14.0 SAFETY ANALYSIS

14.1 ORGANIZATION AND METHODOLOGY

This chapter presents analytical evaluations of the Nuclear Steam Supply System (NSSS) response to postulated disturbances in process variables and to postulated malfunctions or failure of equipment. These initiating Design Basis Events (DBEs) are postulated and their consequences analyzed despite the precautions which are taken in the design, construction, quality assurance, and plant operation to prevent their occurrence. Such occurrences are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such situations (or to identify the limitations of expected performance), and to assure the health and safety of the public in the event of even the most severe of the hypothetical occurrences analyzed.

All initiating events that required a revision to the methodology or that have been added subsequent to the original Final Safety Analysis Report (FSAR) will contain a brief discussion of the link to the current revision. During the latter stages of the licensing of Unit 1 for initial operation, the phenomenon of fuel densification and its adverse effects on plant operation was discovered (Reference 11). The alternative source term (AST) methodology per 10 CFR 50.67 and Reference 31, was approved by the Nuclear Regulatory Commission (NRC) in Reference 30.

Startup of an Inactive Reactor Coolant Pump (RCP) (Idle Loop Startup) has been eliminated from the Updated Final Safety Analysis Report (UFSAR) as Calvert Cliffs is not presently licensed for part loop operation. Part-length control element assembly (PLCEA) drop and malpositioning has also been eliminated as the PLCEAs have been removed from Calvert Cliffs cores. Excess Feedwater Heat Removal, Reactor Coolant System (RCS) Depressurization, Loss-of-Non-Emergency AC Power (LOAC), Transients Resulting from the Malfunction of one Steam Generator (SG) (Asymmetric SG events), Seized Rotor, Excessive Charging, and Feedline Break events have been subsequently added to the DBEs required by the operating license.

14.1.1 CLASSIFICATION OF TRANSIENTS AND ACCIDENTS

14.1.1.1 Categorization

The assignment of the initiating DBEs to categories is made by collecting events with the same major effect and similar occurrence rates into one of three major groups. The three categories discussed below are structured around the specified acceptance criteria described in Section 14.1.1.2. Table 14.1-1 lists all event groups and initiating events considered in the Calvert Cliffs Unit 1 and Unit 2 Safety Analysis.

a. Anticipated Operational Occurrences

Anticipated operational occurrences (AOOs) are initiating events that may occur during the lifetime of the plant. These events are subdivided into initiating events that require the action of the Reactor Protective System (RPS) and those requiring sufficient initial steady-state thermal margin and/or action of the RPS to prevent exceeding the fuel performance criteria, the RCS pressure upset criteria, and the offsite dose criterion.

b. Postulated Accidents

These initiating events are postulated even though they are not expected to occur during the lifetime of the plant based on the very low probability of occurrence. Nevertheless, such events are analyzed to evaluate the protection afforded by the continual upgrading of the plant design and characteristics, the extensive preventive precautions, and by mechanisms incorporated into the plant design. For all postulated accidents that release radioactivity, the projected offsite dose is shown to be within the acceptance criteria.

c. Postulated Occurrences

Postulated occurrences are events which are not discussed in the above categories, but which could involve a release of radioactivity. For all postulated occurrences that release radioactivity, the projected offsite dose is shown to be within the acceptance criteria.

14.1.1.2 Acceptance Criteria

In order to permit the consistent application of design guidelines to each event, acceptance criteria discussed below have been established for each category.

Anticipated Operational Occurrences are analyzed to demonstrate that the fuel performance, RCS pressure, and offsite dose criteria are not exceeded. Postulated Accidents are analyzed against the RCS Pressure and offsite dose criteria. Postulated Occurrences are analyzed against the offsite dose criteria.

a. Fuel Performance Criterion

The acceptance criteria on fuel performance are the Departure from Nucleate Boiling Ratio (DNBR) and fuel centerline to melt temperature design limit. The steady-state and the transient criterion on clad failure is that the DNBR is greater than or equal to a value consistent with the methods discussed in Reference 47. The steady-state criterion on centerline melt temperature is that the peak fuel centerline temperature is in accordance with Technical Specification 2.1.1.2. For some cases, the reactor power rises rapidly for a very short period of time before the power transient is terminated. For those cases the total energy generated and the corresponding temperature rise at the hot spot are calculated for the duration of the transient to demonstrate that fuel centerline temperatures do not exceed UO_2 melt temperatures.

Section 14.1.4.4 discusses the fuel performance models and associated acceptance criteria.

b. Reactor Coolant System Pressure Criterion

The acceptance criterion on the reactor coolant pressure boundary is the 2750 psia pressure upset limit. This limit is based on 110% of the design pressure of the RCS.

c. Offsite Dose

The acceptance criteria on offsite radiation dose are based on 10 CFR 50.67 criteria. The two-hour site boundary dose limit is 25 REM total effective dose equivalent (TEDE).

d. Loss of Shutdown Margin

Acceptance criteria on loss of shutdown margin have been established to measure the time available to prevent loss of shutdown margin with the RCS cold or cooling. For all reactivity anomalies, the shutdown margin should not go to zero in less than 15 minutes for events initiated in operating Modes 3 through 5 and in less than 30 minutes for events initiated in operating Mode 6 (refueling) following an initiation of the dilution.

14.1.1.3 Section Numbering

The events analyzed in this chapter are titled and numbered as described in Table 14.1-1. For ease of cross-reference with the original FSAR, the table also lists corresponding section numbers for each event.

14.1.2 PLANT CHARACTERISTICS CONSIDERED IN SAFETY ANALYSIS

The following two Sections, 14.1.2.1 and 14.1.2.2, describe the principal parameters and their values used to analyze all the initiating DBEs in this chapter. Specific exceptions for a particular event will be described in the section for each event.

The values of initial conditions and calculated input parameters for each reload core design are compared to the values used to evaluate the UFSAR DBEs. The impact of any variable change on the Safety Analysis is evaluated. If all current cycle values for a particular event are bounded by (conservative with respect to) the UFSAR, no reanalysis is performed.

