



U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

15.0 INTRODUCTION - TRANSIENT AND ACCIDENT ANALYSES

REVIEW RESPONSIBILITIES

Primary - Organizations responsible for review of transient and accident analyses for PWRs/BWRs

Secondary - None

The evaluation of the safety of a nuclear power plant requires analyses of the plant's responses to postulated equipment failures or malfunctions. Such analyses help to determine the limiting conditions for operation, limiting safety system settings, and design specifications for safety-related components and systems to protect public health and safety. These analyses are a focal point of the license amendment request (LAR), design certification (DC), and combined license (COL) reviews.

I. AREAS OF REVIEW

The specific areas of review are as follows:

1. Categorization of Transients and Accidents. The reviewer ensures that the applicant's selection and assembly of the plant transient and accident analyses represent a sufficiently broad spectrum of transients and accidents, or initiating events.

Initiating events are categorized according to expected frequency of occurrence and by type. Categorization by frequency of occurrence provides a basis for selection of the applicable analysis acceptance criteria for each initiating event. Categorization of

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USNRC STANDARD REVIEW PLAN

This Standard Review Plan, NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The Standard Review Plan is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide an acceptable method of complying with the NRC regulations.

The standard review plan sections are numbered in accordance with corresponding sections in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of Regulatory Guide 1.70 have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) are based on Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

These documents are made available to the public as part of the NRC's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by email to NRR_SRP@nrc.gov.

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initiating events by type provides a basis for comparison between events, which makes it possible to identify and evaluate the limiting cases (i.e., the cases that can challenge the analysis acceptance criteria).

- A. Categorization According to Frequency of Occurrence. Each initiating event is categorized as either an anticipated operational occurrence (AOO) or as a postulated accident.

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

The SRP uses the term AOOs to refer to the events that are categorized in Regulatory Guide 1.70 (Ref. 2) and in Regulatory Guide 1.206 (Ref. 3) as incidents of moderate frequency (i.e., events that are expected to occur several times during the plant's lifetime) and infrequent events (i.e., events that may occur during the lifetime of the plant).

Incidents of moderate frequency and infrequent events are also known as Condition II and Condition III events, respectively, in the commonly used, oft-cited but withdrawn American Nuclear Society (ANS) standards (Refs. 4 and 5). The reviewer will continue to evaluate applications, according to the categorizations and acceptance criteria of References 4 and 5, for licensees that have these categorizations in their licensing bases. For these licensees, the reviewer will also examine applications according to the categorizations and acceptance criteria of this SRP section for initiating events that are designated as AOOs, as defined here. The reviewer will evaluate new applications (i.e., those pertaining to plants that are not yet constructed) according to the categorizations and acceptance criteria of this SRP section.

The following are some examples of AOOs in pressurized-water reactor (PWR) and boiling-water reactor (BWR) designs:

- Inadvertent control rod or rod group withdrawal (PWR and BWR)
- Loss or interruption of core coolant flow, excluding reactor coolant pump locked rotor (PWR)
- Inadvertent moderator cooldown (PWR and BWR)
- Inadvertent chemical shim dilution (PWR)
- Depressurization by spurious operation of an active element, such as a relief valve (PWR and BWR)
- Blowdown of reactor coolant through a safety valve (PWR and BWR)
- Loss of normal feedwater (PWR and BWR)
- Loss of condenser cooling (PWR and BWR)

- Steam generator tube leaks (PWR)
- Reactor-turbine load mismatch, including loss of load and turbine trip (PWR and BWR)
- Control rod drop (inadvertent addition of absorber) (PWR)
- Single error of an operator (PWR and BWR)
- Single failure of a control component (PWR and BWR)
- Single failure in the electrical system (PWR and BWR)
- Minor reactor coolant system (RCS) leak or loss of reactor coolant such as from a small ruptured pipe or from a crack in a large pipe (PWR and BWR)
- Minor secondary system break (PWR)
- Loss of offsite power (PWR and BWR)
- Operation with a fuel assembly in an improper position (PWR and BWR)
- Inadvertent blowdown of RCS (BWR)
- Loss of feedwater heating (PWR and BWR)
- Trip of any or all recirculation pumps (BWR)
- Inadvertent pump start in a hot recirculation loop (BWR)
- Condenser tube leak (BWR)
- Startup of an idle recirculation pump in a cold loop (BWR)
- Reactor overpressure with delayed scram (BWR)

The individual event sections of the SRP address specific AOOs and their appropriate variations (e.g., design-specific variations).

