U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.6.3 RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE FAILURE (PWR)

REVIEW RESPONSIBILITIES

Primary - ~~Accident Evaluation Branch (AEB)~~Emergency Preparedness and Radiation Protection Branch (PERB)[1]

Secondary - Reactor Systems Branch ~~(RSB)~~(SRXB)[2]

I. AREAS OF REVIEW

This Standard Review Plan (SRP)[3] section covers the review of the radiological consequences of a postulated steam generator tube failure accident at a pressurized water reactor (PWR) facility and includes the following:

1. A review of the sequence of events and plant procedures for recovery from the accident, as described by the applicant, with and without offsite power available, to ~~assure~~ensure[4] that the most severe case of radioactive releases has been considered.

2. A review of the models and assumptions used by the applicant for the calculation of the thyroid and whole-body doses for the postulated accident.

3. An independent calculation by the staff of the thyroid and whole-body doses for the accident.

4. A comparison of the doses calculated by the applicant and by the staff with the appropriate exposure guide lines, as stated in subsection II below.

5. An evaluation of the technical specifications on the primary and secondary coolant iodine activity concentration.

6. An evaluation of the radiological consequences of steam generator tube failure concurrent with loss of offsite power and the most limiting single failure.[5]

The review includes two cases for the reactor coolant iodine concentration corresponding to (1) a preaccident iodine spike and (2) a concurrent iodine spike. The potential for fuel failures resulting from the postulated accident is routinely evaluated by the ~~RSB~~SRXB[6] as part of its secondary review responsibility for SRP Section 15.6.3[7] and such information is provided to the ~~AEB~~PERB[8] as an additional source of iodine activity in the reactor coolant for consideration in the evaluation of the radiological consequences.

Review Interfaces:[9]

PERB also performs the following reviews under the SRP sections indicated:[10]

Reviews the atmospheric dispersion factors used in the consequence analysis as part of its primary review responsibility for SRP Section 2.3.4.[11]

In addition, the PERB will coordinate other branches' evaluations that interface with the overall review of the system as follows:[12]

1. The steam generator tube failure accident sequence is evaluated by the Containment Systems and Severe Accident Branch (SCSB) in terms of the time at which primary-to-secondary leakage is terminated, the potential for steam generator overfill and the most limiting single failure as part of its review of SRP Sections 6.2.1, 6.2.1.1, 6.2.1.3, and 6.2.4.[13]

2. Emergency operating procedures for the steam generator tube failure accident sequence are evaluated by the Human Factors Assessment Branch (HHFB) as part of its review of SRP Section 13.5.2. This information is provided to PERB as input for its evaluation of the radiological consequences of a steam generator tube rupture accident.[14]

3. Requirements for technical specifications on radioactive materials in primary and secondary coolant will be coordinated with the Technical Specifications Branch (TSB) as part of its primary review responsibility for SRP Section 16.0.[15]

4. The Probabilistic Safety Assessment Branch (SPSB) reviews the contribution of steam generator tube rupture accidents to plant risk as part of its primary review responsibilities for SRP Section 19.1 (proposed). This review may include the applicant’s assessment of design improvements to reduce containment bypass leakage, and best-estimate evaluation of the plant response to a steam generator tube rupture to identify potential design vulnerabilities.[16]

For those areas of review identified as part of the primary responsibility of other branches, the acceptance criteria and methods of application are contained in the referenced SRP section.[17]

I. ACCEPTANCE CRITERIA

The acceptance criteria are based on the relevant requirements of 10 CFR Part 100 as it relates to mitigating the radiological consequences of an accident. The plant site and the dose mitigating engineered safety features are acceptable with respect to the radiological consequences of a postulated steam generator tube failure accident at a PWR facility if the calculated whole-body and thyroid doses at the exclusion area and the low population zone outer boundaries do not exceed the following exposure guidelines:

1. For the postulated accident with an assumed preaccident iodine spike in the reactor coolant and for the postulated accident with the highest worth control rod stuck out of the core, the calculated doses should not exceed the guideline values of 10 CFR 100.11~~Part 100, Section 11~~[18] ~~(Ref. 1)~~.[19]

2. For the postulated accident with the equilibrium iodine concentration for continued full-power operation in combination with an assumed accident initiated iodine spike, the calculated doses should not exceed a small fraction of the above guideline values, i.e., 10% or 25 mSv (2.5 rem) and 300 mSv (30 rem),[20] respectively, for the whole-body and thyroid doses.