14.1.2.1 Initial Conditions

The events discussed in this chapter have been analyzed over the normal operating range for the initial conditions that significantly affect the boundary dose, RCS pressure, and fuel performance. These variables include core power level, core power distribution, core coolant inlet temperature, primary system pressure, RCS flow, and secondary system pressure.

The range of initial conditions used in the Safety Analysis is consistent with the Technical Specifications and bounds the space for which the Safety Analysis is valid. For conservatism in the Safety Analysis, all applicable uncertainties are assumed to occur simultaneously in the most adverse direction except as discussed in Section 14.1.4.

14.1.2.2 Input Parameters

The parameters used in the analysis of UFSAR DBEs are consistent with those listed in UFSAR Chapter 3. The following principal input parameters are discussed based on their significance in the Safety Analysis.

a. Doppler Coefficient

   The effective fuel temperature coefficient (FTC) of reactivity (Doppler coefficient) is shown in UFSAR Chapter 3 for first cycle, beginning of cycle (BOC) and end of cycle (EOC) conditions. These curves were adjusted by 15% to conservatively account for uncertainties in determining the actual fuel temperature reactivity effects.

   The effective fuel temperature is discussed in UFSAR Chapter 3. Equivalent fuel temperatures calculated in the S-RELAP5 computer code are used to interpolate Doppler reactivity changes during the transients.

b. Moderator Temperature Coefficient

   A range of Moderator Temperature Coefficient (MTC) of reactivity is considered. These values include all uncertainties and bound the expected equilibrium cycle MTC for all the cycle exposures, power levels, control element assembly (CEA) configurations, and boron concentrations. The most conservative value of the MTC is assumed for each analysis.

The Steam Line Break (SLB) analysis uses an explicit table of reactivity versus moderator density to model moderator feedbacks instead of a single value of MTC. That table is consistent with the assumed stuck CEA. With the large moderator density variations that occur during an SLB event, an explicit moderator reactivity feedback is necessary.

c. Shutdown CEA Reactivity

The transient shutdown CEA reactivity (SCRAM) is dependent on CEA worth available on reactor trip, axial power distribution, the position of the regulating CEAs, and time in core life. The minimum total negative reactivity worth of CEAs available for the present cycle during a reactor trip from full power and zero power for both beginning and EOC is given in UFSAR Chapter 3. The net CEA worth available at trip is the minimum total worth reduced by the worth of the most reactive CEA. To increase operating flexibility, each initiating event uses a shutdown (SCRAM) reactivity that is consistent with conditions at the time of reactor trip. The Power Dependent Insertion Limit (PDIL) discussed in the Technical Specifications assures that the necessary CEA worth is available upon reactor trip.

Transient shutdown reactivity worth versus CEA position curves are dependent on Axial Shape Index (ASI) and Regulating CEA Group positions. The most conservative one for the particular set of initial conditions is used for each event. The shutdown worth versus CEA position curves predict a lower rate of reactivity insertion than is expected under any allowed initial condition. The shutdown worth versus CEA position curve is combined with a CEA position versus time curve to yield a shutdown worth versus time curve for the analyses. Consequently, a conservative representation of shutdown reactivity insertion rate is used for reactor trips that occur as a result of the events considered in this chapter.

d. Effective Neutron Lifetime Delayed Neutron Fraction

The effective neutron lifetime and delayed neutron fraction are functions of fuel burnup.

For the analysis of each event, either BOC or EOC cycle specific values of the neutron lifetime and of the delayed neutron fraction are selected.

e. Decay Heat Generation Rate

Analyses based on full power initial conditions conservatively assume a decay heat generation rate based on the 1971-1973 Proposed American Nuclear Society (ANS) Decay Heat Standard, which includes contributions from heavy element decay and is based on an infinite reactor operation period at full power.

f. Core Inlet Temperature (T_{cold})

A core inlet temperature range of 525°F to 535°F was originally used in the safety analysis for Hot Zero Power (HZP) events. These events have been evaluated to the minimum temperature of criticality allowed by the Technical Specifications, 515°F, and determined that sufficient analytical conservatisms exist to preserve the conclusions of the existing analysis. For at power events, core inlet temperature is assumed consistent with program temperature as shown in UFSAR Figure 4-9. A temperature indication uncertainty is normally applied in the safety analysis.

14.1.3 ASSUMED PROTECTION SYSTEM ACTIONS

During the course of any event, various systems are called upon to function. Such systems are described in UFSAR Chapters 6, 7, and 9. The manner in which these systems function during events is discussed in the Sequence of Events and Systems Operation subsections of each event description. Section 14.1.3.1 describes the Sequence of Events and Systems Operation. Section 14.1.3.2 describes the plant protection system analysis setpoints and delay times assumed in the analyses presented in this chapter. Section 14.1.3.3 reviews the status of control systems assumed in the transient analysis.

14.1.3.1 Sequence of Events and Systems Operation

The purpose of the Sequence of Events and Systems Operations subsections is to identify:

a. The sequence of events from event initiation to the final stabilized condition;
b. The extent to which normally operating plant instrumentation and controls are assumed to function;
c. The extent to which plant protection systems are required to function;
d. The credit taken for the functioning of normally operating plant systems; and
e. The operation of engineered safety systems.

14.1.3.2 Protection System Setpoints

The Calvert Cliffs Units have two protection systems: The RPS and the Engineered Safety Feature Actuation System (ESFAS).

The RPS is described in UFSAR Chapter 7. Analytical setpoints include instrument uncertainty and are conservative relative to the allowable limits listed in Table 7-1. Credited trip delay times are equal to or conservatively larger than values listed in Table 7-2. Delay times are defined as the elapsed time between the parameter reaching its analysis setpoint and the opening of the reactor trip breakers.

The time interval between opening of the trip breaker and the point at which the magnetic flux of the CEA holding coil has decayed enough to allow CEA motion is conservatively assumed to be 0.50 seconds. Finally, a conservative value of 2.6 seconds is assumed as the elapsed time from the beginning of CEA motion to the time the CEAs are 90% inserted into the reactor core. Thus, a time interval of 3.1 seconds is assumed between the interruption of power to the CEA holding coil and the point of 90% insertion.

The ESFAS is described in UFSAR Chapter 7. The analysis setpoints and systems actuated by the ESFAS during each event are described in the Sequence of Events subsection for each event. Credited response times are equal to or conservative to the values listed in Table 7-4.

