U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

15.3.3 - 15.3.4 REACTOR COOLANT PUMP ROTOR SEIZURE AND REACTOR COOLANT PUMP SHAFT BREAK

REVIEW RESPONSIBILITIES:

Primary - Organization responsible for the review of transient and accidental analyses

Secondary - None

I. AREAS OF REVIEW

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor coolant pump (RCP) in a pressurized water reactor (PWR) or recirculation pump in a boiling water reactor (BWR). Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. This Standard Review Plan (SRP) section is intended to cover both of these accidents.

1. RCP Shaft Seizure or Shaft Break Accidents

The review includes evaluation of the applicant's postulated initial and long-term core and reactor conditions pertinent to the rotor seizure or broken shaft events; the methods of thermal and hydraulic analysis; the postulated sequence of events, including time delays prior to and after protective system actuation; the assumed reactions of reactor system components; the

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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functional and operational characteristics of the reactor protection system in terms of how it affects the sequence of events; and all operator actions required to secure and maintain the reactor in a safe condition.

The results of the applicant's analyses are reviewed to assess fuel damage and to ensure that values of pertinent system parameters are within expected ranges for the type and class of reactor under review. Fuel damage is assessed by the methods described in SRP Section 4.2. System parameters to be reviewed include the following:

- core flow and flow distribution (including hydraulic instabilities),
- channel heat flux (average and hot),
- minimum critical heat flux ratio (or minimum critical power ratio),
- departure from nucleate boiling ratio,
- fuel centerline temperature
- vessel two-phase level,
- thermal power,
- vessel pressure,
- steam line pressure (BWR),
- main steam flow (BWR), and
- feedwater flow (BWR).

The sequence of events described in the safety analysis report (SAR) is reviewed by the PWR Systems Branch for PWR applications while the BWR System Branch evaluates BWR system response. The PWR Systems Branch and the BWR Systems Branch reviews concentrate on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed by Nuclear Performance and Code Review Branch to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the Nuclear Performance and Code Review Branch reviewer initiates a generic evaluation of the new analytical model.

The values of all parameters used in a new analytical model, including the initial conditions of the core and system, are reviewed. It is the responsibility of the Nuclear Performance and Code Review Branch reviewer to ensure that the appropriate physics and fuel data have been used in any staff calculations.

The Nuclear Performance and Code Review Branch performs generic reviews of the thermal-hydraulic computer models used for this transient and also performs additional analyses related to these accidents for selected reactor types as part of its primary review responsibility for SRP Section 4.4.

2. Inspection, Test, Analysis, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the applicant's proposed information on the ITAAC associated with the systems, structures, and components (SSCs) related to this SRP section is reviewed in accordance with SRP Section 14.3, "Inspections, Tests,

Analyses, and Acceptance Criteria - Design Certification." The staff recognizes that the review of ITAAC is performed after review of the rest of this portion of the application against acceptance criteria contained in this SRP section. Furthermore, the ITAAC are reviewed to assure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

3. COL Action Items and Certification Requirements and Restrictions. COL action items may be identified in the NRC staff's final safety evaluation report (FSER) for each certified design to identify information that COL applicants must address in the application. Additionally, DCs contain requirements and restrictions (e.g., interface requirements) that COL applicants must address in the application. For COL applications referencing a DC, the review performed under this SRP section includes information provided in response to COL action items and certification requirements and restrictions pertaining to this SRP section, as identified in the FSER for the referenced certified design.

Review Interfaces

In addition, the PWR Systems Branch or the BWR Systems Branch will coordinate other branches' evaluations that interface with the overall review of the system, as follows:

1. The Instrumentation and Control Branch reviews the instrumentation and controls aspects of the sequence described in the SAR to confirm that reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis as part of its primary review responsibility for SRP Sections 7.2 through 7.5.

2. The Emergency Preparedness and Radiation Protection Branch is notified regarding the extent of the fuel failures that are predicted by the analysis. Emergency Preparedness and Radiation Protection Branch then evaluates the radiological consequences of the event.

