.7-01
I-132	1.3-08	1.6-03
I-133	3.4-07	3.4-01
I-135	4.3-08	1.1-02
Sr-89	5.1-10	1.7-04
Sr-90	3.6-11	1.2-04
Sr-91	2.0-09	4.1-05
Sr-92	1.6-11	2.6-07
Y-90	2.5-11	1.2-06
Y-91m	1.3-09	4.4-07
Y-91	2.0-10	6.7-06
Y-92	3.4-10	5.6-06
Y-93	6.0-10	2.0-05
Zr-95	1.5-11	2.6-07
Zr-97	1.1-12	5.6-08
Nb-95	1.6-11	1.6-07
Nb-97m	1.1-12	N/A
Nb-97	1.2-12	1.3-09
Mo-99	6.3-09	1.6-04
Tc-99m	2.4-09	7.9-07
Ru-103	2.1-11	2.6-07
Ru-105	7.1-12	7.1-08
Ru-106	3.3-12	3.3-07
Rh-103m	2.1-11	2.1-09
Rh-105m	7.2-12	N/A
Rh-105	1.4-13	1.4-09
Rh-106	3.3-12	N/A
Te-129m	4.2-11	2.1-06
Te-129	4.2-11	5.2-08
Te-131m	5.4-11	1.4-06
Te-131	1.1-11	N/A
Te-132	1.3-10	6.6-06
Cs-134	3.1-10	3.5-05
Cs-136	2.0-10	3.4-06
Cs-137	4.7-10	2.3-05
Ba-137m	4.3-10	N/A
Ba-140	2.0-09	9.8-05
La-140	1.4-09	6.8-05
La-141	8.6-12	2.9-06
Ce-141	5.7-11	6.3-07
Ce-143	5.4-12	1.3-07

Table 15.7-8 (Cont'd.)

Isotope	Concentration (uCi/cc)	Fraction of MPC
Ce-144	1.4-11	1.4-06
Pr-143	1.3-11	2.5-07
Pr-144	1.4-11	N/A
Nd-147	1.5-12	2.6-08
Na-24	4.0-10	1.3-05
P-32	2.9-11	1.4-06
Cr-51	8.7-10	4.4-07
Mn-54	1.6-11	1.6-07
Mn-56	6.2-12	6.2-08
Fe-55	1.5-10	1.9-07
Fe-59	3.1-11	6.1-07
Co-58	1.8-10	2.0-06
Co-60	2.0-10	6.5-06
Cu-64	9.3-10	4.6-06
Zn-65	3.0-11	3.0-07
Zn-69m	2.2-11	3.7-07
Ag-110m	1.8-12	6.0-08
Ag-110	3.6-14	N/A
W-187	2.3-10	3.8-06
Np-239	2.8-08	2.8-04

NOTE: 5.2-12=5.2x10^{-12}

TABLE 15.7-9

FUEL HANDLING ACCIDENT
PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Assumptions	Realistic Basis Assumptions
1. Data and assumptions used to estimate radioactive source from postulated accidents		
a. Power level	4,067 MWt	3,489 MWt
b. Radial peaking factor	1.8	1.5
c. No. fuel rods in core	N/A	48,132
d. No. fuel rods damaged	2 full assemblies	124
e. Release of activity from fuel by nuclide	5% noble gas 10% Kr-85 5% halogens 8% I-131	1.8% noble gas 0.32% halogens
f. No. fuel assemblies in core	764	
2. Data and assumptions used to estimate activity released		
a. Reactor building vent release		
(1) Duration	Instantaneous	30 sec
(2) Release rate	N/A	51.7 vol/day
(3) Filter efficiencies	no filter	no filter
(4) Mixing volume	N/A	3.9+06 ft^3
b. Main stack release (via SGTS)		
(1) Duration	NA	119.5 min
(2) Release rate	NA	2.0+05 vol/day
(3) Filter efficiencies	NA	0.99
(4) Reactor building volume	NA	3.9+06 ft^3
3. Dispersion Data		
a. Reactor building vent release (sec/m^3)		
(1) Exclusion area boundary	1.19-04	2.19-05
(2) Low population zone	1.62-05	6.48-06
(3) Control room air intake	1.09-03	2.13-04

TABLE 15.7-9 (Cont'd.)

