15.0 ACCIDENT ANALYSES

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 calculational results contained in Appendix 15E are applicable to Cycle 1 of both units. Some of the accident analyses are cycle dependent and must be performed for each reload core. This chapter 15 contains the reload analysis results. The results of these reload analyses for the current cycles are documented in Appendices 15C and 15D for Units 1 and 2, respectively.

Appendix 15E contains information and analytical results for non-limiting events for the initial cycles for Units 1 and 2. Note that since the data in Appendix 15E is for the initial cycles for Units 1 and 2, the values for key parameters/variables do not represent the actual values if these events were to occur for the current cycles for Units 1 and 2. However the data and figures in Appendix 15E do show qualitative behavior of the non-limiting events.

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

15.0.1 ANALYTICAL OBJECTIVE

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

15.0.2 ANALYTICAL CATEGORIES

Transient and accident events contained in this report are discussed in individual categories as required by Reference 15.0-1. The results of the events are summarized in Tables 15C.0-1 and 15D.0-1 for the current cycles for Units 1 and 2. Table 15E.0-1 contains results of analyses that are for non-limiting events for the initial cycles for Units 1 and 2. Each event 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 increased core flow disturbance events. Increased core flow 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 carryover to the main turbine, feedwater turbines, etc.

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. Radioactive Release from a Subsystem or Component:

   Loss of integrity of a component which contains radioactivity is postulated.

8. Anticipated Transients Without Scram:

   In order to determine the capability of plant design to accommodate an extremely low probability event, a multi-system maloperation plus multi-single active component failures (SACF) situation is postulated.

15.0.3 EVENT EVALUATION

15.0.3.1 Identification of Causes and Frequency Classification

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

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

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

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

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

4. Normal operation - operations of high frequency are not discussed here but are examined along with (1), (2), and (3) in the nuclear systems operational analyses in Appendix 15A.

15.0.3.1 .1 Unacceptable Results for Incidents of Moderate Frequency (Anticipated(Expected) Operational Transients)

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

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

2. Reactor operation induced fuel cladding failure.

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

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

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

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

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

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

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

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

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

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

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

2. Failure of fuel cladding which 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.

15.0.3.2 Sequence of Events and Systems Operations

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

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

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

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

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

5. The operation of engineered safety systems that is required.

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

15.0.3.2.1 Single Failures or Operator Errors

15.0.3.2.1.1 General

For each event, the effect of single failures and/or operator errors is discussed. A plant operational analysis was performed prior to the initial startup of the units (see Appendix 15A). Although this information is historical in nature, it provided initial independent evaluation of the adequacy of systems as they related to the events under study.

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

15.0.3.2.1.2 Initiating Event Analysis

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

   or

2. The undesired starting or stopping of any single component

   or

3. The 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 single operator error is the set of actions that is a direct consequence of a single erroneous decision. The set of actions is limited as follows:

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 single operator errors are as follows:

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

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

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

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

15.0.3.2.1.3 Single Active Component Failure or Single Operator Failure Analysis

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

   or

2. Any single operator error where operator errors are defined as in Subsection 15.0.3.2.1.2.

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 safety limits 1 and 2 (Subsection 4.4.1.4) for incidents of moderate frequency is verified statistically with consideration given to date, calculation, manufacturing, and operating uncertainties. An acceptable criterion has been established to be that 99.9% of the fuel rods in the core would not be expected to experience boiling transition (see Reference 15.0-3). This criterion is met by demonstrating that incidents of moderate frequency do not result in a minimum critical power ratio (MCPR) less than the safety limit established for the current cycle. The reactor steady state CPR operating limit is derived by determining the decrease in MCPR for the most limiting event. All other events result in smaller MCPR decreases and are not reviewed in depth in this chapter. The MCPR during significant abnormal events is calculated using a transient core heat transfer analysis computer program. The computer program is based on a multinode, single channel thermal hydraulic model which requires simultaneous solution of the partial differential equations for the conservation of mass, energy, and momentum in the bundle, and which accounts for axial variation in power generation. The primary inputs to the model include a physical description of the bundle, and channel inlet flow and enthalpy, pressure and power generation as functions of time.

The methods for modeling and analyzing Units 1 and 2 are described in References 15.0-8 through 15.0-10. Unit 1 analyses are performed using the methods in References 15.0-11 and 15-12. Unit 2 analyses are performed using the methods in 15.0-13. Determination of the steady-state operating limit is accomplished as follows:

1. The change in critical power ratio (ΔCPR) is calculated for each event.

2. The ΔCPR value is then added to the safety limit CPR value to result in the event based MCPR. The current cycle MCPR safety limits are given in Tables 15C.0-3 and 15D.0-3 for Units 1 and 2.

The operating limit MCPR is the maximum value of the event MCPRs calculated from the transient analysis. A set of plots of the MCPR Operating Limits (MCPROLs) as a function of core flow and as a function of core power is prepared. Separate plots are prepared that consider core exposure, operability of Recirculation Pump Trip, operability of Main Turbine Bypass, average scram speed, and single loop operation. These plots are prepared prior to the start of a new cycle and issued in the form of a Core Operating Limits Report (COLR). The COLR is prepared in accordance with the SSES Technical Specifications. The COLR for the current cycle of each unit is contained within the Technical Requirements Manual for each unit (see FSAR section 16.3).

Maintaining the CPR operating limit at or above this operating limit assures that the safety limit CPR is never violated for incidents of moderate frequency.

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

These correlations are substantiated by fuel rod failure tests and are discussed in Section 4.4 and Section 6.3.

