NUREG-0800

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

15.2.8 FEEDWATER SYSTEM PIPE BREAK INSIDE AND OUTSIDE CONTAINMENT (PWR)

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

The steam and water release from a postulated feedwater line break results in a loss of secondary coolant which may result in a reactor system cool-down (by excessive energy discharge through the break) or a reactor system heat-up (from the loss of reactor system heat sink). A major feedwater line rupture is defined as a feedwater line break large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. If the break is postulated in the feedwater line between the isolation valves and the steam generator, fluid from the steam generator also is discharged from the break. (A break upstream of the feedwater isolation valves would affect the reactor system only as a loss of feedwater. This case is covered by SRP Section 15.2.7, “Loss of Normal Feedwater Flow.”)

If a feedwater line rupture causes the water in the steam generator to be discharged through the break, the water will not be available for decay heat removal after reactor scram. The break location and size may prevent addition of any feedwater to the affected steam generator. An auxiliary feedwater system (AFWS) makes feedwater available to remove decay heat to prevent

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over-pressurization of the reactor system. In one group of design, such as AP1000, the passive RHR (PRHR) may provide a safety related means of decay heat removal.

The specific areas of review are as follow:

1. Evaluation of the applicant's postulated initial core and reactor conditions pertinent to the feedwater line break.

   The results of the analyses are reviewed for whether the values of pertinent system parameters, addressed in subsection II of this Standard Review Plan (SRP) section, are within expected ranges. The parameters of importance for these transients include:

   A. reactor coolant system (RCS) pressure,

   B. steam generator pressure,

   C. fluid temperatures,

   D. fuel and clad temperatures,

   E. break discharge flow rate,

   F. steamline and feedwater flow rates,

   G. safety and relief valve flow rates,

   H. pressurizer and steam generator water levels,

   I. mass and energy transfer within the containment (for breaks inside containment),

   J. reactor power,

   K. total core reactivity,

   L. hot and average channel heat flux, and

   M. minimum departure from nucleate boiling ratio (DNBR).

2. Methods of thermal and hydraulic analysis, the postulated sequence of events, including analyses to determine the time of reactor trip and time delays prior and subsequent to initiation of reactor protection system (RPS) actions.

   The analytical thermal/hydraulic methods are reviewed for whether the mathematical modeling and computer codes have been reviewed and accepted by the staff. If a referenced analytical method has not been reviewed, the reviewer requests an evaluation of the new analytical model. The parameter values in the analytical model, the initial conditions of the core, and all nuclear design parameters are reviewed. This review includes:

A. power level,

B. power distribution,

C. Doppler coefficients,

D. moderator temperature coefficients,

E. void coefficients,

F. reactor kinetics,

G. departure from nucleate boiling correlations, and

H. control rod worth.

3. The response of the reactor coolant and auxiliary systems, the functional and operational characteristics of RPS effects on the sequence of events, and all operator actions required to secure and maintain the reactor in a safe shutdown condition.

   The sequence of events described in the applicant's safety analysis report (SAR) is reviewed for the performance of the RPS, the engineered safety systems, and operator actions to secure and maintain the reactor in a safe conditions.

4. The auxiliary feedwater system is reviewed for whether the flow is acceptable for transient control following a feedwater line break.

5. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

   For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other SRP sections interface with this section as follows:

1. General information on transient and accident analyses is provided in SRP Section 15.0.

2. Design basis radiological consequence analyses associated with design basis accidents are reviewed under SRP Section 15.0.3.

3. Effects of blow-down loads, including jet propulsion piping and component supports and the design bases for safety and relief valves are reviewed under SRP Sections 3.6.2 and 3.9.1 through 3.9.3. Design bases for safety and relief valves is also reviewed under SRP Section 3.9.3.

4. Values of the parameters in the analytical models of the reactor core are reviewed for compliance with plant design and specified operating conditions, acceptance criteria for fuel cladding damage limits are determined, and the core physics, fuel design, and core thermal-hydraulics data in the SAR analysis are reviewed under SRP Sections 4.2, 4.3, and 4.4.

5. Fracture toughness properties of the reactor coolant pressure boundary and reactor vessel are reviewed under Sections 5.2.3 and 5.3.1.

6. The response of the containment to feedwater line ruptures as to the effects of pressure and temperature on the containment functional capabilities is reviewed under SRP Section 6.2.1. Analytical methods for deriving mass energy releases exiting the postulated break are reviewed under SRP Section 6.2.1.3.

