C.I.15. Transient and Accident Analyses

The evaluation of the safety of a nuclear power plant includes analyses of the plant's responses to postulated disturbances in process variables and postulated equipment failures or malfunctions. Such safety analyses provide a significant contribution to the selection of limiting conditions for operation, limiting safety system settings, and design specifications for components and systems from the standpoint of public health and safety. These analyses are a focal point of the Commission's design certification (DC) and combined license (COL) reviews.

To support its DC or COL application, the applicant should discuss the applicable transient and accident analyses and justify its conformance to the regulations (as specified in Appendix A at the end of this section of DG-1145). Specific acceptance criteria for each transient are discussed in Section 15 of the Standard Review Plan (SRP), as amended. In particular, Title 10 of the *Code of Federal Regulations* (10 CFR) includes the following relevant requirements:

- 10 CFR 50.34(a)(1)(ii), (f)(1)(ii), and (f)(2)(xii)
- 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors"
- 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"
    - < General Design Criterion (GDC) 10, "Reactor Design"
    - < GDC 13, "Instrumentation and Control"
    - < GDC 15, "Reactor Coolant system Design"
    - < GDC 17, "Electric Power Systems"
    - < GDC 19, "Control Room"
    - < GDC 20, "Protection System Functions"
    - < GDC 25, "Protection System Requirements for Reactivity Control Malfunctions"
    - < GDC 26, "Reactivity Control System Redundancy and Capability"
    - < GDC 27, "Combined Reactivity Control Systems Capability"
    - < GDC 28, "Reactivity Limits"
    - < GDC 29, "Protection Against Anticipated Operational Occurrences"
    - < GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary"
    - < GDC 35, "Emergency Core Cooling"
    - < GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment"
    - < GDC 60, "Control of Releases of Radioactive Materials to the Environment"
    - < GDC 61, "Fuel Storage and Handling and Radioactivity Control"
- 10 CFR Part 50, Appendix E, Paragraph IV.E.8, "Emergency Planning and Preparedness for Production and Utilization Facilities"
- 10 CFR Part 50, Appendix K, "Emergency Core Cooling Systems Evaluation Models"
- 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions"
- 10 CFR Part 100, "Reactor Site Criteria"
- 10 CFR 100.21, "Non-Seismic Siting Criteria"

Discuss how the design and analysis of events comply with the requirements of the applicable Three Mile Island (TMI) Action Plan Items in NUREG-0737, "Clarification of TMI Action Plan Requirements" and NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses." Applicable TMI Action Plan Items include I.C.1, II.B.3, II.E.1.1, II.E.1.2, II.E.5.1, II.F.1, II.F.2, II.F.3, II.K.2.16, II.K.2.17, II.K.3.1, II.K.3.5, II.K.3.7, II.K.3.13, II.K.3.30, II.K.3.31, II.K.3.44, and II.K.3.45.

Discuss how the design and analysis of applicable events incorporate the resolution of unresolved safety issues (USIs) and medium- and high-priority generic safety issues (GSIs) identified in the version of NUREG-0933 current 6 months before application date, and how those USIs and GSIs are technically relevant to the applicable system design and transient and accident analyses. Applicable USIs and GSIs include USI-A-9, USI-A-47, USI-B-17, USI-C-4, USI-C-5, USI-C-6, USI-C-10, GSI-3, GSI-22, GSI-23, GSI-24, GSI-40, GSI-75, GSI-125.II.7, GSI-135, GSI-185, and GSI-191.

In addition, demonstrate that the applicable system design and transient and accident analyses incorporate the operating experience insights from generic letters (GLs) and bulletins (BLs) issued up to 6 months before the docket date of the application. Applicable GLs and BLs include GL-80-019, GL-80-035, GL-83-11, GL-83-22, GL-83-32, GL-85-06, GL-85-16, GL-86-13, GL-86-16, GL-88-16, GL-88-17, GL-93-04, GL-97-01, GL-98-02, BL-80-04, BL-80-12, BL-80-18, BL-86-03, BL-93-02, BL-95-02, BL-96-01, BL-96-03, and BL-2001-01.

It should be noted that the items listed in the above lists of GDCs, TMI Action Items, USIs, GSIs, GLs, and BLs may not constitute the total sets of relevant requirements. It is the COL applicant's responsibility to identify all relevant items applicable to their reactor designs.