Anticipated transients without scram (ATWSs) are AOOs in which a reactor scram is demanded but fails to occur because of a common-mode failure in the reactor scram system. ATWS events, therefore, are AOOs that postulate complete failure of the required (single-failure proof) protection system. As such, they are beyond the design basis, and consequently, ATWS events are addressed separately (see SRP Section 15.8 and Ref. 6).

Postulated accidents are unanticipated occurrences (i.e., they are postulated but not expected to occur during the life of the nuclear power plant).

Postulated accidents are also known as Condition IV events in the unofficial ANS standards (Refs. 4 and 5).

The following are some examples of postulated accidents in PWRs and BWRs of current designs:

- Major rupture of a pipe containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant pressure boundary (PWR and BWR)
- Ejection of a control rod assembly (PWR)
- Control rod drop accident (BWR)
- Major secondary system pipe rupture up to and including double-ended rupture (PWR and BWR)
- Single reactor coolant pump locked rotor (PWR)
- Seizure of one recirculation pump (BWR)

The sections of the SRP dealing with the individual events address specific postulated accidents and appropriate variations (e.g., design-specific variations).

- B. Categorization According to Type. AOOs and postulated accidents are also categorized according to type. The type of AOO or postulated accident is defined by its effect on the plant. For example, one type of AOO or postulated accident will cause the RCS to pressurize and possibly jeopardize RCS integrity. Another type will cause the RCS to depressurize and possibly jeopardize fuel cladding integrity. It is useful to categorize and organize analyses of AOOs and postulated accidents according to type, so that analysts can compare them on common bases, effects, and safety limits. Such comparisons can help to identify limiting events and cases for detailed examination and eliminate nonlimiting cases from further consideration.

AOOs and postulated accidents can be grouped into the following seven types (Ref. 2):

- i. Increase in heat removal by the secondary system
- ii. Decrease in heat removal by the secondary system
- iii. Decrease in RCS flow rate
- iv. Reactivity and power distribution anomalies
- v. Increase in reactor coolant inventory
- vi. Decrease in reactor coolant inventory
- vii. Radioactive release from a subsystem or component

The review of AOOs and postulated accident analyses, within a type, can (and should) encompass a variety of cases, each designed to produce effects or results that challenge designated safety limits. For example, one case study of the turbine trip event, an AOO that causes a decrease in heat removal by the secondary system, can be designed to yield a high peak RCS pressure, and

another case study of the same AOO can be designed to yield a low, minimum thermal margin. The former case tests the safety limit for RCS pressure boundary integrity, while the latter case tests the safety limit that protects fuel cladding integrity.

The reviewer considers the possible case variations of AOOs and postulated accidents presented to verify that the licensee has identified the limiting cases. The reviewer evaluates licensees' claims that individual AOOs and postulated accidents are limiting or nonlimiting, or bounded by other AOOs and postulated accidents, with particular attention to the bases used for comparison. Comparison of AOOs to other AOOs within a type, for example, is easily justified. Comparison of AOOs of one type to postulated accidents of another type requires closer scrutiny and more justification from the licensee.

2. Analysis Acceptance Criteria. If the risk of an event is defined as the product of the event's frequency of occurrence and its consequences, then the design of the plant should be such that all the AOOs and postulated accidents produce about the same level of risk (i.e., the risk is approximately constant across the spectrum of AOOs and postulated accidents). This is reflected in the general design criteria (GDC), which generally prohibit relatively frequent events (AOOs) from resulting in serious consequences, but allow the relatively rare events (postulated accidents) to produce more severe consequences.

The reviewer will consider the results of licensees' analyses and evaluations of individual initiating events to ascertain whether the licensee has satisfied the applicable analysis acceptance criteria for each of the events. The licensee may propose the use of alternate acceptance criteria appropriate to the particular plant design and operation (e.g., for new reactor design applications). In such cases, the reviewer will consider the alternate criteria and determine whether they are equivalent, in function and consequences, to the current criteria (see below).