	Design Basis Assumptions	Realistic Basis Assumptions
3. Dispersion Data (cont'd.)		
b. Main stack release (sec/m^3)		
(1) Exclusion area boundary	NA	1.16-07
(2) Low population zone	NA	4.32-07
(3) Control room air intake	NA	8.10-05
4. Control Room Data		
a. Safety zone free air volume	3.81+05 ft^3	4.8+05 ft^3
b. Intake flow rate	N/A	1,500 cfm
c. Recirculation flow rate	N/A	750 cfm
d. Intake and recirculation efficiencies	N/A	99% halogens
5. Dose Data		
a. Method for dose calculation	RG 1.183	RG 1.25
b. Computer code for dose calculations	N/A	DRAGON4
c. Breathing rate	3.5-04 m^3/sec	3.47-04 m^3/sec
d. Doses	Table 15.7-12	Table 15.7-15

NOTE: 1.90-04 = 1.90 x 10^{-4}

TABLE 15.7-10

FUEL HANDLING ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASED TO FUEL POOL

Isotope	Activity (Ci)
Kr-85m	1.59E+02
Kr-85	7.53E+02
Kr-87	2.69E-02
Kr-88	5.01E+01
Xe-131m	2.90E+02
Xe-133m	1.42E+03
Xe-133	4.88E+04
Xe-135	1.18E+04
I-131	3.84E+04
I-132	3.07E+04
I-133	2.37E+04
I-135	4.01E+03
Rb-86	1.62E+02
Cs-134	1.68E+04
Cs-136	4.97E+03
Cs-137	1.00E+04

NOTES:

1. 1.59E+02 = $1.59 \times 10^{+2}$.

TABLE 15.7-11

FUEL HANDLING ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT

Isotope	Activity (Ci)
Kr-85m	1.59E+02
Kr-85	7.53E+02
Kr-87	2.69E-02
Kr-88	5.01E+01
Xe-131m	2.90E+02
Xe-133m	1.42E+03
Xe-133	4.88E+04
Xe-135	1.18E+04
I-131	2.19E+02
I-132	1.76E+02
I-133	1.35E+02
I-135	2.28E+01

NOTES:

1. 1.59E+02 = $1.59 \times 10^{+2}$.

TABLE 15.7-12

FUEL HANDLING ACCIDENT (DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

	Total Effective Dose Equivalent (Rem)
Exclusion area boundary (2 hr)	4.5-01
Low population zone (2 hr)	6.1-02
Control room	3.2+00

NOTE: 4.5-01 = 4.5×10^{-1}.

TABLE 15.7-13

FUEL HANDLING ACCIDENT
(REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN THE REACTOR BUILDING

Isotope	Activity (Ci)
I-129	3.76-05
I-131	1.06+03
I-132	1.36+03
I-133	1.09+03
I-134	5.95-05
I-135	1.80+02
I-136	3.02-11
Br-83	1.42-01
Br-84	1.28-11
Br-85	7.90-12
Br-87	1.33-11
Kr-83m	3.18+00
Kr-85m	4.03+01
Kr-85	7.33+01
Kr-87	6.26-03
Kr-88	1.17+01
Kr-89	1.52-10
Xe-131m	3.84+01
Xe-133m	5.05+02
Xe-133	1.29+04
Xe-135m	1.54+02
Xe-135	2.96+03
Xe-137	3.28-10
Xe-138	3.11-10

