DG-1145, Section C.1.15 - Transient and Accident Analyses

15.0 Introduction

As with other chapters of this Regulatory Guide (RG), some policies and procedures will not be available at the time the combined operating license (COL) application will be submitted. In those cases, make a commitment in the application with a summary description of the procedures to be available by fuel load. Include a discussion of how the design meets the applicable regulatory requirements and regulatory guidance available.

The evaluation of the safety of a nuclear power plant includes analyses of the response of the plant to postulated disturbances in process variables and to postulated malfunctions or failures of equipment. 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 COL and operating license reviews of plants.

To support its DC or COL applications, the applicant should discuss the applicable transient and accident analyses in Table 15.1 and justify its conformance to the regulations. Specific acceptance criteria for each transient are discussed in the respective Standard Review Plan, Section 15, as amended.

The relevant requirements specified in Title 10 Code of Federal Regulations (10 CFR) include the following:

10 CFR 50.34(a)(1)(ii).
10CFR 50.34(f)(1)(ii) and 10CFR50.34.34(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 50, Appendix A, Criterion 10, “Reactor design.”
10 CFR 50, Appendix A, Criterion 13, “Instrumentation and Control.”
10 CFR 50, Appendix A, Criterion 15, “Reactor Coolant system Design.”
10 CFR 50, Appendix A, Criterion 17, “Electric Power Systems.”
10 CFR 50, Appendix A, Criterion 19, “Control Room.”
10 CFR 50, Appendix A, Criterion 20, “Protection System Functions.”
10 CFR 50, Appendix A, Criterion 25, “Protection System Requirements for Reactivity Control Malfunctions.”
10 CFR 50, Appendix A, Criterion 26, “Reactivity Control System Redundancy and Capability.
10 CFR 50, Appendix A, Criterion 27, “Combined Reactivity Control Systems Capability.”
10 CFR 50, Appendix A, Criterion 28, “Reactivity Limits.”
10 CFR 50, Appendix A, Criterion 29, “Protection Against Anticipated Operational Occurrences.”
10 CFR 50, Appendix A, Criterion 31, “Fracture Prevention of Reactor Coolant Pressure Boundary.”
10 CFR 50, Appendix A, Criterion 35, “Emergency Core Cooling.”

10 CFR 50, Appendix A, Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."
10 CFR 50, Appendix A, Criterion 60, "Control of Releases of Radioactive Materials to the Environment."
10 CFR 50, Appendix A, Criterion 61, "Fuel Storage and Handling and Radioactivity Control."
10 CFR 50, Appendix E, Paragraph IV.E.8, "Emergency Planning and Preparedness for Production and Utilization Facilities."
10 CFR 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 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 III.A.1.2, II.B.2, II.B.3, I.C.9, III.D.1.1, III.D.3.4, II.E.1.1, II.E.2.3, II.E.5.1, II.E.5.2, II.F.1, II.K.2.16, II.K.2.17, II.K.2.19, II.K.2.8, II.K.3.1, II.K.3.2, II.K.3.13, II.K.3.16, II.K.3.21, II.K.3.24, II.K.3.25, II.K.3.28, II.K.3.30, II.K.3.31, II.K.3.44,II.K.3.45, II.K.3.5, II.K.3.40, and II.K.3.7.

Discuss how the design and analysis of applicable events incorporate the resolution of the unresolved safety issues (USIs) and medium and high priority generic safety issues (GSIs) that are identified in the version of NUREG-0933 updated six months before application date and are technically relevant to the applicable systems design, and transient and accident analysis. 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-70, GSI-75, GSI-87, GSI-93, GSI-99, GSI-122.2, GSI-124, GSI-125.II.7, GSI-135, GSI-185, and GSI-191.