15.0.3.3.2 Input Parameters and Initial Conditions for Analyzed Events

In general the limiting events analyzed within this section have values for input parameters and initial conditions as specified in Tables 15C.0-2 and 15D.0-2 for Units 1 and 2. These tables include the current conditions for power uprate. Analyses which assume data inputs different than the power uprate values are designated accordingly in the appropriate event discussion. Table 15E.0-2 provides the initial conditions used for the analysis of the non-limiting events for the initial cycle for Units 1 and 2.

15.0.3.3.3 Initial Power/Flow Operating Constraints

The analysis basis for most of the transient safety analyses at a core flow of 108 Mlbs/hr and a power given in Tables 15C.0-2 and 15D.0-2. However to assure that thermal margins are maintained over the entire power/flow operational space, the anticipated operational occurrences were analyzed over a range of power and flow conditions for the current cycles. In addition, single loop operation was analyzed for each of the anticipated operational occurrences and accidents. It was determined that for each anticipated event and the ASME overpressure analysis, the two loop results bound the results from single loop operation. Explicit analyses of LOCA and the pump seizure in single loop operation were also performed.

Figure 15E.0-1 is a typical power/flow map for a BWR. Power/flow maps for the current cycles for Units 1 and 2 are included in their COLR. The COLR for the current cycle of each Unit is contained within the Technical Requirements Manual for each Unit (see FSAR Section 16.3).

Referring to Figure 15E.0-1, the apex of the bounded power/flow map is point A, the upper bound is the design flow control line (105%, rod line A-D'), the lower bound is the zero power line H'-J', the right bound is the rated pump speed line A-H', and the left bound is either the minimum pump speed line D-J or the natural circulation line D'-J'.

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

Any other constraint which may truncate the bounded power/flow map must be observed, such as the recirculation valve and pump cavitation regions, the licensed power limit and other restrictions based on pressure and thermal margin criteria. For instance, if the licensed power is 100% , the power/flow map is truncated by the line B-C on Figure 15E.0-1 and reactor operation must be confined within the boundary B-C-D'-J'-J-L-K-B. If the maximum operating power level has to be limited, such as point F, the operating bounds would be F-M-D’-J’-L-K-F, provided that the MCPR operating limit is not violated by operating in this space. Similarly, if operating limitations are imposed by the analysis of a transient with an initial operating basis at point A, the power/flow boundary for 100% licensed power would be B-C-D'-J'-J-L-K-B. This power/flow boundary would be truncated by the MCPR operating limit, (for which there is no direct correlation to a line on the power/flow boundary), and by the constraints imposed by the safety analysis.

Consequently, the upper operating power/flow limit of a reactor is predicated on the operating basis of the analysis.

This boundary may be truncated by the licensed power and other operating restrictions.

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

15.0.3.3.4 Results

A summary of the results of analytical evaluations are provided for each event. In addition, critical parameters are shown in Tables 15C.0-1 and 15D.0-1 for Units 1 and 2.

15.0.3.5 Barrier Performance

This section primarily evaluates the performance of the Reactor Coolant Pressure Boundary (RCPB) and the Containment System during transients and accidents.

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

Containment integrity is maintained so long as internal pressures remain below the maximum allowable values. The design internal pressures are as follows:

Drywell (primary containment)	53 psig
Suppression Chamber (primary containment)	53 psig
Secondary Containment	7 in. H_2O

Damage to any of the radioactive material barriers as a result of accident-initiated fluid impingement and jet forces is considered in the other portions of the FSAR where the mechanical design features of systems and components are described. Design basis accidents are used in determining the sizing and strength requirements of the essential nuclear system components. A comparison of the accidents considered in this section with those used in the mechanical design of equipment reveals either that the applicable accidents are the same or that the accident in this section results in less severe stresses than those assumed for mechanical design.

15.0.3.6 Radiological Consequences

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

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

1. The first is based on conservative assumptions considered to be acceptable to the NRC for the purposes of bounding the event and determining the adequacy of the plant design to meet 10 CFR Part 50.67 guidelines. This analysis is referred to as the "design basis analysis."

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.

Atmospheric Dispersion Parameters

Short-term site-specific X/Q's were calculated as described in Section 2.3. For the conservative case, the 0.5 percentile X/Q's were used in the dose calculations. The resultant offsite doses are conservative. For the realistic case, 50 percentile X/Q's were used. The valves are given in Table 2.3-92 and 2.3-105 for Units 1 and 2.

15.0.4 Nuclear Safety Operational Analysis (NSQA) Relationship

Appendix 15A is a comprehensive system-level, qualitative FMEA, relative to all the 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-reference of protective actions is an integral part of this chapter and the appendices.

Contained in Appendix 15A is a summary table which classifies events by frequency only (i.e., not just within a given category such as Decrease in Core Coolant Temperature).

15.0.5 REFERENCES

15.0-1 United States Nuclear Regulatory Commission Regulation Guide 1.70 Revision 2 (Preliminary), September 1975, "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants, Light Water Reactor Edition."

15.0-2 "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application," November 1973, (NEDO-10959 and NEDE-10958).

15.0-3 NUREG-0800, U.S. Nuclear Regulatory Commission Standard Review Plan, Section 4.4 Thermal and Hydraulic Design

15.0-4 Deleted

15.0-5 Deleted

15.0-6 Deleted

15.0-7 Deleted

15.0-8 XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, “Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis,” Exxon Nuclear Company, March 1983.

15.0-9 XN-NF-80-19(P)(A) Volume 3 Revision 2, “Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description,” Exxon Nuclear Company, January 1987.

15.0-10 XN-NF-80-19(P)(A) Volume 4 Revision 1, “Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads,” Exxon Nuclear Company, June 1986.

15.0-11 XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, “XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis,” Exxon Nuclear Company, February 1987.

15.0-12 ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, “COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses,” Advanced Nuclear Fuels Corporation, August 1990.

15.0-13 ANP-10300P-A Revision 1, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios, Framatome, January 2018.