3. The Materials and Chemical Engineering Branch reviews the fracture toughness properties of the reactor coolant pressure boundary and reactor vessel as part of its primary review responsibility for SRP Sections 5.2.3 and 5.3.1.

4. The Plant Systems Branch reviews the integrity of the reactor coolant pump seals as part of its primary review responsibility for SRP Section 9.2.2.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding review branch.

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

II. ACCEPTANCE CRITERIA

The PWR Systems Branch acceptance criteria for maintaining the ability to insert the control rods and to cool the core during a RCP rotor seizure or broken shaft event are based on meeting the relevant requirements of the following regulations:

1. General Design Criterion (GDC) 17, as it relates to providing onsite and offsite electric power systems to ensure that structures, systems, and components important to safety will function. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that design conditions of the reactor coolant pressure boundary are not exceeded and the core is cooled in the event of postulated accidents.21

2. General Design Criteria 27 (GDC 27) and 28 (GDC 28), as they relate to the reactor coolant system being designed with appropriate margin to ensure24 that the capability to cool the core is maintained.

3. General Design Criterion 31 (GDC 31), as it relates to the reactor coolant system being designed with sufficient margin to ensure that the boundary behaves in a non-brittle manner and that the probability of propagating fracture is minimized.

4. 10 CFR Part 100, as it relates to the calculated doses at the site boundary.

5. 10 CFR 52.47(a)(1)(vi), as it relates to ITAAC (for design certification) sufficient to assure that the SSCs in this area of review will operate in accordance with the certification.

6. 10 CFR 52.97(b)(1), as it relates to ITAAC (for combined licenses) sufficient to assure that the SSCs in this area of review have been constructed and will be operated in conformity with the license and the Commission's regulations.

The basic objectives of the review of the accident resulting from a rotor seizure or shaft break in a reactor coolant pump are:

1. To identify which of these accidents is the more limiting.

2. To verify that, for the accident, the plant responds in such a way that the criteria regarding fuel damage, radiological consequences, and system pressure are met.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for review described in Subsection I of this SRP section. 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.

The specific criteria necessary to meet the relevant requirements of General Design Criteria 27, 28, and 31 and 10 CFR Part 100 for the rotor seizure and shaft break event are:

1. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs based on acceptable correlations (see SRP Section 4.4). If the DNBR or CPR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. If rod internal pressure exceeds system pressure, then fuel rods may balloon shortly after entering DNB. The effect of ballooning fuel rods must be evaluated with respect to flow blockage and DNB propagation. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

3. Any release of radioactive material must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines.

4. The integrity of the reactor coolant pumps should be maintained such that loss of ac power and containment isolation will not result in pump seal damage.

5. The auxiliary feedwater system must be safety grade and, when required, automatically initiated.

6. A rotor seizure or shaft break in a reactor coolant pump should not, by itself, generate a more serious condition or result in a loss of function of the reactor coolant system or containment barriers.

7. Only safety-grade equipment should be used to mitigate the consequences of the event. Safety functions should be accomplished assuming the worst single failure of a safety system active component (see Refs. 5 and 6). For new applications, loss of offsite power should not be considered a single failure; reactor coolant pump rotor seizures and shaft breaks should be analyzed with a loss of off-site power (see item 9, below) in combination with a single active failure. (This position is based upon interpretation of GDC 17, as documented in the Final Safety Evaluation Report for the ABB-CE System 80+ design certification.)

8. The ability to achieve and maintain long-term core cooling should be verified.

9. This event should be analyzed assuming turbine trip and coincident loss of offsite power and coastdown of undamaged pumps.