In addition, demonstrate that operating experience insights from generic letters (GLs) and bulletins (BLs) issued up to six months before the docket date of the application have been incorporated into the applicable systems design, and transient and accident analysis. Applicable GLs and BLs include GL-80-001, GL-80-019, GL-80-035, GL-82-08, GL-83-11, GL-83-22, GL-83-28, GL-83-32, GL-85-06, GL-85-13, GL-85-16, GL-86-13, GL-86-16, GL-88-16, GL-88-17, GL-89-10, GL-89-13, GL-91-07, GL-93-04, GL-97-01, GL-97-04, GL-98-02, BL-80-04, BL-80-12, BL-80-18, BL-86-01, BL-86-03, BL-88-04, BL-93-02, BL-95-02, BL-96-01, BL-96-03, BL-2001-01, and BL-2002-02.

15.1 Transient and Accident Classification

Organize the transients and accidents and present the results in that will: (1) assure that a sufficiently broad spectrum of initiating events has been considered; (2) demonstrate that the event chosen for analysis is indeed the limiting event and 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 the consistent application of specific acceptance criteria for each postulated initiating event; and (4) identify which transients or accidents are fuel design dependent that are to be analyzed in every fuel cycle.

To accomplish these goals, a number of disturbances of process variables and malfunctions or failures of equipment are postulated. Assign each postulated initiating event 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, and
8. Anticipated transients without scram (ATWS).

Typical initiating events are presented in Table 15-1. For new reactor designs, evaluate the need for additional initiating events not included in Table 15-1. Evaluate each initiating event using the outline in Section 15.6. Tables 15-2 to 15-10 provide guidance that may be useful in presenting the information for the transient and accident analyses.

15.2 Frequency of Occurrence

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

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

b. Accidents are occurrences that are not expected to occur but are postulated.

Different initiating events in the same category/frequency group 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.

15.3 Plant Characteristics Considered in the Safety Evaluation

Summarize 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. Assure the range is sufficiently broad to cover all expected changes predicted for the entire life of the plant. 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.

15.4 Assumed Protection System Actions

List the settings of all protection system functions that are used in the safety evaluation. Typical protection system functions are reactor trips, isolation valve closures, and ECCS initiation. List the uncertainty (combined effect of calibration error, drift, instrument error, etc.) associated with each function together with the expected limiting delay time.

15.5 Evaluation of Individual Initiating Events

Provide an evaluation of each initiating event using the format in Section 15.6. Indicate if an initiating event is applicable to more than one category. Provide the information listed in Section 15.6.1 and 15.6.2 for each initiating event. The extent of the quantitative information to provide in Section 15.6.3 through 15.6.5 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 of Section 15.6.3 through 15.6.5. For example, there are a number of plant transient initiating events that result in minimal radiological consequences. Present a qualitative evaluation to show this to be the case. A detailed evaluation of the radiological consequences need not be performed for each initiating event.

15.6 Event Evaluation

15.6.1 Identification of Causes and Frequency classification

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

15.6.2 Sequence of Events and Systems Operation

Discuss for each initiating event:

a. The step-by-step sequence of events from its initiation to the final stabilized condition. Identify each significant occurrence on a time scale, e.g., flux monitor trip, insertion of control rods begin, primary coolant pressure reaches safety valve set point, safety valves open, safety valves close, containment isolation signal initiated, and containment isolated. Identify all required operator actions. .
b. The extent to which normally operating plant instrumentation and controls are assumed to function.

c. The extent to which plant and reactor protection systems are required to function.
d. The credit taken for the functioning of normally operating plant systems.
e. The operation of engineered safety systems that is required.
f. Discuss the basis in the Emergency Operating Procedures (EOP) for operator response, available instrumentation, and timing.

Only safety-related systems or components can be used for mitigation of transient or accident analysis. However, if non-safety related systems or components are credited in the analysis, proper justification must be provided (e.g., list the credited non-safety related systems or components for each event in tabular form as recommended in Table 15-10). Non-safety related systems or components that may adversely affect transient or accident analyses must be taken into account. Discuss active and passive failures. NRC's SECY-77-439 provides a description of active and passive failures. The definitions used in ANSI/ANS-58.14-1993 or later version are acceptable. For a passive safety system design that utilizes low differential pressure check valves that perform a safety function, except for those whose proper function can be demonstrated and documented, the check valves must be considered an active component subject to single active failure consideration.