7. Aspects of the sequence described in the SAR are reviewed to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. This review includes the instruments and controls required to ensure automatic and manual auxiliary feedwater system initiation and flow indication in the control room and is performed under SRP Sections 7.1 through 7.7. The potential bypass modes and the possibility of manual control by the operator are also reviewed under SRP Sections 7.1 through 7.7.

8. The auxiliary feedwater system is reviewed to verify its ability to function following a steam line break given a single active component failure with either onsite or offsite power under SRP Section 10.4.9.

9. Fission product release assumptions for determining any offsite releases are evaluated and radiological consequences from a feedwater pipe break are verified as within acceptable limits are reviewed under SRP 15.6.5.

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

II. <u>ACCEPTANCE CRITERIA</u>

<u>Requirements</u>

The basic objective of the review of feedwater system pipe break events is to confirm that the reactor primary system is maintained in a safe status for break sizes up to and including a break equivalent in area to the double-ended rupture of the largest feedwater line.

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

1. General Design Criterion (GDC) 13, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of

appropriate controls to maintain these variables and systems within prescribed operating ranges.

2. GDC 17, as to onsite and offsite electric power systems for safety-related SSCs to function. The safety function for each power system (assuming the other system is not functioning) must be of sufficient capacity and capability so design conditions of the reactor coolant pressure boundary are not exceeded and the core is cooled in postulated accidents.

3. GDCs 27 and 28, as to the RCS design with appropriate margin so acceptable fuel design limits are not exceeded and core cooling capability is maintained.

4. GDC 31, as to RCS design with sufficient margin so the boundary is nonbrittle and the probability of fracture propagation is minimized.

5. GDC 35, as to design of the RCS and its auxiliaries for abundant emergency core cooling.

6. 10 CFR Part 100, as to calculated doses at the site boundary.

<u>SRP Acceptance Criteria</u>

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

1. Requirements for maintenance of adequate decay heat removal by the AFWS are in 10 CFR 50.34(f)(1)(ii), (TMI issue II E 1.1) and 10 CFR 50.34(f)(2)(xii), (TMI issue II E 1.2). Requirements for reactor coolant pump (RCP) operation are in 10 CFR 50.34(f)(1)(iii), (TMI issue 2 K 2). The reviewer should see Chapter 20 of the NRC FSAR for AP1000 to see how these post TMI requirements are met by the PRHR, the non-safety related start-up feedwater system (SUFWS) and the canned-motor RCPs of AP1000.

2. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures (American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III) for low-probability events and below 120 percent for very low-probability events like double-ended guillotine breaks.

3. The potential for core damage is evaluated for an acceptable minimum DNBR remaining above the 95/95 DNBR limit for pressurized-water reactors (PWRs) based on acceptable correlations (see SRP Section 4.4). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods not meeting these criteria unless, from an acceptable fuel damage model (see SRP Section 4.2) including the potential adverse effects of hydraulic instabilities, fewer failures can be shown to occur.

Any fuel damage calculated to occur must be of sufficiently limited extent that the core remains in place and intact with no loss of core cooling capability.

4. Calculated doses at the site boundary from any activity release must be a small fraction of the 10 CFR Part 100 guidelines.

5. The integrity of the RCPs should be maintained so loss of alternating current power and containment isolation do not result in seal damage.

6. The AFWS must be safety grade and automatically initiated when required.

7. Certain assumptions should be in the analysis of important parameters that describe initial plant conditions and postulated system failures:

   A. The power level assumed and number of loops operating at the initiation of the transient should correspond to the operating condition which maximizes accident consequences. These assumed initial conditions vary with the particular nuclear steam supply system and sensitivity studies are required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report as references if applicable.

   B. The assumptions as to whether offsite power is lost and the time of loss should be conservative. Offsite power may be lost simultaneously with the pipe break, the loss may occur during the accident, or offsite power may not be lost. A study should determine the most conservative assumption appropriate to the plant design reviewed. The study should take account of the effects that loss of offsite power (LOOP) has on reactor coolant and main feedwater pump trips and on the initiation of auxiliary feedwater and the consequent modification of the sequence of events.

   C. The effects (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) of the postulated feedwater line breaks on other systems should be considered consistently with the intent of Branch Technical Positions (BTP) 3-3 and BTP 3-4.