The applicant's analysis should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 8 through 12 are

acceptable. The NRC staff found References 13 and 14 to be acceptable transient analysis computer codes for design analysis of the Advanced Boiling Water Reactor (ABWR). References 15 through 19 were found to be acceptable computer codes for transient analyses (i.e., except for loss-of-coolant accidents, or LOCAs) for the Combustion Engineering System 80+ final safety evaluation report staff review. In addition, NUREG-1465 contains guidance on accident source terms for light-water nuclear power plants. When conducting transient analyses, the NUREG-1465 guidance is particularly important for reviewing fractions of relevant isotopes (noble gases, iodine, cesium, and rubidium) and chemical species of iodine assumed to exist within the gap between fuel pellets and cladding. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation. There are certain assumptions regarding important parameters used to describe the initial plant conditions and postulated system failures which should be used. These are listed below:

i. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating, plus an allowance to account for power measurement uncertainties. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

ii. The local flow conditions used in the core thermal-hydraulics model should be calculated based upon an inlet flow distribution corresponding to N–1 reactor coolant pumps (initial minus faulted pump) and a conservative time-dependent flow coastdown. Note that the inlet flow distribution will change as more pumps begin to coastdown following turbine trip and coincident loss of offsite power.

iii. Conservative scram characteristics are assumed, i.e., for a PWR maximum time delay with the most reactive rod held out of the core, and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate.

iv. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

Technical Rationale

The technical rationale for application of these requirements to reviewing this SRP section is discussed in the following paragraphs:

The technical rationale for application of these acceptance criteria to reviewing analyses of transients initiated by RCP rotor seizure and shaft break is discussed in the following paragraphs:

1. GDC 17 requires that onsite and offsite electrical power systems be provided to ensure that structures, systems, and components important to safety will perform their intended function. Each power system (assuming the other system is not functioning) is to provide sufficient capacity and capability to ensure that: (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences; and (2) the core is cooled

and containment integrity and other vital functions are maintained in the event of postulated accidents. GDC 17 is applicable to SRP Section 15.3.3-15.3.4 because this section reviews the analysis of events that are classified as abnormal operating occurrences or postulated accidents, depending on the severity of the results. Meeting the requirements of GDC 17 provides assurance that the design conditions of the reactor coolant pressure boundary are not exceeded as a result of reactor coolant pump rotor seizures and shaft breaks and that the core is cooled and containment and other vital functions are maintained.

2. Compliance with GDC 27 requires that reactivity control systems be designed to have a combined capability (in conjunction with poison added by the emergency core cooling system) of reliably controlling reactivity changes, thereby ensuring that the capability for core cooling is maintained under postulated accident conditions and with appropriate margin for stuck rods.

   Compliance with GDC 28 requires that reactivity control systems be designed with appropriate limits on the amount and rate of reactivity increase, thereby ensuring that the effects of postulated reactivity accidents can neither (a) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (b) disturb the core, its support structures, or other reactor pressure vessel internals sufficiently to impair the capability to cool the core. Postulated reactivity accidents to be considered shall include rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor temperature and pressure, and the addition of cold water.

   GDC 27 and GDC 28 are applicable to this section because the reviewer evaluates two events (i.e., RCP rotor seizure and shaft break) that will result in transient conditions having the potential to affect reactor coolant temperature and pressure, which in turn could result in complex changes in core reactivity. The applicant's analyses of these transients in the SAR must demonstrate that reactivity, pressure, and temperature changes will not be severe enough to cause an unacceptable impact on the reactor coolant pressure boundary or on the capability for core cooling. The analyses must be independently reviewed by the staff in accordance with this SRP section.

   Meeting the requirements of GDC 27 and GDC 28 provides a level of assurance that a transient initiated by an RCP rotor seizure or shaft break will not result in (a) unacceptable stress on the reactor coolant pressure boundary or (b) a reduction in the capability of the core cooling or reactivity control systems to perform their design safety functions.

3. Compliance with GDC 31 requires that, under the stress of operating, maintenance, testing, and postulated accident conditions, the reactor pressure boundary shall be designed with sufficient margin to ensure that (a) the boundary behaves in a non-brittle manner and (b) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other boundary material variables under operating, maintenance, testing, and postulated accident conditions. The design will also address such issues as the uncertainties of determining material properties; the effects of irradiation on material properties; residual, steady state, and transient stresses; and the sizes of flaws.