Evaluate the effect of single active failures and the effect of operator errors in each of the above areas. Provide enough detail to permit an independent evaluation of the adequacy of the system as related to the event under study. One method of systematically investigating single failures is the use of a plant operational analysis or a failure mode and effects analysis. List all the 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.

15.6.3 Core and System Performance

a. Evaluation Model

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 are used, describe the method combining 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, include figures showing the analytical model, flow path identification, actual computer listing, and listing of major input data. The detailed description of evaluation models and digital computer codes or listings are preferably included by reference to documents available to the NRC with only summaries provided in the application text.

Provide a table with a list of 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 the 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 the compliance with each of the conditions and limitations in the NRC safety evaluation reports approving the TRs that document the models or codes used.

b. Input Parameters and Initial Conditions

Identify the major input parameters and initial conditions used in the analyses. Table 15-2 provides a representative list of these items. Include the initial values of other variables and additional parameters in the application if they are used in the analyses of the particular event being analyzed. Ensure the parameters and initial conditions used in the analyses are suitably conservative for the event being evaluated except use realistic initial values for the anticipated transient without scram (ATWS) analyses. Discuss the bases used to select the numerical values of the input parameters, including the degree of conservatism. Table 15-5, "Summary of Initial Conditions and Computer Codes Used," gives further guidance.

c. Results

Present the results of the analyses. Present key parameters as a function of time during the course of the transient or accident. The following are examples of parameters to be included:

(1) Neutron power,
(2) Thermal power,
(3) Heat fluxes, average and maximum,
(4) Reactor coolant system pressure,
(5) Minimum DNBR or CPR, as applicable,
(6) Core and recirculation loop coolant flow rates (BWRs),
(7) Coolant conditions -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 -maximum fuel centerline temperature, maximum clad temperature, or maximum fuel enthalpy,
(9) Reactor coolant inventory -total inventory and coolant level in various locations in the reactor coolant system,
(10) Secondary (power conversion) system parameters -steam flow rate, steam pressure and temperature, feedwater flow rate, feedwater temperature, steam generator inventory, and
(11) ECCS flow rates and pressure differentials across the core, as applicable.

In the results discussion, emphasize the margins between the predicted values of various core parameters and the values of these parameters that would represent limiting acceptable conditions.

15.6.4 Barrier performance

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.

a. Evaluation Model

Present and discuss the evaluation model used to evaluate barrier performance. Provide the same type of information as the guidance in Section 15.6.3a. 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.

Provide a table with a list of 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 the 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 the compliance with each of the conditions and limitations in the NRC safety evaluation reports approving the TRs that document the models or codes used.

b. Input Parameters and Initial Conditions

Discuss any input parameters and initial conditions of variables relevant to the evaluation of barrier performance that were not discussed in Section 15.6.3b. Present the numerical values of the input to the analyses and discuss the adequacy of the selected values.

c. Results

Presented 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.

15.6.5 Radiological consequences.

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 boundary and the low population zone as well as those locations within the exclusion 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 that are used many times throughout Chapter 15 can be summarized (or cross-referred) with the bulk of the 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.

Provide an analysis for each limiting event. Base the analyses on design basis assumptions acceptable to the NRC for purposes of determining adequacy of the plant design to meet 10 CFR Part 100 and 10 CFR 50.34 criteria. These design basis assumptions can, for the most part, be found in regulatory guides that deal with radiological releases. For instance, when calculating the radiological consequences of a loss-of-coolant accident (LOCA), it is suggested that the assumptions given in Regulatory Guide (RG) 1.183 (as applicable to the plant design) be used. Refer to this analysis as the "design basis analysis."

There may be instances in which the applicant will not agree with the conservative margins inherent in the design basis approach approved by staff or the applicant may desire to provide a "realistic analysis" for comparison purposes. If this is the case, state the assumptions that are adequately conservative. However, 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 a presentation of assumptions that are more likely to be obtained than those used for purposes of design.

Present the parameters and assumptions used for these analyses, as well as the results, in tabular form. Table 15-3 provides a representative list of these items, although it is not meant to be all-encompassing with regard to the design basis accidents analyzed or the parameters and assumptions that may be included in the table. Table 15-4 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. If, however, a particular assumption cannot be simply or clearly stated in the table, reference a section or an appendix that adequately discusses the information.