Critical assumptions made in the Safety Analysis about valve and pump responses are included in Table 14.1-2.

14.1.3.3 Control System Operational Status

Some normally operating control systems are assumed to function during the course of the events described in this chapter. The operability status of a control system is dependent on the criteria being addressed for a particular initiating event. In general, automatic operation of these normally available control systems is assumed unless operation in the manual mode would make the approach to acceptance criteria more limiting. In the manual mode, the control system is assumed to continue to function as it was prior to the event. For example, for a peak pressurizer pressure event, the Steam Bypass Control System is assumed to be in manual and will remain closed throughout the event, although it would normally be called upon to open at reactor trip. The assumed status of each control system is presented in the discussion of each event.

During normal operating conditions, the plant is operated with the Feedwater Regulating System in automatic and with the SG water level at the Normal Water Level. This mode is therefore assumed for events analyzed in this chapter.

The Pressurizer Level Control System (PLCS) regulates the water level in the pressurizer within ± 5% of the programmed level during normal plant operating conditions. For events analyzed in this chapter, pressurizer level is assumed to be between 116 and 242", except when three charging pumps are operating and letdown flow is less than 29 gpm. Under those conditions pressurizer level is assumed to be between 116 and 227".

Although the plant regulates the pressurizer pressure at 2250 psia for normal plant operating conditions, events analyzed in this chapter assume a value between 2200 and 2300 psia. A pressure indication uncertainty is applied as is appropriate. As of Unit 1 Cycle 19 and Unit 2 Cycle 17, analyses are assessed for initial pressurizer pressure values up to and including 2311 psia.

The boration system is a subset of the Chemical Volume and Control System (CVCS) and ensures that negative reactivity control is available during each mode of facility operation. Typically, the CVCS is assumed to be initially operating with one charging pump, with a total of six gpm to the RCPs and a resultant letdown flow rate, with minimum pressurizer spray, and with minimum heat input from the proportional heaters.

For Units 1 and 2, loss-of-coolant accident (LOCA) flow from the charging pumps is not required and is not credited.

No boration is assumed to occur via the CVCS upon a safety injection actuation signal (SIAS) since the system may be lined-up to dilute directly to the charging pump suction. In this line-up, the boric acid pump head may not overcome the reactor coolant makeup pump head when SIAS actuates. As a result, concentrated boric acid may not be injected upon a SIAS.

For all events, the CEA motion inhibit is assumed to be operable. Control element assembly motion inhibit insures that programmed CEA group overlap will be maintained and that a single CEA withdrawal will not occur.

14.1.4 CORE AND SYSTEM PERFORMANCE

14.1.4.1 Mathematical Models

This section briefly describes the computer codes used in analyzing the DBEs.

A. Non-Loss-of-Coolant Accident Events

1. CESEC

The CESEC digital computer code (Reference 1) simulates a Combustion Engineering, Inc. (CE) NSSS. The code calculates the plant response for non-loss-of-coolant accident initiating events for a wide range of operating conditions.

The primary system components considered in the code include the reactor vessel, the reactor core, the primary coolant loops, the pressurizer, the SGs, and the RCPs. The secondary system components include the secondary side of the SGs, the steam system, the feedwater system, and the various steam control valves. In addition, the program models those plant control and protection systems needed to perform the safety analysis.

The secondary systems in CESEC are modeled as a single node. In events where the peak pressure of the secondary system is considered, the elevation head of the liquid in the steam generator downcomer is added to the CESEC peak steam generator pressure.

2. S-RELAP5

The S-RELAP5 plant transient thermal-hydraulic system code is used to simulate the overall transient response of the RCS and steam systems during the non-LOCA event. The S-RELAP5 plant model includes a thermal model of the fuel, a hydraulic model of the RCS, a point-kinetics model of the reactor, a hydraulic model of the steam system, and control logic which represent various RPS trips. The RCS hydraulic model simulates the hot legs, pressurizer, SGs (primary sides), cold legs, RCPs, reactor vessel, and core. The steam system hydraulic model simulates the SGs (secondary sides), main steam lines, and turbine.

3. FIESTA

FIESTA is no longer used for the feed line break analysis.

4. Scram Insertion

The critical eigenvalue as a function of control rod position is calculated. The scram reactivity is constructed as the combination of a table of scram reactivity versus control rod position and control rod position versus time. The control rods are inserted into a bounding bottom-peaked shape to delay the reactivity insertion. The scram worth is reduced to account for the worst stuck control rod, as appropriate. Delays between the time the trip setpoint is reached and the start of control rod insertion are included in the transient analysis.

5. Statistical Setpoint/Transient Methodology

The analog protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying:

a) Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RPS;

b) Limiting Conditions for Operation (LCO) for reactor system parameters; and,

c) LCO for equipment performance.

The LSSS, combined with the LCO, established the thresholds for protection system action to prevent exceeding acceptable limits during DBEs where changes in DNBR and linear heat rate (LHR) are important. The limits addressed by the RPS are:

a) The reactor fuel shall not experience centerline melt; and
b) The DNBR shall have a minimum allowable limit corresponding to a 95% probability at a 95% confidence level that Departure from Nucleate Boiling (DNB) will not occur.

The RPS trips jointly provide protection for all AOOs. The RPS providing primary protection from centerline melt is the Local Power Density (LPD) LSSS. The RPS providing primary DNB protection is Thermal Margin/Low Pressure (TM/LP) LSSS.

The design of the RPS requires that correlations including uncertainties be applied to express the LSSS in terms of functions of monitored parameters. These functions are the trip limits that are then set into the RPS.

Two methods can be used to compensate for uncertainties. It can be assumed that all applicable uncertainties occur simultaneously in the most adverse direction even though not all of the uncertainties are systematic; some are random and some contain both systematic and random characteristics. This assumption is extremely conservative. Reference 45 documents the methodology used to statistically combine uncertainties explicitly in lieu of the credit previously used.

The scope of Reference 45 encompasses the following objectives:

a) To define the method used to statistically combine uncertainties to determine the TM/LP and LPD LSSSs and LCOs;
b) To describe the methods for statistically confirming the TM/LP and LPD LSSSs and LCOs.