C.I.15.1 *Transient and Accident Classification*

Organize the transients and accidents, and present the results that will (1) ensure that a sufficiently broad spectrum of initiating events has been considered; (2) categorize the initiating events by type and expected frequency of occurrence so that only the limiting cases in each group need to be quantitatively analyzed; (3) permit consistent application of specific acceptance criteria for each postulated initiating event; and (4) identify which transients or accidents are fuel design-dependent and are to be analyzed in every fuel cycle.

To accomplish these goals, a number of process variable disturbances and equipment failures or malfunctions are postulated. Assign each of the postulated initiating events to one of the following categories (additional initiating event categories may be defined based on unique designs of new reactors):

(1) increase in heat removal by the secondary system
(2) decrease in heat removal by the secondary system
(3) decrease in reactor coolant system flow rate
(4) reactivity and power distribution anomalies
(5) increase in reactor coolant inventory
(6) decrease in reactor coolant inventory
(7) radioactive release from a subsystem or component
(8) anticipated transients without scram (ATWS)

Typical initiating events are presented in Appendix A at the end of this section of DG-1145. For new reactor designs, evaluate the need for additional initiating events that are not included in Appendix A. Evaluate each initiating event using the outline in Section C.I.15.6. Appendices B through J at the

end of this section of DG-1145 provide guidance that may be useful in presenting the information for the transient and accident analyses.

C.I.15.2 *Frequency of Occurrence*

Discuss the expected frequency of occurrence for each initiating event according to one of the following frequency groups:

(1) Anticipated operational occurrences (AOO), as defined in Appendix A to 10 CFR Part 50, are those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit.

(2) Accidents are occurrences that are postulated but not expected to occur.

The initiating events for each combination of category and frequency group should be evaluated to identify the events that would be limiting. The intent is to reduce the number of initiating events that need to be quantitatively analyzed. That is, not every postulated initiating event needs to be completely analyzed by the applicant. In some cases a qualitative comparison of similar initiating events may be sufficient to identify the specific initiating event that leads to the most limiting consequences. Only that limiting initiating event should then be analyzed in detail.

Different initiating events in the same category/frequency group combination may be limiting when the multiplicity of consequences are considered. For example, within a given category/frequency group combination, one initiating event might result in the highest reactor coolant pressure boundary (RCPB) pressure, while another initiating event might lead to minimum core thermal-hydraulic margins or maximum offsite doses.

C.I.15.3 *Plant Characteristics Considered in the Safety Evaluation*

The applicant should summarize the plant parameters considered in the safety evaluation (e.g., core power, core inlet temperature, reactor system pressure, core flow, axial and radial power distribution, fuel and moderator temperature coefficient, void coefficient, reactor kinetics parameters, available shutdown rod worth, and control rod insertion characteristics). Specify the range of values for plant parameters that vary with fuel exposure or core reload. Ensure that the range is sufficiently broad to cover expected changes predicted for the fuel cycles to the extent practicable based on the fuel design and acceptable analytical methodology at the time of the DC or COL application. Specify the permitted operating band (permitted fluctuations in a given parameter and associated uncertainties) on reactor system parameters. Use the most adverse conditions within the operating band as initial conditions for transient analysis.

C.I.15.4 *Assumed Protection System Actions*

The applicant should list the settings of all protection system functions that are used in the safety evaluation. Typical protection system functions include reactor trips, isolation valve closures, and emergency core cooling system (ECCS) initiation. List the expected limiting delay time for each protection system function, and describe the acceptable methodology for determining uncertainties (from combined effect of calibration error, drift, instrumentation error, etc.) to be included in the establishment of the trip setpoints and allowable values specified in the plant technical specifications.

C.I.15.5 *Evaluation of Individual Initiating Events*

The applicant should provide an evaluation of each initiating event, using the format in Section C.I.15.6 of this regulatory guide. Indicate whether an initiating event is applicable to more than one category. Provide the information listed in Sections C.I15.6.1 and C.I.15.6.2 for each initiating event. The extent of the quantitative information to provide in Sections 15.6.3–15.6.5 of the final safety analysis report (FSAR) may differ for the various initiating events. For an initiating event that is not limiting, only the qualitative reasoning that led to that conclusion need be presented, along with a reference to the section that presents the evaluation of the more limiting initiating event. For those initiating events that require a quantitative analysis, an analysis may not be necessary for each section (15.6.3–15.6.5). For example, a number of plant transient initiating events result in minimal radiological consequences. In such instances, the applicant should present a qualitative evaluation to show this to be the case; however, a detailed evaluation of the radiological consequences need not be performed for each initiating event.