Use judgment in eliminating unnecessary parameters from the summary table or in adding parameters of significance that do not appear in Table 15-3 or 15-4. Include a summary table with one column for the assumptions used in the design basis analysis and one column for the assumptions used in the realistic analysis.

Include as an appendix a diagram of the dose computation model labeled "Containment Leakage Dose Model" as well as an explanation of the model. The purpose of the appendix is to clearly illustrate the containment modeling, the leakage or transport of radioactivity from one compartment to another or to the environment, and the presence of engineered safety features (ESF) such as filters or sprays that are called on to mitigate the consequences of the LOCA. Use easily identifiable symbols in the diagram such as squares to represent the containment or various portions of it, 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, 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 readily available to the NRC staff. Include the following information:

a. A description of the evaluation model used, including any simplifications or approximations introduced to perform the analyses.

b. An identification and description of any digital computer program used in the analysis. The detailed description of the evaluation models are preferably included by reference with only summaries provided in the application.

c 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.

d. The considerations of uncertainties in calculational methods, equipment performance, instrumentation response characteristics, or other indeterminate effects taken into account in the evaluation of the results.

e. A discussion of the extent of system interdependency (containment system and other engineered safety features) contributing directly or indirectly to controlling or limiting leakages from the containment system or other sources (e.g., from spent fuel handling areas), such as the contribution of:

    (1) containment water spray systems,
    (2) containment air cooling systems,
    (3) air purification and cleanup systems,

(4) reactor core spray or safety injection systems,
(5) postaccident heat removal systems, and
(6) 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 boundary. Provide the doses for the course of the accident at the closest boundary of the low population zone (LPZ) and, when significant, the doses to the control room operators during the course of the accident. Present other organ doses for those cases where a release of solid fission products or transuranic elements are postulated to be released to the containment atmosphere.

f. Justification for 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 TEDE for the exclusion boundary. Provide the TEDE for the duration of the accident at the closest boundary of the low population zone (LPZ) and, when significant, the TEDE to the control room operators for the duration of the accident.

TABLE 15-1: Representative Initiating Events to Be Analyzed

15.0.1 Radiological Consequences Analyses. (The applicant may choose to group all DBA radiological consequences analyses under a single section, or the radiological consequences of each accident may be discussed under the following applicable sections. SRP 15.0.1 may be used until a new SRP Chapter 15 section (15.0.3) is written for new reactors.)

15.1 Increase in Heat Removal by the Secondary System

15.1.1 Decrease in feedwater temperature as a result of feedwater system malfunctions
15.1.2 Increase in feedwater flow as a result feedwater system malfunctions
15.1.3 Increase in steam flow as a result of steam pressure regulator malfunction
15.1.4 Inadvertent opening of a steam generator relief or safety valve steam bypass misoperation (multiple turbine dump valves)
15.1.5 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.

15.2. Decrease in Heat Removal by -the Secondary System

15.2.1 Loss of external load that results in decreasing steam flow
15.2.2 Turbine trip (stop valve closure)
15.2.3 Loss of condenser vacuum
15.2.4 Inadvertent closure of main steam isolation valves (BWR)
15.2.5 Steam pressure regulator failure (closed)
15.2.6 Loss of non-emergency A.C. power to the station auxiliaries
15.2.7 Loss of normal feedwater flow
15.2.8 Feedwater system piping breaks inside and outside containment

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

TABLE 15-1: Representative Initiating Events to Be Analyzed (Continued)