- A. Analysis Acceptance Criteria for AOOs. The following are the specific criteria necessary to meet the requirements of GDC for AOOs:
 - i. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 7).
 - ii. Fuel cladding integrity Revisions should be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and that the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs.
 - iii. The reviewer applies a third criterion, to ensure that there is no possibility of initiating a postulated accident with the frequency of occurrence of an AOO. Some of the questions that licensees must answer to justify making plant modifications without advance review (see 10 CFR 50.59) by the NRC staff reflect this concern.

An AOO should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.

For licensees that have the categorizations of References 4 or 5 (i.e., ANS Condition II, III, and IV events) in their licensing bases, the reviewer will apply the following acceptance criteria:

- (1) Condition II events
 - (a) Same as Criterion i (above), for AOOs.
 - (b) Same as Criterion ii (above), for AOOs.
 - (c) By itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV category without other incidents occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.

- (2) Condition III events
 - (a) No more than a small fraction of the fuel elements in the reactor are damaged, although sufficient fuel element damage might occur to preclude resumption of operation for a considerable outage time.
 - (b) For PWRs, the release of radioactive material may exceed guidelines of 10 CFR Part 20, but Revisions should not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius.

For BWRs, the offsite release of radioactive material is limited to a small fraction of the guidelines of 10 CFR Part 100, which may be the result of the failure of a small fraction of the fuel elements in the reactor.
 - (c) A Condition III incident Revision should not, by itself, generate a Condition IV fault or result in a consequential loss of function of the RCS or reactor containment barriers.

- (3) Condition IV events

ANS Condition IV events are postulated accidents. The reviewer will apply the acceptance criteria for postulated accidents (below) to evaluate Condition IV events.

- B. Analysis Acceptance Criteria for Postulated Accidents. A list of the basic criteria necessary to meet the requirements of GDC for postulated accidents appears below. Individual sections of the SRP may specify additional criteria pertaining

to certain postulated accidents.

- i. Pressure in the RCS and main steam system should be maintained below acceptable design limits, considering potential brittle as well as ductile failures (Ref. 7).
- ii. Fuel cladding integrity will be maintained if the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs. If the minimum DNBR or MCPR does not meet these limits, then the fuel is assumed to have failed.
- iii. The release of radioactive material Revisions should not result in offsite doses in excess of the guidelines of 10 CFR Part 100.
- iv. A postulated accident might cause sufficient damage to preclude resumption of plant operation.
- v. A postulated accident Revisions should not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.

For loss-of-coolant accidents (LOCAs), the following analysis acceptance criteria of 10 CFR 50.46 also apply:

- i. The calculated maximum fuel element cladding temperature Revisions should not exceed 2200 °F.
- ii. The calculated total oxidation of the cladding Revisions should nowhere exceed 0.17 times the total cladding thickness before oxidation.
- iii. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam Revisions should not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
- iv. Calculated changes in core geometry Revisions should be such that the core remains amenable to cooling.
- v. After any calculated successful initial operation of the emergency core cooling system (ECCS), the calculated core temperature Revision should be maintained at an acceptably low value and decay heat Revisions should be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

3. Plant Characteristics Considered in the Safety Evaluation. The reviewer ensures that the application contains the key 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). The reviewer checks that the range of values for plant parameters is

representative of fuel exposure or core reload, and that the range is sufficiently broad to cover the predicted fuel cycle ranges, to the extent practicable, based on the fuel design and acceptable analytical methodology at the time of the LAR, DC, or COL application. The reviewer also ensures that the application specifies the permitted fluctuations and uncertainties associated with reactor system parameters and assumes the appropriate conditions, within the operating band, as initial conditions for transient analysis.