Operation within the DNB and LPD LCOs provides the necessary initial DNB and LHR margin to prevent exceeding acceptable limits during DBEs where changes in DNBR and LHR are important.

With the introduction of AREVA/Framatome fuel assemblies, a new method for the statistical combination of uncertainties was used to verify the TM/LP LSSS, the LPD LSSS, the DNB LCO, and LPD LCO.

The LCOs and LSSSs protect against fuel failure in LOCAs, prevent DNB, and meet the specified acceptable fuel design limits (SAFDLs) for fuel centerline melt (FCM). Loss-of-coolant accident limits are based on the LHGR used in the LOCA analyses, DNB limits are based on correlations which have been approved by the NRC, and FCM limits are calculated for each reload cycle and fuel design.

Statistically combining the uncertainties involved in calculating LCOs and LSSSs establishes conservative and meaningful values for those settings. The statistical approach provides an accurate method for accounting for uncertainties and can require a large number of calculations.

A bounding approach is used to reduce the number of calculations for some cases. This approach is used for cases that have many nominal cases. The uncertainties are combined at each nominal point and a margin is defined. In general, the number of calculations used in the analysis can be reduced by statistically combining the bulk of the uncertainties at a single nominal point and applying this calculational uncertainty to every nominal point. The nominal point used is conservatively chosen to provide the greatest uncertainty in the calculated results and therefore, a conservative estimate at all other points.

The nominal point is chosen by finding the location where the difference between the nominal point and the deterministic calculation is maximum. This method is used in deriving and confirming the setpoints. Deriving an LCO or LSSS is somewhat more difficult than confirming an existing setpoint, because the power uncertainty coming from the ASI uncertainty depends on the functional form and the process becomes iterative.

LPD LSSS and LCO

Protection against FCM is provided by the LPD LSSS and the LPD LCO. The LPD LSSS protects against FCM by monitoring the power level of the reactor and tripping the reactor when the power level exceeds the trip setpoint corresponding to the ASI. The LPD LCO limits power operation based on the ASI. The function of the LPD LCO is to protect against the LPD exceeding the LHGR limit established in the LOCA analysis.

These functions are based on the worst axial (F_z) and radial (F_r) power distributions for a given ASI. Radial power distribution does not affect the ASI directly. Therefore, the radial peaking factor assumed for all values of ASI is the Technical Specification value. The F_r is augmented to account for increased peaking when CEAs are inserted.

Each axial power distribution has a value for ASI. However, an unlimited number of axial power distributions can correspond to the same ASI. In determining the LPD LSSS and LCO, the most limiting axial distribution for a given ASI sets the limit.

The LPD LSSS setting, without uncertainties, is generated by determining the power at which FCM is predicted to occur. The allowed power for each axial power shape and corresponding F_q is calculated. The resulting power and ASI points are plotted and a curve is drawn below all of the power versus ASI points. This curve represents the LPD LSSS, without accounting for uncertainties. The curve provides the power and ASI combination where any axial shape at this power level and ASI value will protect against FCM.

To confirm the LPD LSSS, a series of calculations is performed using each axial power shape. Those shapes for which the melt power exceeds the power and ASI combination by the offset of the Variable High Power Trip (VHPT), adjusted for uncertainties, are not considered. For the remaining shapes, the nominal margin between the trip power and the FCM power is calculated, then adjusted for uncertainties. This adjustment is made by calculating the probability distribution in margin between the trip power and the power at which FCM would occur. Using a one-sided, lower 95% of the margin from the distribution, a table of margin versus ASI is created.

The uncertainties that must be accounted for are from two sources: measurement and calculation uncertainties. Calculation uncertainties include model structural deficiencies and parameter uncertainties. The uncertainties included in the LPD LSSS include;

- Engineering uncertainty
- Power peaking uncertainty
- Azimuthal tilt allowance
- Power uncertainty, which may be power dependent
- Trip power uncertainty (for LSSS only)
- Transient decalibration and trip overshoot bias (for LSSS only)
- ASI uncertainty

The ASI measurement drift and power measurement are pure measurement uncertainties and can be applied directly to the results of an uncertainty analysis in which they are not varied.

The LPD LCO is similar in form to the LPD LSSS and is based on preventing the plant from exceeding a reduced LHGR during operation. The value of the reduced LHGR is no greater than the value used in the LOCA analysis. The LPD LCO comes into effect only when the in-core detectors are not in service. Since the LPD LCO has no trip associated with it, uncertainties associated with the trip are not applied to the limits.

TM/LP LSSS and LCO

Protection against DNB is provided by the TM/LP LSSS and the DNB LCO. The TM/LP LSSS trips the reactor when conditions approach a 95% probability that the limiting fuel pin undergoes DNB. The DNB LCO ensures that AOOs will not result in DNB with at least a 95% probability at a 95% confidence level.

The TM/LP trip is actuated when the measured pressure falls below a calculated limit. The calculated limit is based on analysis of the DNBR as a function of pressure, ASI, power, and inlet temperature. The TM/LP has a minimum pressure that will result in a trip. The TM/LP trip pressure is the maximum of the calculated trip pressure and the floor.

The power term is adjusted, depending upon the ASI. Peaking factors are adjusted for part-power rodded configurations. The power input to the TM/LP is the maximum of that calculated by two different methods: the excore (neutron) detectors and a thermal calculation.

An iterative scheme is used to obtain the DNB portion of the variable pressure setting for a selected value of ASI, power, and T_{inlet}. The axial power shape that produces the worst DNBR for a range of ASI values centered around a selected ASI is used to calculate the pressure corresponding to DNB. Values of power which are not permitted by the LPD LSSS or the MSSV settings are not considered in the analysis. The radial peaking factor is set at its nominal value. Flow is fixed at the thermal design limit and the DNBR target is set to the DNBR mean, adjusted for rod bow and mixed core penalties. Power and T_{inlet} are chosen and the pressure corresponding to DNB is calculated. The iteration provides a single point for DNB pressure.

The process is repeated for a set of ASI values ranging from -0.6 to 0.6 and over a large range of powers and T_{inlet}. The results of the nominal calculation set provides a series of pressures as a function of power, ASI, and T_{inlet} at which the CHF calculated by the DNBR correlation and corrected for rod bow and mixed core penalties, is equal to the calculated heat flux.

