## 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 LCOs, 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 COL reviews.

To support its application, the COL applicant should discuss the applicable transient and accident analyses and justify its conformance to the regulations (as specified in Appendix A to this section of the regulatory guide). The regulatory requirements at 10 CFR 52.79 and 10 CFR 52.80 provide the technical contents of a COL application. Section 15 of the SRP, as amended, discusses specific acceptance criteria for each transient. Particularly, the relevant requirements include 10 CFR 50.34, "Contents of Applications; Technical Information," 10 CFR 50.34(a)(1)(ii), 10 CFR 50.34(f)(1)(ii), and 10 CFR 50.34(f)(2)(xii); 10 CFR 50.46; 10 CFR 50.49; Appendix A to 10 CFR Part 50; GDC 10, "Reactor Design"; GDC 13; GDC 15; GDC 17; GDC 19; GDC 20, "Protection System Functions"; GDC 25, "Protection System Requirements for Reactivity Control Malfunctions"; GDC 26, "Reactivity Control System Redundancy and Capability"; GDC 27; GDC 28, "Reactivity Limits"; GDC 29; GDC 31; GDC 34; GDC 35; GDC 55; GDC 60; GDC 61, "Fuel Storage and Handling and Radioactivity Control"; Appendix E "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50, paragraph IV.E.8; Appendix K, "Emergency Core Cooling Systems Evaluation Models," to 10 CFR Part 50; 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions"; 10 CFR Part 100; and 10 CFR 100.21

The COL applications should discuss how the design and analysis of events comply with the applicable TMI Action Plan Items in NUREG-0737 and NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses." Applicable TMI Action Plan items are shown in individual SRP sections of NUREG-0800, in relation to this chapter and the list includes 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. These demonstrate compliance with any technically relevant portions of the TMI-related requirements set forth in 10 CFR 50.34(f)(1)(vi) and 10 CFR 50.34(f)(1)(iii), for DC and COL reviews, respectively.

The term "Unresolved Safety Issues" refers to the USIs contained in Section 2, "Task Action Plan Items," of NUREG-0933. The status of cited USIs has not been modified for the purpose of the COL application. Applicants are required by 10 CFR 52.79 to address technical resolutions of all USIs that are technically relevant to their design.

The application should discuss how the design and analysis of applicable events incorporate the resolution of USIs and medium- and high-priority GSIs identified in the version of NUREG-0933, that is current 6 months before the application submittal date. The application should also discuss 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, the applicant should demonstrate that the applicable system design and transient and accident analyses incorporate the operating experience insights from generic letters and bulletins issued up to 6 months before the submittal date of the application or fully justify acceptable alternatives. Applicable generic letters and bulletins include the following:

- GL-80-19, "Resolution of Enhanced Fission Gas Release Concern," March 10, 1988
- GL-80-35, "Effect of a DC Power Supply Failure on ECCS Performance," April 25, 1980
- GL-83-11, "Licensee Qualification for Performing Safety Analysis in Support of Licensing Actions," February 8, 1983
- GL-83-22, "Safety Evaluation of 'Emergency Response Guidelines,'" June 3, 1983
- GL-83-32, "NRC Staff Recommendations Regarding Operator Action for Reactor Trip and ATWS," December 2, 1983
- GL-85-06, "Quality Assurance Guidance for ATWS Equipment That Is Not Safety-Related," January 16, 1985
- GL-85-16, "High Boron Concentrations"
- GL-86-13, "Potential Inconsistency between Plant Safety Analyses and Technical Specifications," July 23, 1986
- GL-86-16, "Westinghouse ECCS Evaluation Models," October 22, 1986
- GL-88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 3, 1988
- GL-88-17, "Loss of Decay Heat Removal"
- GL-93-04, "Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies, 10 CFR 50.54(f)," June 21, 1993
- GL-97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," April 1, 1997
- GL-98-02, "Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition," May 28, 1993
- GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," September 13, 2004
- BL-80-04, "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition," February 8, 1980
- BL-80-12, "Decay Heat Removal System Operability," May 9, 1980
- BL-80-18, "Maintenance of Adequate Minimum Flow thru Centrifugal Charging Pumps Following Secondary Side High Energy Line Rupture"
- BL-86-03, "Potential Failure of Multiple ECCS Pumps due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line"
- BL-93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers"
- BL-95-02, "Unexpected Clogging of a Residual Heat Removal"
- BL-96-01, "Control Rod Insertion Problems," March 8, 1996