4. Assumed Protection System Actions. The reviewer ensures that the application lists the settings of all the protection system functions that are used (i.e., credited) in the safety evaluation. Typical protection system functions include reactor trips, isolation valve closures, and ECCS initiation. The reviewer also ascertains that the application lists the expected limiting delay time for each protection system function and describes the acceptable methodology for determining uncertainties (from the combined effects of calibration error, drift, instrumentation error, and other factors) to be included in the establishment of the trip setpoints and allowable values specified in the plant technical specifications.
5. Evaluation of Individual Initiating Events. The reviewer ensures that the application includes an evaluation of each initiating event, using the format in Subsection I.6 of this SRP section. For initiating events that are determined to be not limiting, the reviewer may evaluate qualitative justifications and conduct comparisons with the corresponding, more limiting initiating events.
6. Event Evaluation
 - A. Identification of Causes and Frequency Classification. For each initiating event evaluated, the reviewer ensures that the application includes a description of the occurrences that can lead to the event and a categorization of the event as either an AOO or postulated accident. The reviewer also checks for clear definitions of the analysis acceptance criteria appropriate to the specific nature of the initiating event, as well as the event's categorization.
 - B. Sequence of Events and Systems Operation. The reviewer verifies that the application addresses the following considerations for each initiating event:
 - i. Step-by-step sequence of events, from event initiation to the final stabilized condition (i.e., identification on a time scale of each significant occurrence, including flux monitor trips, insertion of control rods, attainment of primary coolant safety valve set points, opening and closing of safety valves, generation of containment isolation signals, and containment isolation) and identification of all operator actions credited in the transient and accident analyses for consequence mitigation
 - ii. Extent to which normally operating plant instrumentation and controls are assumed to function
 - iii. Extent to which plant and reactor protection systems are required to function
 - iv. Credit taken for the functioning of normally operating plant systems

- v. Credited operation of engineered safety systems
- vi. Assurance of consistency between the safety analyses and the emergency response guidelines/emergency procedure guidelines or emergency operating procedures with respect to the operator response (including action time) and available instrumentation

The reviewer verifies that the applicant has specified only safety-related systems or components for use in mitigating transient or accident conditions. The reviewer may consider the licensee's technical justifications for the operation of nonsafety-related systems or components (e.g., when they are used as backup protection and when they are not disabled, except by a detectable, random, and independent failure).

The reviewer ascertains that the applicant has evaluated the effects of single active failures (Ref. 8) and operator errors and that the licensee's application contains sufficient detail to permit independent evaluation of the adequacy of systems, as they relate to the subject events.

C. Core, System, and Barrier Performance

- i. Evaluation Model. The reviewer ensures that the applicant has discussed the evaluation model used and any simplifications or approximations introduced to perform the analyses and identified digital computer codes used in the analysis. If the analysis uses more than one computer code, the applicant should describe the method used to connect the codes. The reviewer verifies that the applicant has discussed the important output of the codes under "results" with emphasis on the input data and the extent or range of variables investigated and that the applicant has included detailed descriptions of evaluation models and digital computer codes or listings by referencing documents that are available to the NRC.

The reviewer ensures that the applicant has provided a table listing the titles of topical reports (TRs) that describe models or computer codes used in transient and accident analyses and listed the associated NRC safety evaluation reports approving those TRs. The reviewer checks that implementations of NRC-approved models or codes are within the applicable ranges and conditions and that the applicant has demonstrated compliance with each of the conditions and limitations imposed by the NRC staff in its safety evaluation reports that approve the TRs.

- ii. Input Parameters and Initial Conditions. The reviewer verifies that the applicant has (1) identified the major input parameters and initial conditions used in the analyses; (2) included the initial values of other variables and parameters in the application if they are used in the analyses of the particular event under study; (3) ensured that the parameters and initial conditions used in the analyses are suitably conservative; and (4) discussed the bases (including the degree of conservatism) used to select the numerical values of the input parameters.

iii. Results. The reviewer ensures that the applicant has presented 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:

- Neutron power
- Thermal power
- Heat fluxes, average and maximum
- RCS pressure
- DNBR or CPR, as applicable
- Core and recirculation loop coolant flow rates for BWRs
- Coolant conditions, including inlet temperature, core average temperature (for PWRs), core average steam volume fraction (for BWRs), average exit and hot channel exit temperatures, and steam volume fractions
- Temperatures, including maximum fuel centerline temperature, maximum clad temperature, or maximum fuel enthalpy
- Reactor coolant inventory, including total inventory and coolant level in various locations in the RCS
- Secondary (power conversion) system parameters, including steam flow rate, steam pressure and temperature, feedwater flow rate, feedwater temperature, and steam generator inventory
- ECCS flow rates and pressure differentials across the core, as applicable
- Containment pressure
- Relief and/or safety valve flow rate
- Flow rate from the RCS to the containment system, if applicable
- Pressurizer water volume (for PWRs)

In addition, the discussion of the results 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.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 10 CFR Part 20, "Standards for Protection Against Radiation"
2. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (especially 10 CFR 50.46 and Appendix A)
3. 10 CFR Part 100, "Reactor Site Criteria"
4. 10 CFR Part 52, "Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants"