The TM/LP LSSS protects against hot-leg saturation and DNB during slow transients. The uncertainties that are accounted for include;

- Radial power peaking factor
- DNB correlation uncertainty
- Loop flow
- Pressure measurement
- Inlet temperature measurement uncertainty (RTDs)
- Power measurement uncertainty
- ASI uncertainty

The DNB LCO is designed to protect the DNBR SAFDL during an AOO. It is set and confirmed by the limiting transient events. The limiting transients are those that produce the largest decrease in DNBR from an initial steady-state power and ASI. Operation of the reactor within the limits of power versus ASI, disregarding uncertainties, means no AOO would result in DNBR less than or equal to the adjusted mean of the applicable DNBR correlation.

Uncertainties in the parameters and model structure are accounted for by using the probability distribution in DNB power at the most sensitive point. This point corresponds to the axial power shape that produces the greatest difference between the nominal and the deterministic percent allowed power. Protection against DNB is provided by limiting the reactor power based on the peripheral (external) ASI. The shape of the DNB LCO, in conjunction with radial power peaking limits and other LCOs, protects against DNB.

The uncertainties applicable to the DNB LCO that must be accounted for are listed below. Applicability of the uncertainty is dependent upon the event analyzed.

- Radial power peaking factor
- Pressure measurement
- DNB correlation uncertainty
- Loop flow
- Inlet temperature measurement uncertainty (RTDs)
- ASI uncertainty

- Power measure uncertainty
- Pump coastdown (for loss-of-coolant flow)
- Low flow trip uncertainty (for loss-of-coolant flow)
- Scram delay (for loss-of-coolant flow)
- Scram worth (for loss-of-coolant flow)
- Scram rate (for loss-of-coolant flow)

Either the excore or the incore detectors can be used to monitor the LHR LCO. The DNB LCO and ASI is monitored on the excore detectors.

Below 20% power, ASI limits for the LHR and DNB LCO are not required. At low powers, the axial power distribution trip will limit the allowed ASI during operation.

6. Control Element Assembly Withdrawal

Analysis is performed at HFP and HZP conditions. Appendix C of the Technical Specifications prohibits changing Core Operating Limits Report Figures 3.1.6, 3.2.3, and 3.2.5 until an NRC accepted generic, or Calvert Cliffs specific, basis is developed for analyzing the CEA Rod Bank Withdrawal event, the CEA Drop event, and the CEA Ejection event (power level sensitive transients) at full power conditions only.

B. Containment Response

RELAP5/MOD2-B&W is used to generate the mass and energy release rates used in the containment analysis for the main steam line break (MSLB) inside Containment. RELAP5/MOD2-B&W is an advanced system analysis computer code designed to analyze a variety of thermal-hydraulic transients in light water reactor systems. RELAP5/MOD2-B&W is advanced over its predecessors due to its six-equation, full non-equilibrium, two-fluid model for the vapor-liquid flow field and partially implicit numerical integration scheme for more rapid execution. As a system code, it provides simulation capabilities for the reactor primary coolant system, secondary system, feedwater trains, control systems, and core neutronics. Special component models include pumps, valves, heat structures, electric heaters, turbines, separators, and accumulators. Code applications include the full range of safety evaluation transients, LOCAs, and operating events. RELAP5/MOD2-B&W is approved by the NRC for performing analysis on plants with recirculating steam generators.

The mass and energy released into Containment during a LOCA has been calculated by Westinghouse.

C. Loss-of-Coolant Accident

1. S-RELAP5

The S-RELAP5 code (Reference 38) is a RELAP5 based thermal-hydraulic system code for performing small break LOCA (Reference 39) and realistic analyses of a large break LOCA (Reference 40) in PWRs. RELAP5 is a light water reactor transient analysis code developed at the Idaho National Engineering Laboratory for the NRC. The main purpose of the RELAP-5 code is to calculate the behavior of a RCS during a transient by simulating a wide variety of hydraulic and thermal transients in both nuclear and non-nuclear systems involving mixtures of steam, water, non-condensable gas, and solute. The code includes models of hydrodynamic systems, heat transfer and heat conduction, fuel, reactor kinetics, control system, and trip system models.

Improvements to the code have been incorporated as required to provide congruency with the unmodified literature correlations and those required to obtain adequate simulation of key LOCA experiments.

2. RODEX3A

RODEX3A is a best-estimate fuel code which has been approved for use in the performance of realistic large break LOCA analyses. The RODEX3A code simulates the thermal and mechanical response of a fuel rod in a coolant channel as a function of exposure for the normal and power ramp conditions encountered in a PWR. Phenomenological rate-dependent models are used to evaluate the temperature-, stress-, and exposure-dependent changes in the state of the fuel and cladding materials and in the release of the inert gaseous fission products. A quasi-steady-state computational procedure is used to evaluate the response of a fuel rod as a function of time.

3. ICECON

The ICECON code has been incorporated into the S-RELAP5 code to provide the required containment boundary conditions for the large break LOCA analysis. ICECON was developed to predict the long-term behavior of PWR nuclear plant containment systems.

4. RODEX2

The RODEX2 code (References 43 and 44) is a fuel rod code which has been approved for use in the performance of small break LOCA analyses (Reference 39). The RODEX2 code simulates the thermal and mechanical response of a fuel rod in a coolant channel as a function of exposure for the normal and power ramp conditions encountered in pressurized (PWR) and boiling water reactors. The code incorporates models to describe the thermal-hydraulic condition of the fuel rod in a flow channel; the gas release, swelling, densification, and cracking in the pellet; the gap conductance; the radial thermal conduction; the free volume and gas pressure internal to the fuel rod; the fuel and cladding deformations; and the cladding corrosion. The calculations are performed on a time incremental basis with conditions being updated at each calculated increment.