C.I.15.6 *Event Evaluation*

C.I.15.6.1 <u>Identification of Causes and Frequency Classification</u>

For each initiating event evaluated, the applicant should include a description of the occurrences that lead to the event under consideration. Determine and state the frequency of occurrence as either an AOO or an accident.

C.I.15.6.2 <u>Sequence of Events and Systems Operation</u>

The applicant should discuss the following considerations for each initiating event:

(1) step-by-step sequence of events from event initiation to the final stabilized condition [Identify each significant occurrence on a time scale (e.g., flux monitor trips, insertion of control rods begins, primary coolant pressure reaches safety valve set point, safety valves open, safety valves close, containment isolation signal is initiated, and containment is isolated). Identify all operator actions credited in the transient and accident analyses for consequence mitigation.]

(2) extent to which normally operating plant instrumentation and controls are assumed to function

(3) extent to which plant and reactor protection systems 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) assure consistency between the safety analyses and the emergency response guidelines/emergency procedure guidelines (ERGs/EPGs) or EOPs with respect to the operator response (including action time) and available instrumentation

Only safety-related systems or components can be used to mitigate transient or accident conditions. However, non-safety related systems or components may be assumed operable in analyses for the following cases:

(1) when a detectable and non-consequential random and independent failure must occur in order to disable the system, and

(2) when non-safety related components are used as backup protection.

For example, under case (1), continued operation of the main feedwater control system (MFCS) may be assumed in those design-basis events not related to feedwater malfunction, loss of ac, or turbine trip, if it can be shown that a failure in the MFCS is not a consequence of the initiating events, and the probability of a random, independent failure occurring in the MFCS within the time of the initiating event is extremely low. Under case (2), the turbine stop and control valves can be credited in the design-basis analyses for backup protection if the valves are demonstrated to be reliable and subject to surveillance requirements in the Technical Specifications.

For any non-safety related systems or components credited in the design-basis analyses for mitigating the event consequences, proper justification must be provided. Non-safety related systems or components that may adversely affect transient or accident analyses must be taken into account. List the non-safety related systems or components assumed in the analyses for each event in a tabular form as recommended in Appendix J of this section. The applicant should provide a discuss of how the definitions for active and passive failures, as described in SECY-77-439, "Single-Failure Criterion," dated August 1977 (ADAMS Accession No. ML 060260236) have been applied to the analyses. For passive system designs, applicants should ensure that low differential pressure check valves that perform a safety function are considered active components subject to single active failure consideration, except where their proper function can be demonstrated and documented.

Evaluate the effects of single active failures and operator errors. Provide sufficient detail to permit independent evaluation of the adequacy of the system as it relates to the event under study. One method of systematically investigating single failures is to use a plant operational analysis or failure mode and effects analysis. List all single failures or operator errors considered in the transient and accident analysis, and identify the limiting single failure for each event.

The results of these types of analyses can be used to demonstrate that the safety actions required to mitigate the consequences of an event are provided by the safety systems essential to performing each safety action.

C.I.15.6.3 Core and System Performance

C.I.15.6.3.1 *Evaluation Model*

The applicant should discuss the evaluation model used and any simplifications or approximations introduced to perform the analyses. Identify digital computer codes used in the analysis. If a set of codes is used, describe the method used to combine these codes. Present and discuss the important output of the codes under "Results." Emphasize the input data and the extent or range of variables investigated. The detailed descriptions of evaluation models and digital computer codes or listings should be included by referencing documents that are available to the NRC, if possible, and providing only summaries in the text of the application itself.

The applicant should provide a table listing the titles of topical reports (TRs) that describe models or computer codes used in transient and accident analyses, and list the associated NRC safety evaluation reports approving those TRs. Demonstrate that the use of the NRC-approved models or codes is within the applicable range and conditions of the models or codes. Provide a discussion to address compliance with each of the conditions and limitations in the NRC safety evaluation reports approving the TRs that document the models or codes used.

C.I.15.6.3.2 ***Input Parameters and Initial Conditions***

The applicant should identify the major input parameters and initial conditions used in the analyses. Appendix B (at the end of this section of DG-1145) provides a representative list of these items. Include the initial values of other variables and parameters in the application if they are used in the analyses of the particular event under study. Ensure that the parameters and initial conditions used in the analyses are suitably conservative for the event under study, but use realistic initial values for the ATWS analyses. Discuss the bases (including the degree of conservatism) used to select the numerical values of the input parameters. Appendix E (at the end of this section of DG-1145) gives further guidance regarding initial conditions and computer codes.