The following GDC from Appendix A to 10 CFR Part 50 are relevant to SRP Section 15:

1. GDC 2, as it relates to the seismic design of structures, systems, and components (SSCs) whose failure could cause an unacceptable reduction in the capability of the residual heat removal system.
2. GDC 4, as it relates to the requirement that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accident conditions, including such effects as pipe whip and jet impingement.
3. GDC 5, as it relates to the requirement that any sharing among nuclear power units of SSCs important to safety will not significantly impair their safety function.
4. GDC 10, as it relates to the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations including AOOs.
5. GDC 13, as it relates to instrumentation and controls provided to monitor variables over anticipated ranges for normal operations, for AOOs, and for accident conditions.
6. GDC 15, as it relates to the RCS and its associated auxiliaries being designed with sufficient margin to ensure that the pressure boundary are not to be exceeded during normal operations, including AOOs.
7. GDC 17, as it relates to the requirement that an onsite and offsite electric power system be provided to permit the functioning of SSCs important to safety. The safety function for each system (assuming the other system is not working)Revision should be to provide sufficient capacity and capability to ensure that the acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded during an AOO and that core cooling, containment integrity, and other vital functions are maintained in the event of an accident.

8. GDC 19, as it relates to the requirement that a control room be provided from which personnel can operate the nuclear power unit during both normal operating and accident conditions, including a LOCA.
9. GDC 20, as it relates to the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that the plant does not exceed specified acceptable fuel design limits during any condition of normal operation, including AOOs.
10. GDC 25, as it relates to the requirement that the reactor protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods.
11. GDC 26, as it relates to the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded even during AOOs. This is accomplished by ensuring that the applicant has allowed an appropriate margin for malfunctions such as stuck rods.
12. GDC 27 and 28, as they relate to the RCS being designed with an appropriate margin to ensure that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained.
13. GDC 29, as it relates to the design of the protection and reactivity control systems and their performance (i.e., to accomplish their intended safety functions) during AOOs.
14. GDC 31, as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.
15. GDC 34, as it relates to the capability to transfer decay heat and other residual heat from the reactor so that fuel and pressure boundary design limits are not exceeded.
16. GDC 35, as it relates to the RCS and associated auxiliaries being designed to provide abundant emergency core cooling.
17. GDC 55, as it relates to the isolation requirements of small-diameter lines connected to the primary system.
18. GDC 60, as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment.
19. GDC 61, as it relates to the requirement that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions.

SRP Acceptance Criteria

Subsection I.2 of this SRP section discusses general acceptance criteria, and SRP Chapter 15 subsections discuss specific acceptance criteria for transients or accidents. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an

applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

III. REVIEW PROCEDURES

To evaluate the LAR, DC, or COL application, the reviewer verifies that the applicant has performed the applicable transient and accident analyses needed to demonstrate conformance to the regulations.

SRP Chapter 15 subsections discuss specific review procedures for transients or accidents.

For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

IV. EVALUATION FINDINGS

SRP Chapter 15 subsections discuss the statements and conclusions of evaluation findings for transients or accidents.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

The referenced regulatory guides contain implementation schedules for conformance to parts of the method discussed here.

VI. DEFINITIONS

| Term | Definition |
|--|--|
| anticipated operational occurrences (AOOs) | <p>Conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power (Ref. 1).</p> <p>AOOs are also known as Condition II and III events (Refs. 4 and 5).</p> |
| anticipated transient without scram (ATWS) | <p>AOO followed by the failure of the reactor trip portion of the protection system specified in GDC 20 (Ref. 1), because of common-mode failure.</p> |
| common-mode failure | <p>The result of an event which, because of dependencies, causes a coincidence of failure states of components in two or more separate channels of a redundancy system, leading to the failure of the defined system to perform its intended function.</p> |
| critical power ratio (CPR) | <p>That power in the assembly that will cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.</p> |
| departure from nucleate boiling (DNB) | <p>The DNB acceptance criterion for an AOO is met when there is a 95 percent probability at a 95 percent confidence level (the 95/95 DNB criterion) that DNB will not occur, and the fuel centerline temperature stays below the melting temperature.</p> |
| design basis | <p>Information that identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design.</p> <p>These values may be (1) restraints derived from generally accepted state of the art practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.</p> |
| design-basis accidents | <p>Postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components.</p> |