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 mis-operation (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 the 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 15.0.1 or new SRP section 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 15.0.1 or new SRP section 15.0.3)
15.5.	<u>Increase in Reactor Coolant Inventory</u>
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.	<u>Decrease in Reactor Coolant Inventory</u>
15.6.1	Inadvertent opening of a pressurizer safety or relief valve in a PWR or a safety or relief valve in a BWR
15.6.3	Radiological consequences of the steam generator tube failure (May not be necessary if discussed above under 15.0.1 or new SRP section 15.0.3)
15.6.4	Radiological consequences of main steam line failure outside the containment (BWR) (May not be necessary if discussed above under 15.0.1 or new SRP section 15.0.3)
15.6.5	LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary, including steam line breaks inside of 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 15.0.1 or new SRP section 15.0.3)
15.6.5B	Radiological consequences of a design basis LOCA: leakage from engineered safety feature components outside containment (May not be necessary if discussed above under 15.0.1 or new SRP section 15.0.3)
15.6.5D	Radiological consequences of a design basis LOCA: leakage from main steam isolation valve leakage control system (BWR) (May not be necessary if discussed above under 15.0.1 or new SRP section 15.0.3)

TABLE 15-1: Representative Initiating Events to Be Analyzed (Continued)

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	<u>Radioactive Release from a Subsystem or Component</u>
15.7.3	Postulated radioactive releases due to liquid tank failures
15.7.4	Radiological consequences of a fuel handling accident (May not be necessary if discussed above under 15.0.1 or new SRP section 15.0.3)
15.7.5	Spent fuel cask drop accidents
15.8	<u>Anticipated Transients Without Scram</u>
15.8.1	Loss of feedwater
15.8.2	Loss of electrical load
15.8.3	Turbine trip
15.8.4	Loss of condenser vacuum
15.8.5	Loss of off-site power
15.8.6	Closure of main steam line isolation valves
15.8.7	Inadvertent control rod withdrawal

TABLE 15-2: 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 DNBR or 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
CVCS Flow and Boron Concentration (if these vary during the course of the transient or accident being analyzed)
Control Rod Worth, Differential, and Total
SLES Flow and Boron Concentration
ECCS Flow

Table 15-3: 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:
      (1) the maximum allowable equilibrium iodine concentration and
      (2) the 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 (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. x/Qs 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)

Table 15-4: 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
      (1) Holdup time prior to purge initiation (assuming recombiners are inoperative)
      (2) Iodine reduction factor
      (3) x/Q values at appropriate time of release
      (4) Purge rates for at least 30 days after initiation of purge
      (5) LOCA plus purge dose at LPZ

   b. Equipment Leakage Contribution to LOCA Dose
      (1) Iodine concentration in sump water after LOCA
      (2) Maximum operational leak rate through pump seals, flanges, valves, etc.
      (3) Maximum leakage assuming failure and subsequent isolation of a component seal
      (4) Total leakage quantities for (2) and (3)
      (5) Temperature of sump water vs time
      (6) Time intervals for automatic and operator action
      (7) Leak paths from point of seal or valve leakage to the environment
      (8) Iodine partition factor for sump water vs temperature of water
      (9) Charcoal adsorber efficiency assumed for iodine removal

   c. Main Steam Line Isolation Valve Leakage Contribution to LOCA Dose (BWR)
      (1) Time of leakage control system actuation, if applicable
      (2) Fraction of isolation valve leakage from each release point
      (3) Flow rates vs time for each release path
      (4) Location of each release point
      (5) Transport time to each release point
      (6) Iodine removal constants or decontamination factors, either by the leakage control system or by deposition and plateout, as applicable

2. Main Steam Line and Steam Generator Tube Failures

   a. Characterize 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, b, c, d, and 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

Table 15-5: Summary of Initial Conditions and Computer Codes Used

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

Table 15-6: Nominal Values of Pertinent Plant Parameters used in the Accident Analyses

Provide in a tabular form the reactor trip functions, the engineered safety feature functions, and other equipment available for mitigation of each transient and accident.

Table 15-7: **Safety Analysis RPS and ESFAS Trip Setpoints and Delay Times**

Provide in a tabular form a summary of the trip setpoints, total delay times of the reactor protection system and the engineered safety feature 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.

Table 15-8: Single Failures

Provide in a table all single failures considered for determining the limiting single failure used in each transient or accident analyzed.

Table 15-9: Limiting Single Failures Assumed in Transient and Accident Analyses

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

Table 15-10: Non-Safety Related System and Equipment used for Mitigation of Transients and Accidents

Provide in a tabular form a list of non-safety related system and equipment used for mitigation of transients and accidents.