C.I.15.6.3.3 ***Results***

The applicant should present the results of the analyses, including key parameters as a function of time during the course of the transient or accident. The following are examples of parameters that should be included:

(1) neutron power

(2) thermal power

(3) heat fluxes, average and maximum

(4) reactor coolant system pressure

(5) minimum departure from nucleate boiling ratio (DNBR) or critical power ratio (CPR), as applicable

(6) core and recirculation loop coolant flow rates (BWRs)

(7) coolant conditions, including inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam volume fractions

(8) temperatures, including maximum fuel centerline temperature, maximum clad temperature, or maximum fuel enthalpy

(9) reactor coolant inventory, including total inventory and coolant level in various locations in the reactor coolant system

(10) secondary (power conversion) system parameters, including steam flow rate, steam pressure and temperature, feedwater flow rate, feedwater temperature, and steam generator inventory

(11) ECCS flow rates and pressure differentials across the core, as applicable

In addition, the results discussion should emphasize the margins between the predicted values of various core parameters, as well as the values of those parameters that would represent limiting acceptable conditions.

C.I.15.6.4 Barrier Performance

The applicant should discuss the evaluation of the parameters that may affect the performance of the barriers, other than fuel cladding, that restrict or limit the transport of radioactive material from the fuel to the public.

C.I.15.6.4.1 *Evaluation Model*

The applicant should present and discuss the evaluation model used to evaluate barrier performance. Provide the same types of information specified in the guidance in Section C.I.15.6.3.1. Include any simplifications or approximations introduced to perform the analyses. If the model is identical (or nearly identical) to that used to evaluate core performance, only describe the differences.

The applicant should provide a table listing the titles of TRs that describe models or computer codes used in transient and accident analyses, and list the associated NRC safety evaluation reports approving those TRs. Demonstrate that the use of the NRC-approved models or codes is within the applicable range and conditions of the models or codes. Provide a discussion to address compliance with each of the conditions and limitations in the NRC safety evaluation reports approving the TRs that document the models or codes used.

C.I.15.6.4.2 *Input Parameters and Initial Conditions*

The applicant should discuss any input parameters and initial conditions of variables that are relevant to the evaluation of barrier performance and were not discussed in Section C.I.15.6.3.2. Present the numerical values of inputs to the analyses, and discuss the adequacy of the selected values.

C.I.15.6.4.3 *Results*

The applicant should present and describe the results in detail. As a minimum, present the following information as a function of time during the course of the transient or accident:

(1) reactor coolant system pressure
(2) steam line pressure
(3) containment pressure
(4) relief and/or safety valve flow rate
(5) flow rate from the reactor coolant system to the containment system, if applicable

C.I.15.6.5 Radiological Consequences

The applicant should summarize the assumptions, parameters, and calculational methods used to determine the doses that result from accidents. Provide sufficient information to allow an independent analysis to be performed. Include all pertinent plant parameters that are required to calculate doses for the exclusion area boundary and low population zone, as well as those locations within the exclusion area boundary where significant site-related activities may occur (e.g., the control room).

The elements of the dose analysis that are applicable to several accident types or are used many times throughout Chapter 15 can be summarized (or cross-referenced) with the bulk of information appearing in appendices. If there are no radiological consequences associated with a given initiating event, include a statement indicating that containment of the activity was maintained and by what margin.

The applicant should provide an analysis for each limiting event, basing the analyses on design-basis assumptions acceptable to the NRC for purposes of determining the adequacy of the plant design to meet the criteria of 10 CFR Part 100 and 10 CFR 50.34. These design-basis assumptions, for the most part, can be found in regulatory guides that deal with radiological releases. For instance, when calculating the radiological consequences of a loss-of-coolant accident (LOCA), the NRC staff recommends using the assumptions given in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors" (as applicable to the plant design). This analysis should be designated as the "design-basis analysis."

There may be instances in which the applicant may not agree with the conservative margins inherent in the staff-approved design-basis approach, or may desire to provide a "realistic analysis" for comparison. In such instances, the applicant should state the assumptions that are adequately conservative. However, the applicant should use the known NRC assumptions in the design-basis analysis, and provide justification for any deviation from applicable regulatory guidance. Any "realistic analysis" provided will help quantify the margins that are inherent in the design-basis approach. A "realistic analysis" need not include a consequence assessment, and may be limited to presentation of assumptions that are more likely to be obtained than those used for purposes of design.