GDC 31 is applicable to this section because the reviewer evaluates two events (RCP rotor seizure and shaft break) that could result in transient conditions having the potential for adversely affecting the reactor coolant pressure boundary. Loss of a reactor cooling pump will cause a rapid reduction in coolant flow through the core and, consequently, an increase in temperature and pressure. The amount of stress to which the reactor coolant pressure boundary is subjected depends on the severity of the transient. The severity of the transient is assessed by the applicant in the SAR and reviewed by the staff in accordance with this SRP section.

Meeting the requirements of GDC 31 provides a level of assurance that a transient initiated by an RCP rotor seizure or shaft break will not result in an unacceptable stress on the reactor coolant pressure boundary or on the ability to cool the reactor core.

4. To establish the suitability of a nuclear power plant site, 10 CFR Part 100 specifies how the exclusion area, low population zone, and population center distance should be determined. Further, radiation exposure criteria stipulated in 10 CFR Part 100 provide reference values to be used in the site suitability determination based on postulated fission product releases associated with accidental events.

   10 CFR Part 100 is applicable to this section because it specifies the methodology for calculating radiation exposures at the site boundary for postulated accidents or events such as loss of a reactor coolant pump. For transients having a moderate frequency of occurrence, any release of radioactive material must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines. A small fraction is interpreted to be less than 10 percent of the 10 CFR Part 100 reference values. For the purpose of this review, the radiological consequences of a RCP rotor seizure or shaft break must include consideration of the containment, confinement, and filtering systems. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies such as those contained in NUREG-1465.

   Meeting this requirement provides a level of assurance that, in the event of a transient initiated by a reactor coolant pump rotor seizure or shaft break, radiation exposures at the site boundary will not exceed a small fraction of the reference values specified in 10 CFR Part 100.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review specified in subsection I of this SRP section, the review procedure is identified below. These review procedures are based on the identified SRP acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the applicant’s evaluation of how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the relevant NRC requirements identified in subsection II.

The procedures below are used during the construction permit (CP), operating license (OL), and combined license (COL)reviews. During the CP review, the values of system parameters and set-points used in the analysis will be preliminary in nature and subject to change. At the OL or COL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The applicant's analyses of the rotor seizure and shaft break events are reviewed by PWR Systems Branch regarding the occurrences leading to the initiating event. The sequence of events, from initiation until a stabilized condition is reached, is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The extent to which credit is taken for the functioning of normally operating plant systems.

4. The extent to which the operation of engineered safety systems is required.

5. The extent and time at which operator actions are required.

6. That appropriate margin for malfunctions, such as stuck rods (see II.3.b), are accounted for.

If the SAR states that one of the accidents is not as limiting as the other, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the SAR of the accident that is determined to be more limiting. For the accident that is found more limiting, the reviewer confirms that the effects of the accident are determined for each mode of operation (e.g., one-, two-, three-, or four-loop) allowed by the technical specifications. Either a separate analysis should be presented or each mode of operation or the effects of each mode should be referenced to the limiting case.

For the more limiting accident, the PWR Systems Branch reviewer, with the aid of the Instrumentation and Control Branch reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the accident to acceptable levels. The PWR Systems Branch reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The Instrumentation and Control Branch review of Chapter 7 of the SAR confirms that the instrumentation and control systems design is consistent with the requirements for safety systems actions for these events.

To the extent deemed necessary, the PWR Systems Branch reviewer evaluates the effect of single active failures of safety systems and components which may alter the course of the accident. For new applications, the loss of off-site power is not considered a single active failure, but considered in addition to a single active failure as discussed in subsection II.7. This phase of the review uses the system review procedures described in the SRP sections for Chapters 5, 6, 7, and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by PWR Systems Branch to determine if these models have been previously reviewed and found acceptable by the staff. If not, PWR Systems Branch initiates a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by the PWR Systems Branch. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated.

The results of the applicant's analysis are reviewed and compared to the acceptance criteria presented in subsection II regarding the maximum pressure in the reactor coolant and main steam systems. Time-related variations of the following parameters are reviewed:

- reactor power;
- heat fluxes (average and maximum);
- fuel centerline temperature
- reactor coolant system pressure;
- minimum DNBR (PWR) or CPR (BWR);
- core and recirculation loop coolant flow rates (BWR);
- coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions);
- steam line pressure;
- containment pressure;
- pressure relief valve flow rate; and
- flow rate from the reactor coolant system to the containment system (if applicable).