D. Thermal Hydraulics

1. XCOBRA-IIIC

Based on the overall core conditions calculated by S-RELAP5 at selected times during the transient, the XCOBRA-IIIC fuel assembly thermal-hydraulic code (Reference 41) is used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly at those times. The XCOBRA-IIIC model consists of a thermal-hydraulic model of the core (representing each assembly by a single “channel”) linked to a detailed thermal-hydraulic model of the limiting assembly (representing each subchannel by a single “channel”). The limiting assembly DNBR calculations are performed using an approved DNB correlation.

2. COAST

In the original FSAR, the COAST (Reference 6) computer code was used to calculate the reactor coolant flow coastdown transient characteristic for any combination of active and inactive pumps and forward or reverse flow in the hot or cold legs.

The equations of conservation of momentum are written for each of the flow paths of the COAST model assuming unsteady one-dimensional flow of an incompressible fluid. The equation for conservation of mass is written for the appropriate nodal points. Pressure losses due to friction, bends, and shock losses are assumed proportional to the flow velocity squared. Pump dynamics are modeled using a head-flow curve for a pump at full speed and using four-quadrant curves, which are parametric diagrams of pump head and torque on coordinates of speed versus flow, for a pump at other than full speed.

The current coolant flow coastdown transient is based on actual plant data taken in 1981, adjusted for the effect of steam generator tube plugging as predicted by COAST.

E. Fuel Performance

The RODEX2-2A and RODEX2-3A fuel codes are used to calculate fuel performance for input to non-LOCA and LOCA transients. RODEX2-2A (References 43 and 44), was developed to perform calculations for a fuel rod under normal operating conditions. The code incorporates models to describe the thermal-hydraulic condition of the fuel rod in a flow channel; the gas release, swelling, densification and cracking in the pellet; the gap conductance; the radial thermal conduction; the free volume and gas pressure internal to the fuel rod; the fuel and cladding deformations; and the cladding corrosion.

For realistic large break LOCA, fuel conditions are computed using RODEX3A prior to starting the S-RELAP5 portion of the analysis. RODEX3A computes the initial fuel stored energy, fission gas release, and fuel-cladding gap conductance at the reference fuel temperature and zero power. RODEX3A data is transferred to S-RELAP5 and a steady-state S-RELAP5 calculation is required to initialize the S-RELAP5 calculation at the power of interest.

F. Reactor Physics Computer Codes

Numerous computer codes are used to produce the reactor physics parameter input values required by the NSSS simulation and other codes previously described. These reactor physics codes are described in Chapter 3.

14.1.4.2 Operator Action Requirement

Operator actions assumed to mitigate the consequences of the events presented in this chapter are delineated in each event description requiring such action. The operator should be alerted to the need for action by an unambiguous alarm. An unambiguous alarm is one which, within the time period allowed for diagnosis, would make the operator aware of the need to take the action assumed. There may or may not be redundant or diverse plant alarms for a particular action; however, there are always continuously operating, non-alarming visual indications of relevant plant process parameters in the Control Room which serve as back-ups.

A time delay is assumed between the unambiguous alarm and the accomplishment of any manually-initiated action. This delay conservatively accounts for the time required by the operator to diagnose the event, decide what action to perform, and then initiate this action.

After initiation of the Boron Dilution event, operator action is initiated within 15 minutes when in Modes 1 through 5 and within 30 minutes when in Mode 6.

14.1.4.3 Activity Release Methodology

This subsection summarizes the assumptions, parameters, and calculational methods used to determine the doses that result from postulated events. The total activity released to the site boundary is dependent upon the initial activity in the primary and secondary systems as well as any changes in activity resulting from the event. Depending on the initiating event, radioactivity could be released to the site boundary through the atmosphere dump valves, main steam safety valves (MSSVs), steam turbine-driven auxiliary feed pumps, condenser air removal system, and/or leak through the Containment and engineered safety feature (ESF) system.

a. Primary System Activity

The primary system activity is based upon the initial activity in the primary coolant and the activity released to the coolant due to failed fuel rods during the event. The initial primary system concentration is conservatively assumed to be at the Technical Specification limit of 0.5 μCi/gm Dose Equivalent curies (DEQ) I-131 and 100/E μCi/gm gross activity.

b. Fuel Cladding Failure

The release of gas gap activity from pins calculated to experience clad failure is considered. For each pin calculated to experience clad failure, a conservative fraction of the total pin activity of iodines and noble gases is assumed to be present in the gas gap. This amount is assumed to be released instantaneously to the RCS. This release increases the primary system specific activity.

c. Secondary Activity

A conservative set of secondary side specific activities is used to calculate releases from the SGs. The DEQ I-131 activity limit is the Technical Specification limit of 0.1 μCi/gm.

d. Steam Generator Tube Leakage

All events considered, with the exception of the Steam Generator Tube Rupture (SGTR) event, were analyzed assuming the Technical Specification limit of 100 gal/day through each SG. For the SGTR event, 200 gal/day is assumed to leak through the unaffected SG.

e. Containment Leakage

The assumed leak rate from containment is the maximum Technical Specification limit of 0.16% by weight of containment air per day.

f. Calculational Factors

A conservative set of decontamination factors (i.e., reciprocal of partition factor) for iodine releases from the primary and secondary side was used to calculate activity releases to the atmosphere. The Decontamination Factor for dose calculations is functionally defined as the ratio of the concentration of iodine in water to that in steam. These values are presented in Table 14.1-3.

A breathing rate of $3.5x10^{-4}$ m^3/sec is assumed in the analysis which is characteristic of the active portion of a normal adult workday.