| Term | Definition |
|--------------------------------------|---|
| design-basis events | Conditions of normal operation, including AOOs, design-basis accidents, external events, and natural phenomena, for which the plant must be designed to ensure functions of safety-related electric equipment that ensures the integrity of the reactor coolant pressure boundary; the capability to shut down the reactor and maintain it in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures. |
| general design criteria (GDC) | Reference 1 lists the GDC. Those that mention RevisionThe GDC that mention AOOs are 10, 13, 15, 17, 20, 26, 29, 60, and 64. GDC that mention RevisionThe GDC that mention postulated accidents are 4, 16, 17, 22, 27, 28, 31, 41, 51, 61, and 64. |
| loss-of-coolant accident (LOCA) | A postulated accident that results in the loss of reactor coolant at a rate in excess of the replacement capability of the reactor coolant makeup system. |
| M CPR safety limit | This limit ensures that during normal operation and during AOOs, at least 99.9 percent of the fuel rods in the core do not experience transition boiling. |
| minimum critical power ratio (M CPR) | The smallest CPR that exists in the core for each class of fuel. |
| overpressurization | The condition that occurs when pressure exceeds the design pressure of the component of interest by more than 10 percent, in accordance with the ASME Code (Ref. 7). |
| postulated accidents | Unanticipated conditions of operation (i.e., not expected to occur during the life of the nuclear power unit). Postulated accidents are also known as Condition IV events (Refs. 4 and 5). |
| single failure | An occurrence that results in a component's loss of capability to perform its intended safety functions (Ref. 1). |

VII. REFERENCES

1. Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Plants."
2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."
3. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."
4. ANS 51.1, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants" (replaces ANSI N18.2), 1983 Revision(withdrawn in 1998).

5. ANSI/ANS-52.1-1983, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants" Revision(withdrawn in 1998).
6. 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water Cooled Nuclear Power Plants."
7. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.
8. SECY-77-439, "Single-Failure Criterion," August 1977 (ADAMS Accession No. ML060260236).
9. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."
10. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

PUBLIC PROTECTION NOTIFICATION

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SRP Section 15.0

Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in (Draft) Revision 3, dated June 1996 of this SRP. See ADAMS accession number ML052070675.

In addition this SRP section was administratively updated in accordance with NRR Office Instruction, LIC-200, Revision 1, "Standard Review Plan (SRP) Process." The revision also adds standard paragraphs to extend application of the updated SRP section to prospective submittals by applicants pursuant to 10 CFR Part 52.

The technical changes are incorporated in Revision 3 dated [Month] 2007:

RevisionChange 1.

Chapter 15 events are categorized as either anticipated operational occurrences (AOOs) or postulated accidents. This categorization of events is to be used in new applications, and may be used by current licensees, if they so choose. Current licensees also have the option of continuing to use the RG 1.70 categorization (events of moderate frequency or Condition II, infrequent events or Condition III, and limiting faults or Condition IV).

Anticipated Transients Without Scram (ATWS) are considered separately (see SRP Section 15.8).

Acceptance criteria are specified, and examples of events are listed for each of the event categories.

Rationale:

Currently, there are three event categories: incidents of moderate frequency (Condition II), infrequent incidents (Condition III), and limiting faults (Condition IV), are defined in RG 1.70 and in two standards that were issued by the American Nuclear Society (ANS). Both standards: ANS 51.1-1983, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, and ANSI/ANS-52.1-1983, Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants, were withdrawn in 1998.

The number of event categories is reduced from three to two. The two event categories are defined in 10 CFR 50, Appendix A, and in RG 1.206. AOOs are defined as incidents that are expected to occur one or more times during the life of the nuclear power unit. Postulated accidents are defined as incidents that are not expected to occur; but are postulated because of the potential for the release of significant amounts of radioactive material.