The methodology and assumptions for calculating the radiological consequences should reflect the regulatory positions of Regulatory Guide 1.4 ~~(Ref. 2)~~[21] except for the atmospheric dispersion factors which are reviewed under SRP Section 2.3.4. Plant technical specifications are required for iodine activity in the primary and secondary coolant systems. These specifications are acceptable if the calculated potential radiological consequences from the steam generator tube failure accident are within the exposure guidelines for the above two cases.

A plant-specific technical specification is required for iodine activity in the primary and secondary coolant systems. The specification is acceptable with respect to a postulated failure if the calculated doses resulting from the failure are within the above exposure guidelines.[22]

Technical Rationale:[23]

The technical rationale for application of these acceptance criteria is discussed in the following paragraphs:[24]

Compliance with 10 CFR 100.11 requires that the exclusion area, low population zone, and population center distance be determined based on a fission product release from the plant and meteorological conditions pertinent to the site.

Identification of an exclusion area, a low population zone, and a population center distance is an integral aspect of the siting criteria for new nuclear power plants. Radiation dose reference values — a total radiation dose to the whole body in excess of 250 mSv (25 rem) or a total radiation dose to the thyroid from iodine exposure in excess of 3000 mSv (300 rem) — are associated with the exclusion area and the low population zone. To demonstrate that the proposed nuclear plant design will meet these reference values at the exclusion area and low population zone boundaries, a calculation of the expected offsite radiation doses is performed

using a radioactive source term based on the concentration of radioactive material in the primary and secondary coolant, primary-to-secondary leakage, the size of the break, the occurrence of an iodine spike occurs before or concurrent with the event, and site atmospheric dispersion characteristics. For a steam generator tube failure event with an assumed preaccident iodine spike, calculated doses should not exceed the full reference values specified in 10 CFR 100.11. For equilibrium iodine concentrations at continued full-power operation in combination with an accident-initiated iodine spike, the calculated doses should not exceed a small fraction of the reference values (i.e., 10%) for the whole-body and thyroid doses.

Meeting the requirements of 10 CFR 100.11 for doses resulting from steam generator tube failure provides assurance that offsite radiation doses from postulated accidents will not exceed the guideline doses specified in 10 CFR Part 100.[25]

III. REVIEW PROCEDURES

The reviewer selects and emphasizes specific aspects of this SRP section as appropriate for a particular plant. The judgment which areas need to be given attention and emphasis during the review is based on ~~a~~the reviewer's determination ~~if the material presented is similar to that recently reviewed on other plants and~~of[26] whether items of special safety significance are involved.

At the construction permit (CP) or standard design certification[27] stage, the review is limited to a brief survey of the pertinent portions of the safety analysis report (SAR)[28] regarding the plant design and the applicant's accident evaluation to determine that there are no unusual features which would prevent limitation of radiological consequences to acceptable levels by appropriate limits on coolant activity concentrations. The detailed review of the radiological consequences of a steam generator tube failure is done at the operating license (OL) or combined license (COL)[29] stage when system parameters and accident analyses are fully developed.

For standard design certification reviews, the calculation of hypothetical offsite radiological consequences of a steam generator tube failure is performed using proposed technical specification limits on coolant radioactivity and limits on atmospheric diffusion parameters specified in the site parameter envelope.[30]

Standard Technical Specifications (References 5, 6, and 7)[31] for each of the three PWR vendors' nuclear steam supply system (NSSS)[32] include limits on the primary and secondary coolant activities which are used in the staff's independent dose calculations ~~(Ref. 3, 4, and 5)~~.[33] If the applicant proposes to use these standard limits and the plant is one of the standard NSSS/BOP plants for which the tube failure accident has been evaluated generically with the standard coolant activity and leakage limits, the reviewer need not reevaluate the offsite doses from this accident provided that the atmospheric dispersion factors (χ/Q values) for the site under review do not exceed the limiting χ/Q values used in the generic review of the standard plant tube failure accident.