An atmospheric dispersion coefficient Chi/Q of $\geq 1.3x15^4$ sec/m^3 is assumed in the analysis. The relative concentration is conservative for the 0-2 hour site boundary condition described in Chapter 2.

g. Calculational Methods

The isotopic activities released from the failed fuel per unit power were generated by the isotope generation and depletion computer code SAS2H/ORIGEN-S (References 32 and 33) and multiplied by the relevant power level and release fractions. The SAS2H control module performs the depletion/decay analysis using the well-established codes and data libraries provided in the SCALE system. Problem-dependent resonance processing of neutron cross-sections is performed using the Bondarenko resonance self-shielding module BONAMI-S and the Nordheim Integral Treatment resonance self-shielding module NITAWL-II. The XSDRNPM-S module is used to produce spectral weighted and collapsed cross-sections for the fuel depletion calculations. COUPLE updates the cross-section constants included on an ORIGEN-S nuclear data library with data from the cell-weighted cross-section library produced by XSDRNPM-S. The weighting spectrum computed by XSDRNPM is applied to update all nuclides in the ORIGEN-S library that were not specified in the XSDRNPM analysis. The point-depletion ORIGEN-S module is used to compute time-dependent concentrations and source terms for isotopes simultaneously generated and depleted through neutronic transmutation, fission, and radioactive decay. The cross-section library 44GROUPNDF5 was utilized in this analysis. 44GROUPNDF5 is a 44-energy group library derived from the latest ENDF/B-V files with the exception of O-16, Eu-154, and Eu-155, which were taken from the improved ENDF/B-VI files. Note that the SAS2H/ORIGEN-S libraries include 689 light elements, such as clad and structural materials, 129 actinides including fuel nuclides and their decay and activation products, and 879 fission product nuclides.

The RADTRAD (References 34 through 36) computer code can calculate TEDE and thyroid doses to personnel at the site boundary, low population zone, and Control Room per the AST methodology of 10 CFR 50.67 and Reference 31 resulting from any postulated accident which releases radioactivity within the Containment, SFP, or within the primary system. RADTRAD models the transport of radioactivity from up to 63 radioisotopes from the sprayed and unsprayed regions of a primary containment or a SFP area, through the secondary containment if any, and then to the environment and to the Control Room. The code includes the capability to model time-dependent activity release; containment spray, filtration, and leakage; control room filtration and inleakage; primary and secondary containment purge filters; Control Room intake filters; atmospheric dispersion; and natural decay. The activity released to the environment is transported to the site boundary and to the Control Room via appropriate atmospheric dispersion coefficients.

MICROSHIELD (Reference 37) is a comprehensive photon/gamma ray shielding and dose assessment point kernel program that is used for calculating shine doses from the Containment volume, Containment and Control Room filters, and plume activities. The MICROSHIELD point kinetics code is capable of modeling various geometries including distance and orientation between source and dose point; dimension of the source region; and the dimensions, locations, and orientations of intervening shields. MICROSHIELD has solution algorithms for 16 source geometries including points, lines, disks, spheres, cylinders, rectangular volumes, truncated cones, infinite planes, and infinite slabs with various shields. The source and shield material compositions, densities, and buildup factors can be specified. A source strength as a function of isotopic content or photon energy spectrum can also be specified.

14.1.4.4 Fuel Performance Models and Acceptance Criteria

The acceptability of fuel rod performance is assessed by comparing the predicted thermal and mechanical behavior to appropriate guidelines. The computer codes used to determine the acceptability of fuel rod performance are described in Section 14.1.4.1. S-RELAP5 predicts the transient core average heat flux, the core inlet average coolant condition, and the RCS pressure, the values of which are all input to XCOBRA-IIIC. XCOBRA-IIIC predicts the DNBR in the reactor core based on a closed-channel, thermal-hydraulic model. RODEX predicts the hot rod fuel and clad temperatures and the pressure differential across the clad. The analysis of all events has been taken to the point of showing decreasing fuel temperatures, increasing DNBR, and a sufficient coolant inventory to meet the criteria.

The following sections describe the models and fuel behavior acceptance criteria used in analyzing each category of DBEs.

A. Anticipated Operational Occurrences

1. Departure from Nucleate Boiling Ratio

The steady-state and transient SAFDL on DNBR is that the minimum DNBR shall provide at least a 95% probability with a 95% confidence of not experiencing DNB on the fuel rod with that DNBR. Compliance with this limit ensures that there is a low probability of fuel rods being damaged due to cladding overheating.

Values of the following system parameters are used to determine the minimum DNBR during an event. Uncertainties are included.

1. Thermal power,
2. Integrated radial peaking factor,
3. Core inlet temperature,
4. Pressurizer pressure,
5. Parametric analyses in axial shape, and
6. RCS core flow.

The XCOBRA-IIIC computer code is used for the evaluations of minimum DNBR. The XCOBRA-IIIC code is a steady-state thermal-hydraulics code that calculates the axial and radial flow and enthalpy distribution within assemblies and sub-channels for non-LOCA events. When used in conjunction with core boundary conditions from the S-RELAP5 transient analysis and high thermal performance DNB correlation; XCOBRA-IIIC also calculates the corresponding Minimum Departure from Nucleate Boiling Ratio (MDNBR). Minimum DNBR calculations are performed in a two-step process. Calculations are first performed on a core-wide basis to calculate the axially varying flow and enthalpy distribution in the peak powered fuel assembly. Next, these flow and enthalpy boundary conditions are applied to a sub-channel model of the peak powered fuel assembly. Then, these flow and enthalpy boundary conditions are applied to a sub-channel model of the peak powered assembly to determine the local conditions for the calculation of MDNBR.

2. <u>Fuel Temperature</u>

The SAFDL on fuel temperature is that no significant fuel melting will occur during steady-state operation or during a transient. The fuel melting point is assumed to be that described in the Technical Specifications. Compliance ensures that the fuel rod is not damaged as a result of material property changes and increases in fuel pellet volume which could accompany fuel melting.

3. <u>Site Boundary Dose</u>

During AOOs, the site boundary dose shall not exceed a small fraction of 10 CFR 50.67 guidelines. The site boundary dose is controlled by meeting the DNBR and fuel temperature SAFDLs described in Items 1 and 2 above, which preclude any significant cladding damage.

4. <u>RCS Pressure Upset Limit</u>

The RCS Pressure Upset Limit is 110% of design pressure and is equal to 2750 psia.

B. <u>Postulated Accidents</u>

1. <u>Site Boundary Dose</u>

For Postulated Accidents, the site boundary dose shall not exceed the 10 CFR 50.67 guidelines. This is met by minimizing the number of fuel rods predicted to fail.

For the CEA Ejection event, radioactive gases are assumed to be released from the pellet to the coolant if the deposited energy is predicted to exceed the threshold value. For this event, Reference 46 is used to calculate the deposited energy (Section 14.13).