The table summarizes and compares the categorization systems.

| ANS standards | RG 1.70 | 10 CFR 50 | This SRP version |
|--|---|--|--|
| Incidents of moderate frequency (Condition II) | Incidents of moderate frequency (expected to occur several times during the plant lifetime). | Incidents that are expected to occur one or more times during the life of the nuclear power unit | AOOs |
| Infrequent Events (Condition III) | Infrequent events (may occur during the lifetime of the plant). | Incidents that are expected to occur one or more times during the life of the nuclear power unit | AOOs |
| Limiting faults (Condition IV) | Limiting faults (not expected to occur but postulated because of the potential for the release of significant amounts of radioactive material). | Postulated accidents (not expected to occur but postulated because of the potential for the release of significant amounts of radioactive material). | Postulated accidents (not expected to occur but postulated because of the potential for the release of significant amounts of radioactive material). |
| ATWS | special case, regulated by 10 CFR 50.62 | | |

Since RG 1.70, the format for Safety Analysis Reports (SARs), distinguishes between incidents of moderate frequency, infrequent incidents, and limiting faults, and licensees have written their SARs according to this format, the NRC staff will continue to review license amendment requests that are based upon the event categorization and acceptance criteria of RG 1.70 (i.e., consistent with licensees' current licensing bases). The NRC staff will also review license amendment requests according to the event categorization and acceptance criteria of this SRP version, if the applicants so choose. New license applications will be reviewed according to the event categorization and acceptance criteria of this SRP version.

Acceptance criteria and event categories are linked by the principle of constant risk, i.e., frequent events must not result in serious consequences, whereas rare events may produce relatively severe consequences. Therefore, acceptance criteria are specified, in SRP Section 15.0, together with the event categories, and examples of events for each category.

This change bases the acceptance criteria of the SRP events on 10 CFR Part 50, Appendix A; not on two ANS standards that were withdrawn in 1998.

Change 2.

One of the acceptance criteria for Condition II events requires that a Condition II event be prevented from developing into a more serious Condition III or IV event, without the occurrence of other, independent faults. There is a similar criterion for Condition III events, that prohibits Condition III events from becoming Condition IV events. These criteria are retained for plants with licensing bases that allude to RG 1.70, and ANS 51.1 (for PWRs) or ANSI/ANS-52.1-1983 (for BWRs). A reference to RIS 2005-29 is added, for further information.

For new applications and for licensees that wish to adopt the event categorization scheme of SRP Section 15.0, this criterion is updated, to require that AOOs be prevented from developing into postulated accidents, without the occurrence of other, independent faults.

Rationale:

If the risk of an event is defined as the product of the event's frequency of occurrence and its consequences, then the plant should be designed such that all events, AOOs, and postulated accidents alike, produce about the same level of risk. This is reflected in the GDC, which generally prohibit relatively frequent events (i.e., the AOOs) from resulting in serious consequences; but allow the relatively rare events (i.e., the postulated accidents) to produce more severe consequences.

This acceptance criterion, which is found in the aforementioned ANS standards, prevents the initiation of a serious event (e.g., a postulated accident) with the (higher) frequency of occurrence of a less serious event (e.g., an AOO). In other words, this acceptance criterion establishes a barrier that blocks the migration of an event into a more serious category.

This concern is reflected in some of the questions licensees must answer to justify making plant modifications without prior NRC staff review (10 CFR 50.59). Licensees, all of whom are presumed to be operating plants that are designed according to the principle of constant risk, who wish to make changes to their plants, without prior NRC review, are required to satisfactorily answer the questions of 10 CFR 50.59. These questions pertain to the effects of implementing licensees' changes, particularly with respect to the frequencies of occurrence and consequences of licensing basis events. If an anticipated change can alter the frequency of occurrence and/or the consequences of one or more licensing basis events, then it must be reviewed by the NRC staff.

This acceptance criterion is retained, since it is important to plant design and analysis, and it is addressed in the regulations (10 CFR 50.59, promulgated in 1999). The revised portions of the SRP Section deal with the categorization of licensing basis events according to frequency of occurrence, and according to the acceptance criteria for each of the event categories.

Change 3.

Delete the requirement that an event's category shifts to the next, more serious category, when it is combined with the coincident occurrence of a single failure.

Rationale:

This requirement is not the same as the acceptance criterion that prohibits an event from developing into a more serious event, without the occurrence of other, independent faults, since the single failure is considered to be an independent fault.

An AOO, when combined with a single failure (e.g., another AOO), is no longer an AOO, since its probability of occurrence is significantly reduced (i.e., its probability of occurrence is the product of the probabilities of occurrence of the two AOOs). The combined event, therefore, is a postulated accident; subject to the acceptance criteria of postulated accidents.