The review of the steam generator tube failure accident at the OL or COL[34] stage includes the following:

1. ~~Review of the~~The[35] applicant's description of the tube failure accident, with and without offsite power. This includes a review of the sequence of events, the bases for the occurrence, and assurance of an adequate degree of conservatism.

2. ~~Review of the~~The[36] signals available to the reactor operator that indicate the occurrence of the accident and the state of the system throughout the recovery period. Automatic and required manual operations by the operator as a function of time are reviewed. The ~~AEB~~PERB[37] reviewer verifies with the ~~RSB~~SRXB and HHFB[38] the acceptability of the applicant's description of events, including operator actions, to ~~assure~~ensure that the most ~~severe case~~ limiting single failure[39] has been considered with respect to the release of fission products and calculated doses. This part of the review should include an HHFB evaluation of operator actions related to preventing steam generator overfill.[40]

3. The plant-specific postaccident thermohydraulic characteristics and radiological consequences of this accident ~~are plant-specific~~.[41] The reviewer, determines postaccident thermohydraulic profiles and compares these with those presented by the applicant. The purpose of such comparison is not to attain an exact match but to confirm the validity of the applicant's calculated results.

4. The appropriate atmospheric dispersion factors (χ/Q values) for the staff's independent dose analysis ~~will be~~as[42] determined by the assigned meteorologist in accordance with SRP Section 2.3.4.

5. Determination of the initial primary and secondary coolant activity concentrations. The reviewer assumes the primary and secondary coolant activity concentrations allowed by the technical specifications (SAR Chapter 16 or the standard technical specifications given in References~~:~~ ~~3~~5, ~~4~~6, and ~~5~~7)[43] as equilibrium conditions prior to the accident.

6. Determination of iodine spiking effects. For the dose calculations the following two cases of iodine spiking are analyzed:

   (a) A reactor transient has occurred prior to the postulated steam generator tube failure accident and has raised the primary coolant iodine concentration to the maximum value permitted by the standard technical specifications (i.e., a preaccident iodine spike case). The primary coolant iodine concentration for this case is obtained from Figure 3.4-1 of the NSSS vendor standard technical specification (References~~:~~ ~~3~~5, ~~4~~6, or ~~5~~7)[44] or from the plant-specific technical specifications proposed in Chapter 16 of the applicant's SAR.

   (b) The reactor trip or the primary system depressurization associated with the postulated accident creates an iodine spike in the primary system (References. ~~6~~4 and ~~7~~8).[45] The increasing primary coolant iodine concentration is estimated using a spiking model which assumes that the iodine release rate from the fuel rods to the primary coolant — expressed in becquerels (curies)[46] per unit time —

increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value stated in the NSSS vendor standard technical specifications or from the plant-specific technical specifications (i.e., concurrent iodine spike case).

7. Evaluation of the effects of fuel failure. As a result of the steam generator tube failure accident, fuel failures can occur, releasing fission products into the reactor coolant and thus making additional activity available for release to the atmosphere. The ~~RSB~~SRXB[47] reviews the effects of the accident on the core thermal margins and the associated amount of fuel failures, assuming that the highest worth control rod is stuck at its fully withdrawn position. The ~~RSB~~SRXB,[48] as a secondary review branch, informs the ~~AEB~~PERB[49] of the fuel failure estimate. If the accident is predicted to cause such fuel failure, the dose analysis will be performed with the corresponding iodine activity but without a concurrent iodine spike.