   D. The worst single active component failure should be assumed to occur in the systems required to control the transient. For new applications, LOOP should not be considered a single failure; feedwater pipe breaks should be analyzed with and without LOOP, as in assumption B, in combination with a single, active failure. (This position is based upon interpretation of GDC 17 as documented in the FSER for the ABB-CE System 80+ DC.)

   E. The maximum rod worth should be assumed to be held in the fully withdrawn position per GDC 25. An appropriate rod reactivity worth versus rod position curve should be assumed.

F. The core burn-up (time in core life) should be selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

G. The initial core flow assumed for the analysis of the feedwater line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin is the result for a feedwater line rupture inside containment; however, this assumption may not be the most conservative. For example, maximum initial core flow increases RCS cool-down and depressurization, decreases shutdown margin, and increases the possibility that the core will become critical and return to power. As it is not clear which initial core flow is most conservative, the applicant's assumption should be justified by appropriate sensitivity studies.

H. During the initial 10 minutes of the transient, if credit for operator action is required (i.e., RCP trip), an assessment for the limiting consequence must account for operator delay and/or error.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:

1. 10 CFR Part 100 specifies how the exclusion area, low population zone, and population center distance should be determined. Further, 10 CFR Part 100 radiation exposure criteria provide reference values for the site suitability determination based on postulated fission product releases from accidental events.

   10 CFR Part 100 applies to this section because it specifies the methodology for calculating radiation exposures at the site boundary for postulated accidents or events like loss of an RCP. For transients with moderate frequencies of occurrence, the calculated doses at the site boundary from any release of radioactive material must be a small fraction, less than 10 percent, of the 10 CFR Part 100 guidelines. For purposes of this review, consideration of the radiological consequences of any feedwater system pipe break must include the containment, confinement, and filtering systems. The applicant's source terms and methodologies as to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies.

2. GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to assure adequate safety, and of controls that can maintain these variables and systems within prescribed operating ranges.

   GDC 13 applies to this section because the reviewer evaluates the sequences of events, including automatic actuations of protection systems, and manual actions, and determines whether the sequence of events is justified, based upon the expected values of the relevant monitored parameters and instrument indications.

3. GDC 17 requires onsite and an offsite electric power systems to permit functioning of SSCs important to safety. The safety function for each power system (assuming the other system is not functioning) is to provide sufficient capacity and capability to assure that (A) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences (AOOs) and (B) the core is cooled and containment and other vital functions are maintained in postulated accidents.

   GDC 17 applies because review under this section covers feedwater system pipe breaks, which can be classed as AOOs or accidents, depending upon severity.

4. GDC 27 requires reactivity control systems designed with a combined capability, with poison added by the emergency core cooling system (ECCS), to control reactivity changes reliably to maintain core cooling capability under postulated accident conditions with appropriate margin for stuck rods.

   GDC 28 requires reactivity control systems designed with appropriate limits on the potential amount and rate of reactivity increase so the effects of postulated reactivity accidents 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 core cooling capability significantly. These postulated reactivity accidents must include consideration of rod ejection (PWR) (unless prevented by positive means), rod drop (boiling-water reactor (BWR)), steam line rupture, reactor temperature and pressure changes, and cold water addition.

   GDCs 27 and 28 apply because this SRP section is for the review of feedwater system pipe breaks inside and outside containment that can result in transient conditions affecting reactor coolant temperature and pressure with consequent changes in core reactivity. The SAR analyses of these transients must demonstrate that reactivity, pressure, and temperature changes will not be severe enough for an unacceptable impact on the reactor coolant pressure boundary or on core cooling capability. The analyses must be reviewed by the staff independently in accordance with this SRP section.

5. GDC 31 requires reactor pressure boundary design with sufficient margin to ensure that, when stressed under operation, maintenance, test, and postulated accident conditions, the boundary is nonbrittle and the probability of rapidly propagating fracture is minimal. The design must reflect consideration of service temperatures and other conditions of the boundary material under operation, maintenance, test, and postulated accident conditions and the uncertainties in determining material properties; effects of irradiation on material properties; residual, steady state, and transient stresses; and flaw sizes.