- BL-96-03, “Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors”
- BL-2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles”

It should be noted that the items listed in the above lists of GDC, TMI action items, USIs, GSIs, generic letters, and bulletins may not constitute the total set of relevant requirements or guidance. It is the COL applicant’s responsibility to identify all relevant items applicable to its reactor design (see Section C.IV.8-1).

### C.I.15.1 <u>Transient and Accident Classification</u>

The application should organize the transients and accidents and present the results that will (1) ensure that the applicant has considered a sufficiently broad spectrum of initiating events, (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. The applicant should 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 RCS 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

An additional event category is ATWS. ATWS events are events in which the reactor scram system (the reactor protection and reactivity control systems) is postulated to fail to operate as required following anticipated transients such as a loss of feedwater, loss of load, turbine trip, inadvertent control rod withdrawal, loss of ac, loss of condenser vacuum in PWRs, and a closure of main steamline isolation valves in BWRs. ATWS events per se are beyond-DBEs as they are anticipated initiating transients plus the failure of an automatic scram resulting from common-mode failure of the scram system. The ATWS rule promulgated in 10 CFR 50.62 specifies the requirements for the operating PWRs and BWRs of various vendor designs to reduce the probability of unacceptable consequences resulting from ATWS events. The NRC based these requirements on the results of the staff studies of the operating PWRs and BWRs under licensed conditions at the time of the ATWS rulemaking. With new reactor designs and power uprates of the operating reactors, evaluation is necessary to determine if these requirements remain appropriate and sufficient. Even though ATWS events are beyond-DBEs and are subject to PRA, the physics and thermal-hydraulic phenomena of the plant response to ATWS events should be evaluated. Therefore, ATWS events are included as a category of postulated accident analyses.

Appendix A to this section of the regulatory guide presents typical initiating events. 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 to this section of the regulatory guide 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) AOOs, as defined in Appendix A to 10 CFR Part 50 and categorized in RG 1.70, 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 are not expected to occur.

The applicant should evaluate initiating events for each combination of category and frequency group 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, the applicant does not need to analyze completely every postulated initiating event. 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 is considered. For example, within a given category/frequency group combination, one initiating event might result in the highest 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 design certification 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 ECCS initiation. In evaluating AOOs and postulated accidents, the performance of each credited protection system is required to include the effects of the most limiting single active failures. This verifies the satisfaction of the GDC criteria that require protection systems to adequately perform their intended safety function in the presence of single active failures. List the expected limiting delay time for each protection system function, and describe the acceptable methodology for determining uncertainties (e.g., from combined effect of calibration error, drift, instrumentation error) to be included in the establishment of the trip setpoints and allowable values specified in the plant TS.

## 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.I.15.6.1 and C.I.15.6.2 of this regulatory guide for each initiating event. The extent of the quantitative information necessary to complete Sections C.I.15.6.3–C.I.15.6.5 to be included in the FSAR may differ for the various initiating events. For an initiating event that is not limiting, the applicant need only present the qualitative reasoning that led to that conclusion, 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 (C.I.15.6.3-C.I.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, the applicant need not perform a detailed evaluation of the radiological consequences for each initiating event.