8. Determination of the primary to secondary system leakage in the unaffected steam generators. The operating primary-to-secondary leakage is assumed to exist in the unaffected steam generators at the maximum rate allowed by the Standard Technical Specifications (References. ~~3~~5, ~~4~~6, and ~~5~~7)[50] or from the plant-specific technical specifications.[51] ~~This value is 1 gpm. However, a lower value may be needed to limit the consequences of other events such as a control rod ejection accident.~~[52]

9. Determination of the coolant flow through the failed tube. In conjunction with review step 3 above, the flow rates through the two ends of the failed tube are calculated using a suitable flow model, taking credit for critical flow where appropriate.

10. Determination of most limiting single failure. Evaluate the steam generator tube failure event with the following assumptions:

    - Steam generator tube failure without loss of offsite power,
    - Steam generator tube failure with loss of offsite power, and
    - Steam generator tube failure with loss of offsite power and the most limiting single failure.[53]

11. Determination of the iodine transport to the atmosphere. The iodine transport model to be used is described in Reference ~~8~~9[54]. A fraction of the iodine in the primary coolant escaping to the secondary system is assumed to become airborne immediately due to flashing and atomization. Credit may be given for "scrubbing" of iodine contained in the steam phase and in the atomized primary coolant droplets suspended in the steam phase for release points which are below the steam generator water level. That fraction of the primary coolant iodine which is not assumed to become airborne immediately enters the secondary system water and is assumed to become airborne at a rate determined by the steaming rate and iodine partition coefficient. An iodine partition coefficient of 100 between steam generator water and steam phases may be conservatively assumed

unless the applicant presents reasonable evidence that the use of some other value is justified.

12. Calculation of the exclusion area and low population zone boundary doses. The reviewer performs an independent calculation of the doses for the steam generator tube failure accident, using the two iodine concentrations in item (6) above. A breathing rate of 3.47 x 10^{-4} m^3/sec is used in the calculation of thyroid doses for the first 8 hours following the steam generator tube failure and the dose conversion factors are in accordance with Regulatory Guide 1.4 ~~(Ref. 3)~~.[55]

13. Review of dose calculations. The whole-body and thyroid doses calculated by the staff and by the applicant are compared with the acceptance criteria stated in subsection II. If the doses calculated by the staff are not within the exposure guidelines (i.e., they are not less than 10% of 10 CFR ~~Part 100, Section 11~~100.11[56]), then the staff will pursue alternatives with the applicant to reduce the doses to within the guideline values.

14. For new applications, the assessment of potential design improvements to mitigate the amount of containment bypass leakage that could result from a SGTR is reviewed per guidance contained in SECY 93-087 (Reference 3).[57]

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[58]

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided by the applicant and that the applicant's analysis and the staff's independent calculations support conclusions of the following type, to be included in the ~~AEB~~PERB[59] safety evaluation report (SER)[60] at the ~~operating license~~OL or COL[61] stage:

The steam generator tube failure accident has been evaluated ~~with and without a concurrent loss of offsite power.~~assuming the following:

- Steam generator tube failure without loss of offsite power,
- Steam generator tube failure with loss of offsite power, and
- Steam generator tube failure with loss of offsite power and the most limiting single failure.[62]

The assumptions used in our analysis are listed in Table _____. The calculated doses are presented in Table _____.

The staff concludes that the distances to the exclusion area and to the low population zone outer boundaries for the (insert PLANT NAME) site, in conjunction with the operation of the dose-mitigating ~~ESF~~ engineered safety feature systems,[63] are sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated steam generator tube failure accident do not exceed (a) the exposure guidelines as set forth in 10 CFR ~~Part 100, Section 11~~100.11[64] for the accident with an assumed preaccident iodine spike or with the highest worth control rod stuck out of the core and (b) 10% of these exposure guidelines, for the accident with an equilibrium iodine concentration in combination with an assumed accident generated iodine spike.