The applicant should present the parameters and assumptions used for these analyses, as well as the results, in tabular form. Appendix C (at the end of this section of DG-1145) provides a representative list of these items, although it is not intended to be all-encompassing with regard to the design-basis accidents analyzed or the parameters and assumptions that may be included in the table. Appendix D (at the end of this section of DG-1145) summarizes additional items that may be provided when dealing with specific types of accidents. When possible, provide the necessary quantitative information in a summary table. However, if a particular assumption cannot be simply or clearly stated in the table, reference a section or appendix that adequately discusses the assumption.

The applicant should use judgment in eliminating unnecessary parameters from the summary table or adding parameters of significance that do not appear in Appendices C or D at the end of this section of DG-1145. Include a summary table with one column for assumptions used in the design-basis analysis and one column for assumptions used in the realistic analysis.

The applicant should include as an appendix a diagram of the dose computation model, labeled "Containment Leakage Dose Model," as well as an explanation of that model. The purpose of this appendix is to clearly illustrate (1) the containment modeling, (2) the leakage or transport of radioactivity from one compartment to another or to the environment, and (3) the presence of engineered safety features (ESFs) such as filters or sprays that are called on to mitigate the consequences of a LOCA. Use easily identifiable symbols in the diagram, such as squares to represent the containment (or various portions thereof), lines with arrowheads drawn from one compartment to another or to the environment to indicate leakage or transport of radioactivity, and other suitably labeled or defined symbols to indicate the presence of ESF filters or sprays. Individual sketches (or equivalent) may be used for each significant time interval in the containment leakage history (e.g., separate sketches showing the pulldown of a dual containment annulus and the exhaust and recirculation phases once negative pressure in the annulus is achieved, with the appropriate time intervals given).

In presenting the assumptions and methodology used in determining the radiological consequences, the applicant should ensure that analyses are adequately supported with backup information, either by reporting the information where appropriate, by referencing other sections in the application, or by referencing documents that are readily available to the NRC staff. Include the following information:

(1) a description of the evaluation model used, including any simplifications or approximations introduced to perform the analyses

(2) an identification and description of any digital computer program used in the analysis (note that detailed descriptions of the evaluation models are preferably included by reference, with only summaries provided in the application)

(3) an identification of the time-dependent characteristics, activity, and release rate of the fission products or other transmissible radioactive materials within the containment system that could escape to the environment via leakages in the containment boundaries and leakage through lines that could exhaust to the environment

(4) considerations of uncertainties in calculational methods, equipment performance, instrumentation response characteristics, or other indeterminate effects taken into account in evaluating the results

(5) a discussion of the extent of system interdependency (containment system and other ESFs) that directly or indirectly contribute to controlling or limiting leakages from the containment system or other sources (e.g., from spent fuel handling areas), such as the following:

< containment water spray systems
< containment air cooling systems
< air purification and cleanup systems
< reactor core spray or safety injection systems
< postaccident heat removal systems
< main steam line isolation valve leakage control systems (BWR)

Present the results of the dose calculations giving the potential 2-hour integrated whole body and thyroid doses for the exclusion area boundary. Provide the doses for the course of the accident at the closest boundary of the low population zone and, when significant, the doses to control room operators during the course of the accident. Present other organ doses for those cases where solid fission products or transuranic elements are postulated to be released to the containment atmosphere.

(6) justification for any deviation from known NRC guidance on analysis of radiological consequences of accidents as applicable to the plant design, including assumptions and methodologies

Present the results of the dose calculations giving the maximum potential 2-hour integrated total effective dose equivalent (TEDE) for the exclusion area boundary. Provide the TEDE for the duration of the accident at the closest boundary of the low population zone and, when significant, the TEDE to control room operators for the duration of the accident.

Appendix A. Representative Initiating Events To Be Analyzed

15.0 Radiological Consequences Analyses

[The applicant may choose to group all design-basis accident (DBA) radiological consequences analyses under a single section, or discuss the radiological consequences of each accident under the following applicable sections. Standard Review Plan (SRP) 15.0.1 may be used until a new SRP 15.0.3 is written for new reactors.]

15.1 Increase in Heat Removal by the Secondary System

15.1.1 <u>Decrease in Feedwater Temperature as a Result of Feedwater System Malfunctions</u>

15.1.2 <u>Increase in Feedwater Flow as a Result of Feedwater System Malfunctions</u>

15.1.3 <u>Increase in Steam Flow as a Result of Steam Pressure Regulator Malfunction</u>

15.1.4 <u>Inadvertent Opening of a Steam Generator Relief or Safety Valve Steam Bypass Misoperation (Multiple Turbine Dump Valves)</u>

15.1.5 <u>Steam System Piping Failures Inside and Outside of Containment In a PWR, Including Lower Mode, Hot Zero Power, Hot Full Power, Pre-Trip Power Excursion, and Return-to-Critical Conditions</u>