NOTE: 3.76-05 = 3.76×10^{-5}

TABLE 15.7-14

FUEL HANDLING ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT

Isotope	Activity (Ci)
I-129	1.02-08
I-131	2.94-01
I-132	3.77-01
I-133	3.02-01
I-134	1.65-08
I-135	4.99-02
Br-83	3.93+00
Br-84	0
Kr-83m	3.32+00
Kr-85m	4.10+01
Kr-85	7.46+01
Kr-87	6.37-03
Kr-88	1.19+01
Xe-131m	3.92+01
Xe-133m	5.15+02
Xe-133	1.31+04
Xe-135m	4.15+02
Xe-135	2.35+03

NOTE: 1.02-08 = 1.02×10^{-8}

TABLE 15.7-15

FUEL HANDLING ACCIDENT (REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole-Body Dose (Rem)	Thyroid Dose (Rem)	Beta Dose (Rem)
Exclusion area boundary	2.33-04	2.84-03	3.10-04
Low population zone	2.59-04	8.69-04	3.37-04
Control room	1.57-03	2.11-04	4.53-02

NOTE: 2.33-04 = $2.33x10^{-4}$

TABLE 15.7-16

SHIPPING CASK DROP ACCIDENT PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS

	Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents	
A. Power level, MWt	3,536
B. Radial peaking factor	1.5
C. Fuel damage	24 assemblies
D. Fuel in core	764 assemblies
E. Release of activity from fuel	
1. Kr-85	30%
2. I-131	12%
II. Data and assumptions used to estimate activity released	
A. Duration, hours	2
B. Reactor building volume	3.9+06 ft^3
C. Release rate	138 vol/day
D. Filtration	None
E. Release of activity to environment	
1. Kr-85	100%
2. I-131	10%
III. Dispersion data	
A. X/Q to EAB	1.9-04 sec/m^3
B. X/Q to LPZ	1.78-05
C. X/Q to control room	2.13-04
IV. Control room data	
A. Free air volume	3.81+05 ft^3
B. Intake flow rate	0-30 sec 1650 cfm 30 sec - 20 min 2750 cfm 20 min - 720 hr 1650 cfm
C. Recirculation flow rate	675 cfm
D. Filter efficiencies	99% halogens*

TABLE 15.7-16 (Cont'd.)

	Assumptions
V. Dose Data	
A. Method for dose calculations	RG 1.25
B. Computer code for dose calculations	DRAGON
C. Breathing rate, all locations	3.47-04 m^3/sec
D. Doses	Table 15.7-17

NOTE: 3.9+06 = 3.9×10^6

* Following a 30-sec delay in activating the control room air intake and recirculation filter.

TABLE 15.7-17

SHIPPING CASK DROP ACCIDENT (DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole-Body Dose (Rem)	Thyroid Dose (Rem)	Beta Dose (Rem)
Exclusion area boundary (2 hr)	1.9-03	2.3+00	1.5-01
Low population zone (2 hr)	1.8-4	2.1-01	1.4-02
Control room*	1.8-04	1.1-01	2.5-01

* Control room quoted doses are the maximum values calculated for a 30-day time period.

NOTE: 1.9-03 = 1.9×10^{-3}

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

15.8.1 Requirements

The issue of a postulated failure to scram the reactor following an anticipated transient, i.e., an ATWS, has been under consideration by the NRC. As a result of its assessment, the NRC has required additional plant modifications for the BWR in 10CFR50.62. The NRC has determined that the current risk from an ATWS event is acceptably small; therefore, any plant modifications would only be required for long-term resolution of the ATWS issue, and such modifications need not satisfy the requirements for a design basis event.

The GNF2 Amendment 22 Compliance to GSTAR II (Reference 11) requires a plant-specific demonstration that the limiting ATWS event response is within the ATWS acceptable criteria. For Unit 2, the calculated ATWS results of record have sufficient margin to the vessel overpressure and suppression pool temperature limit to not require explicit analysis and allowed per Reference 11. Therefore, while it is recognized that the GNF2 New Fuel Introduction (NFI) will impact the ATWS results, Unit 2 is able to demonstrate compliance to the ATWS acceptable criteria for ATWS events with GNF2 fuel.