## C.I.15.6 Event Evaluation

### C.I.15.6.1 *Identification of Causes and Frequency Classification*

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 *Sequence of Events and Systems Operation*

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 setpoint, 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 RPS are required to function

(4) credit taken for the functioning of normally operating plant systems

(5) operation of engineered safety systems that is required

(6) consistency between the safety analyses and the emergency response guidelines/emergency procedure guidelines or EOP with respect to the operator response (including action time) and available instrumentation

Only safety-related systems or components should be used to mitigate transient or accident conditions. However, analyses may assume that nonsafety-related systems or components are operable for the following cases:

(1) when a detectable and nonconsequential random and independent failure must occur in order to disable the system

(2) when nonsafety-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 DBEs 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 nonsafety-related systems or components credited in the design-basis analyses for mitigating the event consequences, the applicant must provide proper justification. The application must take into account nonsafety-related systems or components that may adversely affect transient or accident analyses. List the nonsafety-related systems or components assumed in the analyses for each event in a tabular form as recommended in Appendix J to this section of the regulatory guide. The applicant should provide a discussion of how the definitions for active and passive failures, as described in SECY-77-439, "Single-Failure Criterion," issued August 1977 have been applied to the analyses. For passive system designs, applicants should follow the guidance of the SECY-94-84, and ensure that low differential pressure check valves that perform a safety function are considered active components subject to single active failure consideration, except when their proper function can be demonstrated and documented. This is in general considered to enhance overall system reliability.

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 modes 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 that describe models or computer codes used in transient and accident analyses and listing the associated NRC safety evaluation reports approving those topical reports. The applicant should ensure that these referenced topical reports are also included in the table of material referenced that should be provided in FSAR Section 1.6. 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 topical reports 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 to this section of the regulatory guide 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 to this section of the regulatory guide 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) RCS pressure

(5) minimum departure from nucleate boiling ratio or critical power ratio, 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 RCS

(10) secondary (power conversion) system parameters, including steamflow 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 and 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 topical reports that describe models or computer codes used in transient and accident analyses and listing the associated NRC safety evaluation reports approving those topical reports. The applicant should ensure that these referenced topical reports are also included in the table of material referenced that should be provided in FSAR Section 1.6. 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 topical reports 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 that 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) RCS pressure
(2) steamline pressure
(3) containment pressure
(4) relief and/or safety valve flow rate
(5) flow rate from the RCS 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 EAB and LPZ, as well as those locations within the EAB where significant site-related activities may occur (e.g., the control room).

The applicant can summarize (or cross-reference) elements of the dose analysis that are applicable to several accident types or are used many times throughout Chapter 15 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, 10 CFR 50.34, and 10 CFR 52.79. 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 LOCA, the NRC staff recommends using the assumptions given in RG 1.183 (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 to this section of the regulatory guide provides a representative list of these items, although it is not intended to be all-encompassing with regard to DBA analyzed or the parameters and assumptions that may be included in the table. Appendix D to this section 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 to this section of the regulatory guide. 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 figure 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 ESFs such as filters or sprays that are relied upon 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
- MSIVLCS (BWR)

(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 TEDE for the EAB; provide the TEDE for the duration of the accident at the closest boundary of the LPZ and, when significant, the TEDE to control room operators for the duration of the accident.

## Appendix C.I.15-A

### Representative Initiating Events to be Analyzed

| | |
|---|---|
| 15.0 | Introduction - Transient and Accident Analyses Review Responsibilities (The applicant may choose to group all DBA radiological consequences analyses under a single section, or discuss the radiological consequences of each accident under the following applicable sections). |
| 15.0.1 | Radiological Consequence Analyses Using Alternative Source Terms (applicable to operating reactors licensed before issuance of this regulatory guide) |
| 15.0.2 | Review of Transient and Accident Analysis Methods |
| 15.0.3 | Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors |
| 15.1.1-15.1.4 | Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve |
| 15.1.5A | Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR (may not be necessary if discussed above under SRP 15.0.3). |
| 15.2.1 - 15.2.5 | Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed) |
| 15.2.6 | Loss of Nonemergency AC Power to the Station Auxiliaries |
| 15.2.7 | Loss of Normal Feedwater Flow |
| 15.2.8 | Feedwater System Pipe Breaks inside and outside Containment (PWR) |
| 15.3.1 - 15.3.2 | Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions |
| 15.3.3 - 15.3.4 | Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break |
| 15.4.1 | Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low-Power Startup Condition |
| 15.4.2 | Uncontrolled Control Rod Assembly Withdrawal at Power |
| 15.4.3 | Control Rod Misoperation (System Malfunction or Operator Error) |
| 15.4.4 - 15.4.5 | Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate |
| 15.4.6 | Inadvertent Decrease in Boron Concentration in the Reactor Coolant (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 (Pressurized-Water Reactor) (may not be necessary if discussed above under SRP 15.0.3)