The staff conclusion is based on (1) the staff review of the applicant's analysis of the radiological consequences, (2) the independent dose calculation by the staff using conservative assumptions including atmospheric dispersion factors as discussed in Chapter 2 of this report, (3) the applicant's analysis and the staff's independent dose calculations which were performed using the guidelines of Regulatory Guide 1.4, and (4) the (insert NSSS VENDOR) Standard Technical Specifications for the iodine concentration in the primary and secondary coolant system, and for the primary to secondary leakage in the steam generators. The staff will review the (PLANT NAME) specific technical specifications to ~~assure~~ensure that the dose guidelines stated above are not exceeded.

The following paragraph is inserted prior to the last paragraph if fuel damage is found to be a possible consequence of the accident:

The steam generator tube failure accident has also been evaluated with _____% fuel damage in the core as a result of the most reactive control rod remaining fully withdrawn. The resulting doses, listed in Table 15._____, are within the guidelines of 10 CFR Part 100.

At the construction permit stage, the following paragraph is included in the staff's SER:

On the basis of our experience with the evaluation of steam generator tube failure accidents for pressurized water reactor plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible primary and secondary coolant system radioactivity concentrations so that potential offsite doses are small. At the operating license stage the staff will include appropriate limits on primary and secondary coolant activity concentrations in the technical specifications.

For a standard design certification review, the following paragraph is included in the staff's SER:

The staff has calculated hypothetical offsite radiological consequences of a steam generator tube failure using limits on the concentration of radioactive materials in the coolant based on the proposed technical specifications and atmospheric diffusion parameters specified in the site parameter envelope. The hypothetical offsite consequences are within the reference values of 10 CFR Part 100.[65]

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[66]

V. IMPLEMENTATION

The following provides guidance to applicants and licensees regarding the staff's plans for using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[67] Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of 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.[68]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides.

VI. REFERENCES

1. 10 CFR ~~Part 100, Section 11~~100.11,[69] "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."

3. SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," April 2, 1993.[70]

~~7~~4.[71] A. K. Postma and P. S. Tam, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," NUREG-0409, USNRC, January 1978.

~~3~~5. Standard Technical Specifications for Combustion Engineering ~~PWRs, NUREG-0212.~~ Plants, NUREG-1432.[72]

~~4~~6. Standard Technical Specifications for Westinghouse ~~PWRs, NUREG-0452.~~ Plants, NUREG-1431.[73]

~~5~~7. Standard Technical Specifications for Babcock and Wilcox ~~PWRs, NUREG-0103.~~ Plants, NUREG-1430.[74]

~~6~~8. W. F. Pasedag, "Iodine Spiking in BWR and PWR Coolant Systems," CONF-770708, 3 717 (1977).

~~8~~9. R. R. Bellamy, "A Regulatory Viewpoint of Iodine Spiking During Reactor Transients," Trans. Am. Nucl. Soc., 28 (1978).

SRP Draft Section 15.6.3
Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Current PRB name and abbreviation	Changed PRB to Emergency Preparedness and Radiation Protection Branch (PERB).
2.	Current SRB abbreviation	Changed SRB to SRXB.
3.	Editorial	Defined SRP.
4.	Editorial	Global change for this SRP section of "assure" to "ensure" for proper usage.
5.	Integrated Impact No. 196	Added reference to most limiting single failure in AREAS OF REVIEW.
6.	Current SRB abbreviation	Changed SRB to SRXB.
7.	Editorial	Identified secondary review responsibility for SRXB.
8.	Current PRB abbreviation	Changed PRB to PERB.
9.	SRP-UDP format item	Added "Review Interfaces" to AREAS OF REVIEW and organized in numbered paragraph form.
10.	Editorial	Added typical lead-in sentence for interfacing sections that are the responsibility of the same PRB as the section under review.
11.	Editorial	The existing SRP section contains an interface with SRP Section 2.3.4 in the Acceptance Criteria and Review Procedures. This change incorporates the existing interface in the new Review Interfaces subsection.
12.	Editorial	Added typical lead-in sentence for interfacing sections that are the responsibility of other PRBs.
13.	Integrated Impact Nos. 193, 195, and 196	Added review interface with SCSB on the duration of primary to secondary leakage, steam generator overfill and the most limiting single failure.
14.	Integrated Impact No. 193	Added review interface with HHFB on emergency operating procedures.
15.	Integrated Impact No. 194	Added review interface with TSB.
16.	**Integrated Impact 195.**	Added a review interface with SRP Section 19.1 (currently numbered 19.2) regarding the review of steam generator tube rupture accidents by the probabilistic safety assessment branch. The risk contribution of steam generator tube ruptures, including the assessment of potential mitigating systems was evaluated in Chapter 19 of the staff's FSER for the CE System 80+ design.