The more important of these parameters (as listed in subsection I of this SRP section) are compared to those predicted for other similar plants to confirm that they are within the expected range. The percentage of fuel rods that experience failure is reviewed and Emergency Preparedness and Radiation Protection Branch is notified regarding the extent of fuel failures predicted by the analysis.

For reviews of DC and COL applications under 10 CFR Part 52, the reviewer should follow the above procedures to verify that the design set forth in the safety analysis report, and if applicable, site interface requirements meet the acceptance criteria. For DC applications, the reviewer should identify necessary COL action items. With respect to COL applications, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit, or other NRC-approved material, applications, and/or reports.

After this review, SRP Section 14.3 should be followed for the review of Tier I information for the design, including the postulated site parameters, interface criteria, and ITAAC.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and that the review supports the following kinds of statements and conclusions which should be included in the staff's safety evaluation report (SER).

The staff concludes that the consequences of postulated rotor seizure or broken shaft events meet the requirements set forth in the General Design Criteria 17, 27, 28, and 31 regarding the ability to insert control rods and to cool the core, and 10 CFR Part 100 guidelines regarding radiological dose at the site boundary. This conclusion is based upon the following:

(a) The applicant has demonstrated that the resultant fuel damage was limited such that the ability to insert control rods would be maintained, and that no loss of core cooling capability resulted. The minimum departure from nucleate boiling ratio (DNBR) or critical power ratio (CPR) experienced by any fuel rod was _____, resulting in ____% of the rods experiencing cladding perforation.

(b) The applicant met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

(c) The analyses and effects of pump rotor seizure and shaft breaks, during various modes of operation and with and without offsite power, have been reviewed.

(d) The accidents analyzed were evaluated using a mathematical model that has been previously reviewed and found acceptable by the staff.

(e) The parameters used as input to this model were reviewed and found to be suitably conservative.

(f) The radioactivity release has been evaluated using the computer code SARA and a conservative description of the plant response to the accident. A decontamination factor of _____ between the water and steam phases and a X/Q value of _____ sec/m3 has been used in our evaluation of radiological consequences. The calculated doses are presented in Table _____ . Technical specification limits on primary and secondary coolant activities will limit potential doses to a small fraction of the 10 CFR Part 100 exposure guidelines. The potential doses are within 10 CFR Part 100 exposure guidelines even if the accident should occur coincident with an iodine spike.

For DC and COL reviews, the findings will also summarize (to the extent that the review is not discussed in other SER sections), the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable, and site interface requirements and combined license action items relevant to this SRP section.

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 superceded by a later revision.

Implementation schedules for compliance with parts of the method addressed are in the referenced NUREGs except for the position of subsections II.7 and III of this SRP section on loss of offsite power and assumed single failures. This new position will be applied to new applications (for a construction permit, a manufacturing license, or DC).

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 17, "Electric Power Systems."

2. 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control Systems Capability."

3. 10 CFR Part 50, Appendix A, General Design Criterion 28, "Reactivity Limits."

4. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

5. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

6. "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from the Director, NRR, to NRR Staff" (Issue No. 1), U.S. Nuclear Regulatory Commission, NUREG-0318, November 1976.

7. "Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum from the Director, NRR, to NRR Staff" (Issue No. 22), U.S. Nuclear Regulatory Commission, NUREG-0158, December 1976.

8. F. M. Bordelon, "Calculation of Flow Coastdown after Loss of Reactor Coolant Pump," WCAP-7973, Westinghouse Electric Corporation, August 1970.

9. C. D. Morgan, H. C. Cheatwood, and J. R. Glandermans, "RADAR - Reactor Thermal and Hydraulic Analysis During Reactor Flow Coastdown," BAW-10069, Babcock and Wilcox Company, July 1973.