The pre-trip SLB event assumes that all fuel rods with a DNBR less than the NRC-approved safety limit experience DNB. The safety limit DNBR gives a 95% probability, at a 95% confidence level, that the hottest fuel rod will not experience DNB. For dose calculations, all rods experiencing DNB are assumed to fail. For the return to power SLB methodology, DNB is assumed when the rod experiences a DNBR less than the limit associated with the modified Barnett CHF correlation (Reference 42).

The modified Barnett correlation CHF limit associated with Framatome HTP™ fuel is expressed as a function of pressure when applied to transients that result in core exit pressures that are ≤ 1047 psia. The function used to determine the limit is expressed as (2.33 + 0.000615 x core exit pressure). The modified Barnett correlation CHF limit is applied as a single value of 4.25 when the transient results in core exit pressures between 1047 and 1540 psia.

2. RCS Pressure Upset Limit

   The RCS Pressure Upset Limit is 110% of design pressure and is equal to 2750 psia.

3. Containment Pressure

   The containment design pressure limit is 50 psig. It is not explicitly analyzed for events that bypass Containment (SGTR) or have little or no mass release, such as Seized Rotor.

4. Emergency Core Cooling System Criteria

   The LOCA event shall meet the criteria in Reference 10.

5. Coolable Core Geometry

   Fuel geometry is maintained such that a path for removal of decay heat is ensured. This criteria is met by limiting the extent of fuel melting.

C. Postulated Occurrences

1. Site Boundary Dose

   For postulated occurrences, the site boundary dose shall not exceed the 10 CFR 50.67 guidelines. This is met by minimizing the number of fuel rods predicted to fail.

14.1.5 REFERENCES

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TABLE 14.1-1

DESIGN BASIS EVENTS

A. Anticipated Operational Occurrences for which action of RPS is required to prevent exceeding acceptable limits:

UFSAR NUMBER	ORIGINAL FSAR NUMBER	EVENT
14.3	14.3	Boron Dilution
14.5	14.9	Loss of Load
14.6	14.10	Loss of Feedwater (LOFW) Flow
14.8	--	RCS Depressurization

Anticipated Operational Occurrences for which the RPS trips and/or sufficient initial steady-state thermal margin, maintained by the LCOs, are necessary to prevent exceeding the acceptable limits:

UFSAR NUMBER	ORIGINAL FSAR NUMBER	EVENT
14.2	14.2	Control Element Assembly (CEA) Withdrawal
14.4	14.11	Excess Load
14.7	--	Excess Feedwater Heat Removal
14.9	14.6	Loss-of-Coolant Flow
14.10	--	LOAC
14.11	14.4	CEA Drop
14.12	--	Asymmetric SG Events
14.25	--	Excessive Charging

B. Postulated Accidents:

UFSAR NUMBER	ORIGINAL FSAR NUMBER	EVENT
14.13	14.13	CEA Ejection
14.14	14.12	SLB
14.15	14.14	SGTR
14.16	--	Seized Rotor
14.17	14.15	LOCA
14.26	--	Feedline Break

C. Postulated Occurrences:

UFSAR NUMBER	ORIGINAL FSAR NUMBER	EVENT
14.18	14.5	Fuel Handling Incident
14.19	14.8	Turbine - Generator Overspeed Incident
14.20	14.16	Containment Pressure Analysis
14.22	14.17	Waste Gas Incident
14.23	14.20	Waste Processing System Incident
14.24	14.18	Maximum Hypothetical Incident

TABLE 14.1-2

SAFETY ANALYSIS VALVE AND PUMP ASSUMPTIONS

COMPONENT		INSTALLED QUANTITY	ANALYSIS TOTAL CAPACITY	ANALYSIS SETPOINT
Turbine Bypass Valves		4	40%[(a)]	Start open = 895 psia (820)[(f)] Full open = 905 psia (830)[(f)]
Atmospheric Dump Valves		2	5%[(a)]	Start open = 8°F (T_{avg}-532°F) Full open = 30°F (T_{avg}-532°F) Close = 3°F (T_{avg}-532°F)
Turbine Control Valves		4	---	Closure time ≤ 2 sec[(b)]
Turbine Stop Valves		4	---	Closure time 0.15 sec
Main Steam Isolation Valves (MSIVs)		2	---	Closure time 6 sec
Pressurizer Safety Valves (PSVs)		2	---	Full Open = 2575 psia and 2600 psia[(h)]
Main Steam Safety Valves (MSSVs)			---	Full Open (psia)
		4		1029.25
		4		1049.7
		8		1064.7
High Pressure Safety Injection (HPSI) Pumps		3	Variable	1195 psia[(e)]
CVCS Charging Pumps		3	Variable	---
AFW Pumps				
Unit 1	Steam-driven	2	Variable	Total Developed Head = 2490'
	Motor-driven	1	Variable	Total Developed Head = 2490'
Unit 2	Steam-driven	2	Variable	Total Developed Head = 2490'
	Motor-driven	1	Variable	Total Developed Head = 2490'

(a) % of HFP steam flow.
(b) Each valve is assumed to have a signal delay time of 0.90 seconds in the safety analysis, not included in the closure time.
(c) No longer used.
(d) Deleted.
(e) The RCS pressure at which the initiation of HPSI flow delivery to the RCS is credited, when only one HPSI pump is operating.
(f) A calculation has been performed to evaluate TBV setpoint at 820-830 psia. The setpoint of 895-905 psia is bounding as long as the SG level trip setpoint is nominally at -50.0”.
(g) Deleted.
(h) The pressurizer safety valves and the MSSVs are assumed to be fully open at the maximum value allowed. See each event for specific opening setpoint.

TABLE 14.1-3

DECONTAMINATION FACTORS USED IN OFFSITE DOSE CALCULATIONS

	DECONTAMINATION FACTORS[a]	
RELEASE PATH	IODINES	OTHER ISOTOPES
Primary leak outside-containment[b]	1	1
Releases through dump valves or safety valves	100[c]	1

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[a] Decontamination factor equal to 10 means 1/10 of the total initial activity in the mass is released to the air.

[b] Certain events show appreciable releases from containment (e.g., CEA ejection). Treatment of these releases is discussed in the appropriate individual sections.

[c] In the event of SG dryout due to blowdown to the environment, the entire SG radionuclide inventory is assumed to be released to the environment.