15.2 Decrease in Heat Removal by the Secondary System

15.2.1 <u>Loss of External Load That Results in Decreasing Steam Flow</u>

15.2.2 <u>Turbine Trip (Stop Valve Closure)</u>

15.2.3 <u>Loss of Condenser Vacuum</u>

15.2.4 <u>Inadvertent Closure of Main Steam Isolation Valves (BWR)</u>

15.2.5 <u>Steam Pressure Regulator Failure (Closed)</u>

15.2.6 <u>Loss of Non-Emergency AC Power to the Station Auxiliaries</u>

15.2.7 <u>Loss of Normal Feedwater Flow</u>

15.2.8 <u>Feedwater System Piping Breaks Inside and Outside Containment</u>

15.3 Decrease in Reactor Coolant System Flow Rate

15.3.1 Single and Multiple Reactor Coolant Pump Trips

15.3.2 Flow Controller Malfunctions

15.3.3 Reactor Coolant Pump Shaft Seizure

15.3.4 Reactor Coolant Pump Shaft Break

15.4 Reactivity and Power Distribution Anomalies

15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low-Power Startup Condition (Assuming the Most Unfavorable Reactivity Conditions of the Core and Reactor Coolant System), Including Single Control Rod, Bank of Control Rods, and Temporary Control Device Removal Error During Refueling

15.4.2 Uncontrolled Control Rod Assembly Withdrawal at the Particular Power Level (Assuming the Most Unfavorable Reactivity Conditions of the Core and Reactor Coolant System) That Yields the Most Severe Results (Subcritical Through Full-Power)

15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)

15.4.4 Startup of an Inactive Reactor Coolant Loop or Recirculating Loop At an incorrect Temperature

15.4.5 Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate

15.4.6 Chemical and Volume Control System Malfunction That Results in A decrease in Boron Concentration in the Reactor Coolant of a PWR

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

15.4.8 Spectrum of Rod Ejection Accidents in a PWR

15.4.8A Radiological Consequences of a Control Rod Ejection Accident (PWR) (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)

15.4.9 Spectrum of Rod Drop Accidents in a BWR

15.4.9A Radiological Consequences of a Control Rod Drop Accident (BWR) (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)

15.5 Increase in Reactor Coolant Inventory

15.5.1 Inadvertent Operation of ECCS During Power Operation

15.5.2 Chemical and Volume Control System Malfunction (Or Operator Error) That Increases Reactor Coolant Inventory

15.5.3 A Number of BWR Transients, Including Items 15.2.1 Through 15.2.6 and Item 15.1.2

15.6 Decrease in Reactor Coolant Inventory

15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve in a Pwr Or a safety or Relief Valve in a BWR

15.6.2 Radiological Consequences of the Steam Generator Tube Failure (May not be Necessary If Discussed above under 15.0 or New SRP 15.0.3)

15.6.3 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR) (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)

15.6.4 LOCAs Resulting from the Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary, Including Steam Line Breaks Inside Containment in a BWR

15.6.5A Radiological Consequences of a Design-Basis LOCA, Including Containment Leakage Contribution (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)

15.6.5B Radiological Consequences of a Design-Basis LOCA, Including Leakage from Engineered Safety Feature Components Outside Containment (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)

15.6.5C Radiological Consequences of a Design-Basis LOCA, Including Leakage from Main steam Isolation Valve Leakage Control System (BWR) (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)

15.6.6 A Number of BWR Transients, Including Items 15.2.7, 15.2.8, and 15.4.6 a Boron Dilution in the Reactor Coolant of a PWR

15.7 Radioactive Release from a Subsystem or Component

15.7.1 Postulated Radioactive Releases Attributable to Liquid Tank Failures

15.7.2 Radiological Consequences of a Fuel Handling Accident (may not be necessary if discussed above under 15.0 or new SRP 15.0.3)

15.7.3 Spent Fuel Cask Drop Accidents

15.8 Anticipated Transients Without Scram

15.8.1 <u>Loss of Feedwater</u>

15.8.2 <u>Loss of Electrical Load</u>

15.8.3 <u>Turbine Trip</u>

15.8.4 <u>Loss of Condenser Vacuum</u>

15.8.5 <u>Loss of Offsite Power</u>

15.8.6 <u>Closure of Main Steam Line Isolation Valves</u>

15.8.7 <u>Inadvertent Control Rod Withdrawal</u>

Appendix B. Typical Input Parameters and Initial Conditions for Transients and Accidents