15.4.9 Spectrum of Rod Drop Accidents in a Boiling-WaterReactor

15.4.9A Radiological Consequences of a Control Rod Drop Accident (Boiling-Water Reactor) (may not be necessary if discussed above under SRP 15.0.3)

15.5.1 - 15.5.2 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

15.6.1 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve

15.6.2 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment (may not be necessary if discussed above under SRP 15.0.3)

15.6.3 Radiological Consequences of a Steam Generator Tube Failure (PWR)

15.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)

15.6.5 Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

15.6.5A Radiological Consequences of a Design-Basis Loss-of-Coolant Accident, Including Containment Leakage Contribution (may not be necessary if discussed above under SRP 15.0.3)

15.6.5B Radiological Consequences of a Design-Basis Loss-of-Coolant Accident, Including Leakage from Engineered Safety Feature Components outside Containment (may not be necessary if discussed above under SRP 15.0.3)

15.6.5D Radiological Consequences of a Design-Basis Loss-of-Coolant Accident, Including Leakage from Main Steam Isolation Valve Leakage Control System (Boiling-Water Reactor) (may not be necessary if discussed above under SRP 15.0.3)

15.7.4 Radiological Consequences of Fuel Handling Accidents (may not be necessary if discussed above under SRP 15.0.3).

15.7.5 Spent Fuel Cask Drop Accidents

15.8 Anticipated Transients Without Scram[1]

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[1] ATWS events are beyond-DBEs. See Section C.I.15.1 for a detailed discussion.

15.9 BWR Core Stability

15.x Spent Fuel Pool Criticality and Boron Dilution Analysis

## Appendix C.I.15-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 or critical power ratio
- 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)
- RCS
- steamflow rate
- steam pressure
- steam quality (temperature if superheated)
- feedwater flow rate
- feedwater temperature
- chemical and volume control system 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 flow and boron concentration (BWR)
- ECCS flow

## Appendix C.I.15-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) burnup
- (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); including the following two values for primary system iodine activity concentration:
  - (i) maximum allowable equilibrium iodine concentration
  - (ii) maximum allowable concentration resulting from a preaccident 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 (e.g., flow rates versus time, mixing factor)
- (f) containment spray first order removal lambdas as determined in Section 6.5.2.3
- (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, EAB, and LPZ)
- (c) atmospheric dispersion factors ($\chi/Q$) at control room, EAB, 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 the EAB, LPZ, and control room)

## Appendix C.I.15-D

### Additional Parameters and Information to be Provided or Referenced in the Summary Tabulations for Specific Design-Basis Accidents

(1) loss-of-coolant accident (LOCA)

- (a) hydrogen purge analysis
  - (i) holdup time prior to purge initiation (assuming recombiners are inoperative)
  - (ii) iodine reduction factor
  - (iii) $\chi/Q$ 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 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, and other sources
  - (iii) maximum leakage assuming failure and subsequent isolation of a component seal
  - (iv) total leakage quantities for (ii) and (iii)
  - (v) temperature of sump water versus 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 versus temperature of water
  - (ix) charcoal absorber efficiency assumed for iodine removal
- (c) main steamline 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 versus 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 steamline 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 preaccident primary system transient

(c) chronological list of system response times, operator actions, valve closure times, and other parameters

(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 steamline 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 3(a–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

## Appendices C.I.15-E - J

*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, ESF 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 and the total delay times of the reactor protection system and ESF actuation system assumed in the analyses of the transients and accidents. The table should also include the trip setpoint values specified in the TS.

*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 Nonsafety-Related System and Equipment Used To Mitigate Transients and Accidents*

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