15.8.2 Plant Capabilities

The Unit 2 design uses diverse, redundant, and reliable scram systems which include the normal scram systems plus the electrically-diverse alternate rod insertion (ARI) system. Each of these systems is frequently tested and would insert the control rods even if multiple component failures should occur, thus making the probability of an ATWS event extremely remote.

The ATWS RPT feature prevents reactor vessel overpressure and possible short-term fuel damage for the most limiting postulated ATWS event. Subsequent to an ATWS event for which the ARI system fails to insert the control rods, the long-term shutdown of the reactor can be accomplished by either manual insertion of the control rods, or simultaneous two-pump injection of sodium pentaborate solution into the vessel.

The features described in Section 15.8.3 have been incorporated in Unit 2. These features exceed the requirements of 10CFR50.62 and are consistent with the Alternate 3A features described in References 1 and 2 in Section 15.8.5.

For operation at power uprate conditions (3,988 MWt), the capability of the ATWS design features to mitigate the consequences of a postulated ATWS event has been confirmed. The limiting ATWS events were reanalyzed for uprated power conditions and with revised setpoints for pertinent functions. These analyses demonstrated that acceptable results are maintained for

the EPU condition with the existing NMP2 design, as documented in Reference 10.

Adequacy of the ATWS design features has also been evaluated under applicable equipment OOS options. These case studies are documented in appendices as follows:

Appendix 15B	Recirculation System Single Loop Operation
Appendix 15C	Two Safety Relief Valves Out of Service
Appendix 15D	One Main Steam Isolation Valve Out of Service
Appendix 15G	Maximum Extended Load Line Limit Operation

15.8.3 Equipment Description

This section describes the equipment and control logic added or modified exclusively for ATWS prevention or mitigation. The description covers design and functional performance and provides references that contain more detailed information. The dynamic and environmental qualifications of the related equipment are described in Sections 3.9, 3.10, and 3.11.

15.8.3.1 Redundant Reactivity Control System

The redundant reactivity control system (RRCS) determines that a transient is underway that exceeds expected operating parameters and immediately activates ATWS prevention equipment. After deciding that a controlled shutdown is not occurring, the RRCS activates ATWS mitigation equipment. The RRCS uses transient detection sensors for high vessel dome pressure and low vessel water level to initiate ARI and RPT. The actuation logic also includes APRM neutron flux "not downscale" to initiate SLC injection and feedwater runback.

The RRCS consists of two completely redundant divisions. Each division is initiated automatically by the ATWS detection sensors, which are independent of the RPS sensors, or manually by switches that require the same type of Operator actions as manual scram.

Additional information on the RRCS is contained in Sections 7.1, 7.2, 7.4, 7.6, and 7.7.

15.8.3.2 Alternate Rod Injection

ARI is designed to provide a parallel path for actuation of the scram valves, which results in control rod insertion. ARI consists of the redundant valves on the scram valve pilot air headers that are actuated automatically by the RRCS logic or manually by the Operator in the main control room. The RRCS logic is designed so that successful ARI performance will avoid

subsequent ATWS mitigation action (feedwater runback and SLCS initiation).

Additional information on the ARI system is contained in Sections 7.1 and 7.2.

15.8.3.3 Recirculation Pump Trip

The recirculation pump motors are tripped by the RRCS logic. The purpose of the RPT is to reduce core flow and create core voids to decrease power generation, thus limiting any power or pressure disturbance. The RPT function is single failure proof and is provided with in-service test capability (except for the action of the final breakers).

Additional information on the RPT function of the RRCS is contained in Sections 7.1 and 7.7.