- Neutron Power
- Moderator Temperature Coefficient of Reactivity
- Moderator Void Coefficient of Reactivity
- Doppler Coefficient of Reactivity
- Effective Neutron Lifetime
- Delayed Neutron Fraction
- Average Heat Flux
- Maximum Heat Flux
- Minimum Departure from Nucleate Boiling Ratio (DNBR) or Critical Power Ratio (CPR)
- Axial Power Distribution
- Radial Power Distribution
- Core Coolant Flow Rate
- Recirculation Loop Flow Rate (BWR)
- Core Coolant Inlet Temperature
- Core Average Coolant Temperature (PWR)
- Core Average Steam Volume Fraction (BWR)
- Core Coolant Average Exit Temperature, Steam Quality, and Steam Void Fraction
- Hot Channel Coolant Exit Temperature, Steam Quality, and Steam Void Fraction
- Maximum Fuel Centerline Temperature
- Reactor Coolant System Inventory
- Coolant Level in Reactor Vessel (BWR)
- Coolant Level in Pressurizer (PWR)
- Reactor Coolant Pressure
- Steam Flow Rate
- Steam Pressure
- Steam Quality (temperature if superheated)
- Feedwater Flow Rate
- Feedwater Temperature
- Chemical and Volume Control System (CVCS) Flow and Boron Concentration (if these vary during the course of the transient or accident being analyzed)
- Control Rod Worth, Differential, and Total
- Standby Liquid Control System (SLCS) Flow and Boron Concentration (BWR)
- ECCS Flow

Appendix C. Representative Parameters To Be Tabulated for Postulated Accident Analyses

(1) Data and assumptions used to estimate radioactive source from postulated accidents

- (a) Power level
- (b) Burn-up
- (c) Percent of fuel perforated
- (d) Release of activity by nuclide
- (e) Iodine fractions (organic, elemental, and particulate)
- (f) Reactor coolant activity before the accident (and secondary coolant activity for PWR). Give the following two values for primary system iodine activity concentration:
  - (i) maximum allowable equilibrium iodine concentration
  - (ii) maximum allowable concentration resulting from a pre-accident iodine spike

(2) Data and assumptions used to estimate activity released

- (a) Primary containment volume and leak rate
- (b) Secondary containment volume and leak rate
- (c) Valve movement times
- (d) Adsorption and filtration efficiencies
- (e) Recirculation system parameters (flow rates versus time, mixing factor, etc.)
- (f) Containment spray first order removal lambdas as determined in Section 6.2.3
- (g) Containment volumes
- (h) Natural deposition and plateout factors or effective decontamination factors for containment and/or piping
- (i) All other pertinent data and assumptions

(3) Dispersion Data

- (a) Location of points of release
- (b) Distances to applicable receptors (e.g., control room, exclusion boundary, and LPZ)
- (c) atmospheric dispersion factors (χ/Q) at control room, exclusion boundary, and LPZ (for time intervals of 2 hours, 8 hours, 24 hours, 4 days, 30 days)

(4) Dose Data

- (a) Method of dose calculation
- (b) Dose conversion assumptions
- (c) Peak [or f(t)] concentrations in containment
- (d) Doses (TEDE for EAB, LPZ and control room)

Appendix D. Additional Parameters and Information to be Provided or Referenced in the Summary Tabulations for Specific Design Basis Accidents

(1) Loss-of-Coolant Accident

(a) Hydrogen Purge Analysis

(i) Holdup time prior to purge initiation (assuming recombiners are inoperative)

(ii) Iodine reduction factor

(iii) χ/Q values at appropriate time of release

(iv) Purge rates for at least 30 days after initiation of purge

(v) LOCA plus purge dose at the low-population zone (LPZ)

(b) Equipment Leakage Contribution to LOCA Dose

(i) Iodine concentration in sump water after LOCA

(ii) Maximum operational leak rate through pump seals, flanges, valves, etc.

(iii) Maximum leakage assuming failure and subsequent isolation of a component seal

(iv) Total leakage quantities for (2) and (3)

(v) Temperature of sump water vs. time

(vi) Time intervals for automatic and operator action

(vii) Leak paths from point of seal or valve leakage to the environment

(viii) Iodine partition factor for sump water vs temperature of water

(ix) Charcoal absorber efficiency assumed for iodine removal

(c) Main Steam Line Isolation Valve Leakage Contribution to LOCA Dose (BWR)

(i) Time of leakage control system actuation, if applicable

(ii) Fraction of isolation valve leakage from each release point

(iii) Flow rates vs. time for each release path

(iv) Location of each release point

(v) Transport time to each release point

(vi) Iodine removal constants or decontamination factors, by either the leakage control system or deposition and plateout, as applicable

(2) Main Steam Line and Steam Generator Tube Failures

(a) Characterization of the primary and secondary (PWR) system:
Give sufficient information to adequately describe the time histories from accident initiation until accident recovery is complete for temperatures, pressures, steam generator water capacity, steaming rates, feedwater rates, blowdown rates, and primary-to-secondary leakage rates.