Item	Source	Description
17.	SRP-UDP format item	Added standard paragraph on responsibilities of PRBs.
18.	Editorial	Corrected format for 10 CFR 100.11.
19.	SRP-UDP format item	Deleted unnecessary callout for Ref. 1.
20.	SRP-UDP format item	Added metric units.
21.	SRP-UDP format item	Deleted unnecessary callout for Ref. 2.
22.	Integrated Impact No. 194	Added paragraph on requirements for technical specifications to conform to SRP section 15.6.2.
23.	SRP-UDP format item	Added "Technical Rationale" to ACCEPTANCE CRITERIA and organized in numbered paragraph form to describe the bases for referencing the regulations.
24.	SRP-UDP format item	Added lead-in sentence for "Technical Rationale."
25.	SRP-UDP format item	Added technical rationale for 10 CFR 100.11.
26.	Editorial	Deleted reference to recent reviews because there may be none, and the reviewer's determination of safety significance is more definitive.
27.	SRP-UDP format item	Added standard design certification review per 10 CFR Part 52.
28.	Editorial	Defined "SAR" as "safety analysis report."
29.	SRP-UDP format item	Added COL review per 10 CFR Part 52.
30.	SRP-UDP format item	Added calculation of offsite consequences for a standard design certification using coolant radioactivity specified in the proposed technical specifications and atmospheric diffusion parameters specified in the site parameter envelope.
31.	Editorial	Moved reference identification for clarity.
32.	Editorial	Defined NSSS.
33.	Editorial	Moved reference identification for clarity.
34.	SRP-UDP format item	Added COL review per 10 CFR Part 52.
35.	Editorial	Removed redundant words.
36.	Editorial	Removed redundant words.
37.	Current PRB abbreviation	Changed PRB to PERB.
38.	Current review branch abbreviation	Changed review branches to SRXB and HHFB.
39.	Integrated Impact No. 196	Added reference to most limiting single failure in REVIEW PROCEDURES.

Item	Source	Description
40.	Integrated Impact No. 193	Added consideration of steam generator overfill in REVIEW PROCEDURES.
41.	Editorial	Revised wording to conform to opening sentence.
42.	Editorial	Revised wording to conform to opening sentence.
43.	Editorial	Revised the reference citation in accordance with SRP-UDP guidance and renumbered the references to correspond with changes to the REFERENCES subsection.
44.	Editorial	Revised the reference citation in accordance with SRP-UDP guidance and renumbered the references to correspond with changes to the REFERENCES subsection.
45.	Editorial	Revised the reference citation in accordance with SRP-UDP guidance and renumbered the references to correspond with changes to the REFERENCES subsection.
46.	SRP-UDP format item	Added metric units.
47.	Current review branch abbreviation	Changed review branch to SRXB.
48.	Current review branch abbreviation	Changed review branch to SRXB.
49.	Current PRB abbreviation	Changed PRB to PERB.
50.	Editorial	Revised the reference citation in accordance with SRP-UDP guidance and renumbered the references to correspond with changes to the REFERENCES subsection.
51.	Editorial	Added reference to plant-specific technical specifications for completeness.
52.	Editorial	Deleted specification of primary to secondary leakage limit and reference to control rod ejection accident as unnecessary.
53.	Integrated Impact No. 196	Added consideration of steam generator tube failure with loss of offsite power and the most limiting single failure to REVIEW PROCEDURES.
54.	Editorial	Revised the reference number to be consistent with reordering of references in the "REFERENCES" subsection.
55.	SRP-UDP format item	Deleted unnecessary callout for Ref 3.
56.	Editorial	Corrected citation format for 10 CFR 100.11.
57.	**Integrated Impact 195**	Added a review procedure related to staff positions in SECY 93-087 regarding steam generator tube rupture accidents.