   GDC 31 applies because this SRP section is for the review of feedwater system pipe breaks inside and outside containment that could result in transient reactor coolant temperature and pressure conditions that could affect the reactor coolant pressure boundary adversely. A feedwater system pipe break could result in either an RCS cool-down by excessive energy discharge through the break or an RCS heat-up by reduced feedwater flow to the affected steam generator. Heat-up of the reactor coolant by reduced feedwater flow to the affected steam generator and by the subsequent addition of decay heat could result in undue stress on the RCS pressure boundary. The amount of stress to which the reactor coolant pressure boundary is subjected depends upon AOO severity, which is assessed in the SAR and reviewed by the staff in accordance with this SRP section.

6. GDC 35 requires a system for abundant emergency core cooling. The system safety function is to transfer heat from the reactor core following any loss of reactor coolant at a rate to prevent fuel and clad damage that could interfere with continued effective core cooling and limit fuel clad metal-water reaction to negligible amounts.

   GDC 35 applies because this SRP section is for the review of feedwater system pipe breaks both inside and outside containment that could result in transient reactor coolant temperature conditions that could challenge the ECCS. A feedwater system pipe break could result in either an RCS cool-down by excessive energy discharge through the break or an RCS heat-up by reduced feedwater flow to the affected steam generator. Heat-up of the reactor coolant by reduced feedwater flow to the affected steam generator and by the subsequent addition of decay heat could initiate ECCS reduction of the core coolant temperature to an acceptable level to prevent fuel and clad damage that could interfere with continued effective core cooling and limit fuel clad metal-water reaction to negligible amounts. The severity of this AOO is assessed in the SAR and reviewed by the staff in accordance with this SRP section.

III. REVIEW PROCEDURES

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

These review procedures are based on the identified SRP acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant’s evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

The procedures are used during reviews of construction permit, operating license, and COL applications. During the construction permit review the values of system parameters and setpoints in the analysis are preliminary in nature and subject to change. At the operating license or COL review stage, final values should be in the analysis, and the reviewer should compare these to the limiting safety system settings in the proposed technical specifications.

1. The values of system parameters and initial core and system conditions as input to the model are reviewed and compared to the initial conditions listed in subsection II of this SRP section. Of particular importance are the reactivity coefficients and control rod worths in the applicant's analysis and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The applicant’s justification for selection of the core burn-up yielding the minimum margins is evaluated. Reactivity parameter values in the applicant's analysis also are reviewed.

2. Analytical models should be of sufficient detail to simulate the reactor coolant (primary), steam generator (secondary), and auxiliary systems. The applicant’s equations, sensitivity studies, proposed models, and justification for methods as conservative

compared to appropriate test data are reviewed. The pressurizer is of particular importance in the modeling of the over-pressure transient, the likely result of a large feedwater line break. Assumptions for pressurizer spray performance if credited should be reviewed as well as heat transfer by condensation within the pressurizer steam space. Test data examples which might be useful in validation of pressurizer models are in "The Pressure Response of a PWR Pressurizer During an Insure Transient," Transactions of the American Nuclear Society, 1983 Annual Meeting, Detroit, MI, June 12-16, 1983.

3. Credit taken for a reactor trip signal or for ESS actuation should be reviewed for the ability of the instrumentation and control systems to respond as assumed under accident conditions.

4. The AFWS ability to supply adequate feedwater flow to the unaffected steam generators during the accident and subsequent shutdown is evaluated as to availability and capability to effect an orderly shutdown. As AFWS designs are diverse and may require both automatic and manual actuation, pre-operational tests should be specified for any necessary operator actions and for the maximum times for their completion.

   To the extent necessary, the reviewer evaluates the effect of system and component single, active failures that may alter the course of the accident. For new applications, the LOOP is not a single, active failure but an addition to a single, active failure as addressed in subsection II.6.D of this SRP section. This phase of the review uses the system review procedures described in the SRP sections for SAR Chapters 5, 6, 7, 8, and 10. During the transient the variations with time of parameter listed in Sections 15.X.X.3(C) and 15.X.X.4(C) of the Standard Format, Regulatory Guide 1.70, are reviewed. The more important of these parameters for the feedwater line break accident (as listed in subsection I of this SRP section) are compared to those predicted for other similar plants for whether they are within the expected range.

5. The reviewer confirms that the amount of secondary coolant expelled from the system is calculated conservatively by evaluation of the applicant's methods and assumptions, by comparison with an acceptable analysis on another plant of similar design, or by comparison with staff calculations.