(b) Potential increase in iodine release rate above the equilibrium value (i.e., iodine spiking) from the fuel to the primary coolant as a result of the accident or a pre-accident primary

system transient

(c) Chronological list of system response times, operator actions, valve closure times, etc.

(d) Steam and water release quantities and all assumptions made in their computation

(e) Description of the iodine transport mechanism and release paths between the primary system and the environment (describe and justify the bases for an assumed partitioning of iodine between liquid and steam phases)

(f) Possible fuel rod failure resulting from the accident, assuming the most reactive control rod remains in its fully withdrawn position

(g) Possible steam generator tube failure resulting from a PWR steam line break accident

(3) Fuel Handling Accident (in the Containment and Spent Fuel Storage Buildings)

(a) Number of fuel rods in core

(b) Number, burnup, and decay time of fuel rods assumed to be damaged in the accident

(c) Radial peaking factor for the rods assumed to be damaged

(d) Earliest time after shutdown that fuel handling begins

(e) Amounts of iodines and noble gases released into pool

(f) Pool decontamination factors

(g) Time required to automatically switch from normal containment purge operation to either safety-grade filters or isolation

(h) Amount of radioactive release not routed through ESF-grade filters

(i) Maximum fuel rod pressurization

(j) Minimum water depth between top of the fuel rods and fuel pool surface

(k) Peak linear power density for the highest power assembly discharged

(l) Maximum centerline operating fuel temperature for the fuel assembly in item k above

(m) Average burnup for the peak assembly in item k above

(4) Control Rod Ejection and Control Rod Drop Accidents

(a) Percent of fuel rods undergoing clad failure

(b) Radial peaking factors for rods undergoing clad failure

(c) Percent of fuel reaching or exceeding melting temperature

(d) Peaking factors for fuel reaching or exceeding melting temperature

(e) Percent of core fission products assumed released into reactor coolant

(f) Summary of primary and secondary system parameters used to determine the activity release through the secondary system (PWRs only) (provide the information specified in items 3a–e of this table)

(g) Summary of containment system parameters used to determine activity release terms from containment leak paths

(h) Summary of system parameters and decontamination factors used to determine activity

release from condenser leak paths (BWR)

(5) Spent Fuel Cask Drop

(a) Number of fuel elements in largest capacity cask

(b) Number, burnup, and decay time of fuel elements in cask assumed to be damaged

(c) Number, burnup, and decay time of fuel elements in pool assumed to be damaged as a consequence of a cask drop (if any)

(d) Average radial peaking factor for the rods assumed to be damaged

(e) Earliest time after reactor fueling that cask loading operations begin

(f) Amounts of iodines and noble gases released into air and into pool

(g) Pool decontamination factors, if applicable

Appendix E. Summary of Initial Conditions and Computer Codes Used

Provide (in tabular form) a summary of the computer codes used, as well as the reactivity coefficients (e.g., moderator density, moderator temperature, and Doppler coefficients) and initial thermal power assumed in the analysis of each transient or accident.

Appendix F. Nominal Values of Pertinent Plant Parameters used in the Accident Analyses

Provide (in tabular form) the reactor trip functions, engineered safety feature functions, and other equipment available to mitigate each transient and accident.

Appendix G. Safety Analysis RPS and ESFAS Trip Setpoints and Delay Times

Provide (in tabular form) a summary of the trip setpoints, total delay times of the reactor protection system and engineered safety features actuation system assumed in the analyses of the transients and accidents. The table should also include the trip setpoint values specified in the Technical Specifications.

Appendix H. Single Failures

Provide (in tabular form) all single failures considered to determine the limiting single failure used in each transient or accident analyzed.

Appendix I. Limiting Single Failures Assumed in Transient and Accident Analyses

Provide (in tabular form) the limiting single failure selected for each transient and accident analyzed.

Appendix J. Non-Safety Related System and Equipment Used To Mitigate Transients and Accidents

Provide (in tabular form) a list of non-safety related system and equipment used to mitigate transients and accidents.