Item	Source	Description
58.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
59.	Current PRB abbreviation	Changed PRB to PERB.
60.	Editorial	Provided "SER" as initialism for "safety evaluation report."
61.	SRP-UDP format item	Added reference to COL per 10 CFR Part 52.
62.	Integrated Impact No. 196	Added consideration of steam generator tube failure with loss of offsite power and the most limiting single failure to EVALUATION FINDINGS.
63.	Editorial	Defined ESF.
64.	Editorial	Corrected format for 10 CFR 100.11.
65.	SRP-UDP format item	Added paragraph to describe evaluation findings for standard design certification applications
66.	SRP-UDP Format Item, Implement 10 CFR 52 Related Changes	To address design certification reviews a new paragraph was added to the end of the Evaluation Findings. This paragraph addresses design certification specific items including ITAAC, DAC, site interface requirements, and combined license action items.
67.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
68.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
69.	Editorial	Corrected format for 10 CFR 100.11.
70.	**Integrated Impact 195.**	Added reference to SECY 93-087, which is cited with regard to staff positions associated with steam generator tube rupture accident analyses for new designs.
71.	Editorial	Reordered and renumbered references in accordance with SRP-UDP guidance.
72.	Integrated Impact No. 194	Corrected reference to latest version of Standard Technical Specifications.
73.	Integrated Impact No. 194	Corrected reference to latest version of Standard Technical Specifications.
74.	Integrated Impact No. 194	Corrected reference to latest version of Standard Technical Specifications.

Integrated Impact No.	Issue	SRP Subsections Affected
193	Consideration should be given to including a Review Procedure to confirm that the applicant has addressed concerns related to the potential for steam generator overfill during a steam generator tube rupture.	Subsection I, AREAS OF REVIEW, Review Interfaces, subparagraph 2 Subsection I, AREAS OF REVIEW, Review Interfaces, subparagraph 3 Subsection III, REVIEW PROCEDURES, fourth paragraph, subparagraph 2
194	Consider incorporating the improved Standard Technical Specifications in the Review Procedures.	Subsection I, AREAS OF REVIEW, Review Interfaces, subparagraph 4 Subsection II, ACCEPTANCE CRITERIA, third paragraph Subsection VI, REFERENCES, Reference 3 Subsection VI, REFERENCES, Reference 4 Subsection VI, REFERENCES, reference 5
195	Consider incorporating a Review Procedure to assess the Applicant's consideration of potential design features that would mitigate the amount of containment bypass that could result from steam generator tube failure.	Subsection I, AREAS OF REVIEW, first paragraph, subparagraph 6 Subsection I, AREAS OF REVIEW, Review Interfaces, subparagraph 2 Subsection III, REVIEW PROCEDURES, fourth paragraph, subparagraph 10 Subsection III, REVIEW PROCEDURES, fourth paragraph, subparagraph 13 Subsection IV, EVALUATION FINDINGS, first paragraph, third subparagraph Subsection IV, EVALUATION FINDINGS, third paragraph, first subparagraph

196	Consideration should be given to revising the Review Procedures and Areas of Review to assure the most limiting single failure is considered for the steam generator tube failure analysis.	Subsection I, AREAS OF REVIEW, first paragraph, subparagraph 7 Subsection I, AREAS OF REVIEW, Review Interfaces, subparagraph 2 Subsection III, REVIEW PROCEDURES, fourth paragraph, subparagraph 2 Subsection III, REVIEW PROCEDURES, fourth paragraph, subparagraph 11 Subsection IV EVALUATION FINDINGS, first paragraph, first subparagraph
1230	Revise the SRP to incorporate the new and revised requirements from proposed rulemaking 59 FR 52255.	This is a placeholder II and will not be processed further.