15.8.3.4 Feedwater Runback

Upon the receipt of a high-pressure signal from the RRCS and after a specified time delay, if core power is not reduced as evidenced by the APRMs reading downscale, feedwater flow would be automatically limited by the RRCS, thereby reducing power and steam discharge to the suppression pool. After a time delay, the system allows manual operation by the Operator to increase the feedwater flow if needed.

Additional information on the feedwater runback function of the RRCS is contained in Sections 7.1 and 7.7.

15.8.3.5 Standby Liquid Control System

The SLCS is automatically actuated by the RRCS or manually initiated by an Operator in the main control room upon indication of a failure to scram and in accordance with plant operating procedures. The system is designed to inject sodium pentaborate solution through the HPCS sparger. Simultaneous operation of the two SLCS pumps provides adequate margin for controlled shutdown even if no rods insert. The system can be periodically tested without affecting its capability to respond to an actuation signal.

Additional information on the SLCS is provided in Sections 3.9, 7.1, 7.4, and 9.3.5.

15.8.3.6 Scram Discharge Volume

The SDV of the CRD system is designed to minimize the potential for a common-mode failure of the scram function. Redundant instrument-volume water level sensors for the CRDs and instrument line piping ensure the availability of sufficient capacity to receive water from a full reactor scram. The design employs

redundant Class 1E sensors and redundant vent and drain valves. Performance of the safety functions is ensured in the event of a single active failure or the bypass of the sensors during plant operation.

Additional information on the SDV is contained in Sections 4.6 and 7.2.

15.8.4 Operator Actions

Operator action during ATWS events will be directed using plant EOPs. These actions include the following, of which items 1 through 3 occur automatically due to RRCS:

1. Initiating boron injection with the SLCS or other systems if SLC is inoperable.

2. Initiation of ARI if ARI has not initiated.

3. Tripping of the recirculation pumps.

4. Manually inserting control rods.

5. Lowering RPV water level until either:

    a. power is below 4 percent, or

    b. containment heatup is terminated, or

    c. RPV water level reaches top of active fuel (TAF).

6. Manually inhibiting ADS (References 2, 3, 4 and 5).

Lowering RPV water level reduces natural circulation in the core and thus lowers reactor power. Lowering RPV water level thus reduces the energy input to the containment. This is acceptable because RPV water level is never intentionally lowered below TAF. Keeping the fuel covered ensures that adequate core cooling is maintained and, as a result, no fuel damage will occur.

Manually inhibiting ADS is required to prevent rapid and uncontrolled injection of large amounts of relatively cold unborated water from low-pressure injection systems as RPV pressure decreases to and below the shutoff heads of the pumps. Such an occurrence would quickly dilute in-core boron concentration and reduce reactor coolant temperature. Unit 2 has installed ADS logic inhibit switches to accomplish this function. Unit 2 plant-specific ATWS analysis takes credit for Operator action to inhibit ADS.

15.8.5 References

1. Assessment of BWR Mitigation of ATWS. NUREG-0460 Alternate 3, Volume 4 (for comment). General Electric, 1979, NEDE-24222.

2. Design Analysis and SAR Inputs for ATWS Performance and Standby Liquid Control System. Nine Mile Point Unit 2 Plant, General Electric, 1982, NEDE-22013.

3. GE letter to NMPC, NMP2-6309, September 24, 1984.

4. GE letter to NMPC, NMP2-6769, April 26, 1985.

5. GE letter to NMPC, NMP2-6835, June 7, 1985.

6. GE Hitachi Nuclear Energy, Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate, NEDC-33351P, May 2009.

7. Blank.

8. TR-MODS-11-1, “Westinghouse Engineering Report Jet Pump Inlet Mixer Form, Fit and Function,” April 2012.

9. ECP-14-000152, Slip Joint Clamp Installation.

10. Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Maximum Extended Load Line Limit Analysis Plus, NEDC-33576P, November 2013.

11. GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II), NEDC-33270P, Revision 5, May 2013.