10. R. H. Stoudt and J. E. Busby, "CADD - Computer Applications to Direct Simulation of Transient Events on Water Reactors," BAW-10080 (nonproprietary) and BAW-10076 (proprietary), Babcock and Wilcox Company, July 1973.

11. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973.

12. R. Linford, "Analytical Methods of Transient Evaluations in the General Electric Boiling Water Reactor," NEDP-10802, General Electric Company, April 1973.

13. General Electric Co., ODYNA - One Dimensional Dynamic Model (proprietary computer software for use in ABWR transient analysis to simulate pressurization events).

14. General Electric Co., REDYA (proprietary computer software for use in ABWR transient analysis to simulate other than pressurization events).

15. CESEC-III (CENPD-107; LD-82-001). (Calculates system parameters such as core power, flow, pressure, temperature, and valve actions during a transient.)

16. TORC (CENPD-161) and CETOP (CENPD-206-P-A). (TORC is used to simulate the three-dimensional fluid conditions within the reactor core. Results from TORC include the core radial distribution of the relative channel axial flow that is used to calibrate CETOP. TORC or CETOP calculations for the DNBR use the CE-1 critical heat flux correlation.)

17. HERMITE (CENPD-188-A). (HERMITE is used to determine short-term response of the reactor core during the postulated reactor coolant pump rotor-seizure event and total loss-of-flow event.)

18. COAST (SSAR; CENPD-98). (Calculates the time-dependent reactor coolant mass flow rate in each loop during reactor coolant pump coastdown transients.)

19. STRIKIN-II (CENPD-133; CENPD-135 Supps. 2 and 4). (Calculates the cladding and fuel temperatures for an average or hot fuel rod.)

20 NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995.

21. Standard Review Plan Section 4.4, "Thermal and Hydraulic Safety."

22. Standard Review Plan Section 4.2, "Fuel System Design."

23. Standard Review Plan Section 7.2, "Reactor Trip System."

24. Standard Review Plan Section 7.3, "Engineered Safety Features System."

25. Standard Review Plan Section 7.4, "Systems Required for Safe Shutdown."

26. Standard Review Plan Section 7.5, "Safety-Related Display Instrumentation."

27. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License."

28. NUREG-0737, "Clarification of TMI Action Plan Requirements."

29. NRC Inspection Manual Chapter IMC-2504, “Construction Inspection Program - Non-ITAAC Inspections,” issued April 25, 2006.

PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the draft 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

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

SRP Section 15.3.3-15.3.4
Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in Draft Revision 3, dated April 1996, of this SRP section. See ADAMS accession number ML052070707.

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 this updated SRP section to prospective applicant submissions pursuant to 10 CFR Part 52.

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

Review Responsibilities - Reflects changes in review branches resulting from reorganization and branch consolidation. Change is reflected throughout the SRP.

I. AREAS OF REVIEW

1. Reformatted the section with new numbering system. Incorporated reference to 10 CFR Part 52 from draft revision 1 - April 1996. Incorporated generic paragraphs relating to certified designs, ESPs, and COLs.

II. ACCEPTANCE CRITERIA

1. Reformatted the section with new numbering system. Incorporated reference to 10 CFR Part 52 from draft revision 1 - April 1996. Incorporated generic paragraphs relating to certified designs, ESPs, and COLs.

2. Under "Specific Acceptance Criteria" added words to include accident scenario for ballooning under rod internal pressure (item 2)

3. Under "Specific Acceptance Criteria" added additional criteria to cover local flow conditions for core thermal-hydraulics model for loss of offsite power (LOOP) simulation. (item 9, ii)

III. REVIEW PROCEDURES

1. Reformatted the section with new numbering system. Incorporated reference to 10 CFR Part 52 from draft revision 1 - April 1996. Incorporated generic paragraphs relating to certified designs, ESPs, and COLs. to address application of Review Procedures in design certification reviews.

IV. EVALUATION FINDINGS

None

V. IMPLEMENTATION

None

VI. REFERENCES

1. Revised references as appropriate to reflect updates to the SRP section.