   The reviewer confirms an SAR commitment to conduct pre-operational tests to verify that valve discharge rates and response times (*e.g.*, opening and closing times (delay times) for main feedwater, auxiliary feedwater, turbine and main steam isolation, and steam generator, pressurizer relief, and safety valves) are modeled conservatively in the accident analyses. In addition, pre-operational testing should include verification of reactor trip delay times, startup delay times for AFWS actuation, safety injection signal delay time, and delay times for delivery of any high-concentration boron injection required to bring the plant to a safe shutdown condition.

6. Using the information developed in the review, the reviewer evaluates the radiological consequences of the design-basis feedwater line break. This evaluation is based on a qualitative comparison to the results of the design-basis steam line break or on a detailed analysis using the approach described in the SRP Section 15.0.3.

7. AFWS reliability and operability are reviewed for compliance with 10 CFR 50.34(f)(1)(ii) and 10 CFR 50.34(f)(2)(xii) as to AFWS performance requirements following feedwater piping failures. (See SRP Acceptance Criteria for AP1000 applicability).

   Reactor RCP seal reliability and integrity during loss of alternating-current power and loss of coolant to the seals (*e.g.*, a result of containment isolation) are reviewed for compliance with 10 CFR 50.34(f)(1)(iii). (See SRP Acceptance Criteria for AP1000 applicability).

8. 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

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

The staff concludes that the applicant’s analysis of consequences of postulated feedwater line breaks meets the requirements of GDCs 13, 17, 27, 28, 31, and 35 for ability to insert control rods and ability to cool the core, 10 CFR Part 100 guidelines for radiological doses at the site boundary, and applicable Three Mile Island Action Plan Items. This conclusion is based upon the following findings:

1. The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of protection systems, automatic and manual, occurred at values of monitored parameters that were within the instruments’ prescribed operating ranges.

2. The applicant meets GDCs 27 and 28 requirements by demonstrating minimal fuel damage, maintained ability to insert the control rod, and no loss of core cooling capability. The minimum DNBR for any fuel rod was _______ with the result of __ percent of the rods experiencing clad perforation.

3. The applicant meets GDC 31 requirements for demonstrating primary system boundary capability to withstand the postulated accident.

4. The applicant meets GDC 35 requirements for demonstrating emergency cooling system adequacy for abundant core cooling and reactivity control (via boron injection).

5. The analyses of effects of feedwater line break accidents inside and outside containment during various modes of operation with and without offsite power have been reviewed and evaluated by a mathematical model previously reviewed and found acceptable by the staff.

6. The input parameters for this model were reviewed and found suitably conservative.

7. The radioactivity release description has been evaluated by the computer code SARA for a conservative description of the plant response to the accident. We use a decontamination factor of ______ between the water and steam phases and a X/Q value of ______ sec/m^3 in our evaluation of radiological consequences. The calculated doses are presented in Table _____. Technical specification limits on primary and secondary coolant activities limit potential doses to small fractions of the 10 CFR Part 100 exposure guidelines. The potential doses are within 10 CFR Part 100 exposure guidelines even if the accident occurs with an iodine spike.

8. The applicant meets 10 CFR 50.34(f)(1)(ii) and 10 CFR 50.34(f)(2)(xii) requirements for demonstrating the adequacy of the auxiliary feedwater design to remove decay heat following feedwater piping failures. (See SRP Acceptance Criteria for AP1000 applicability).

9. The applicant meets 10 CFR 50.34(f)(1)(III) requirements for demonstrating RCP seal capability to withstand the postulated accident. (See SRP Acceptance Criteria for AP1000 applicability).

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL 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 submitted six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

Implementation schedules for compliance with parts of the method addressed here are in the referenced regulatory guides and NUREGs, except for the position in subsections II.6.D and III of this SRP section on LOOP and assumed single failures. This new position applies to new applications (for a construction permit, a manufacturing license, or DC).

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

    A. GDC 13, "Instrumentation and Control."

    B. GDC 17, "Electric Power Systems."

    C. GDC 27, "Combined Reactivity Control System Capability."

    D. GDC 28, "Reactivity Limits."

    E. GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

    F. GDC 35, "Emergency Core Cooling."

2. 10 CFR Part 100, "Reactor Site Criteria."

3. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

4. Branch Technical Position 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

5. Branch Technical Position 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment."

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