

## **APPENDIX I**

### **Integrated Impacts**

# **Recommendations that Resulted in a Proposed Change to NUREG-0800**

## **APPENDIX I**

### **Integrated Impacts**

**Integrated Impact Number: 1      SRP Section Number: 3.2.2**

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures and develop Appendices for the quality group classification of additional pressure retaining systems and components for fluid systems important to safety.

Reg. Guide 1.70 directs applicants to assign quality group classifications to "fluid systems or portions of fluid systems important to safety." The scope of the review indicated by the SRP Review Procedures subsection is consistent with Reg. Guide 1.70. Reg. Guide 1.26 provides the staff position regarding the assignment of quality group classifications to "safety-related components containing water, steam, or radioactive material."

The SRP Review Procedures subsection and Appendices A and B discuss systems and components important to safety for which Reg. Guide 1.26 classification positions are used, for which quality group classification positions supplemental to those stated in Reg. Guide 1.26 exist. Various regulatory documents (e.g., Reg. Guide 1.96; SRP Sections 6.2.4 and 9.3.3; Branch Technical Position CSB 6-3; and Generic Letter 90-06) contain information which appears to be supplemental to Reg. Guide 1.26 with respect to the classification of specific systems and components. Currently, Review Procedures refer to Appendices C and D for additional guidance regarding quality group classification. Appendices C and D have not been developed.

Consideration should be given to revising Review Procedures and developing Appendices C and D to address system quality group classification positions supplemental to Reg. Guide 1.26.

Consideration should also be given to including the following types of information in Appendices C and D; (1) systems/components, (2) quality group classification for each, (3) where appropriate, references to documents that identify positions on pressure boundary design and quality assurance supplemental to Reg. Guide 1.26 guidance. This information could be provided in a table format.

Several Integrated Impacts associated with SRP Section 3.2.2 make references to adding information to Appendices C and/or D. References to Appendices C or D in these other impacts are made in the context of this impact which provides the general scope and framework for development of these Appendices.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

12921/RG 1.7; 12950/RG 1.70; 12963/RG 1.72; 12989/RG 1.96; 13006/RG 1.143;  
13012/RG 1.151; 13085/NRC GENERIC LETTER 90-06; 13140/NRC GENERIC LETTER

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90-06; 13247/RG 1.26; 13255/RG 1.137; 18854/RG 1.26; 25549/RG 1.11; 25550/RG 1.141

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**Integrated Impact Number: 2      SRP Section Number: 3.2.2**

**Suggested Changes to the SRP Section:**

Incorporate Reg. Guide 1.96 staff positions related to quality group classification of BWR main steam isolation valve leakage control systems in SRP Section 3.2.2.

Reg. Guide 1.96 states that the main steam isolation valve leakage control system, and any necessary subsystems, should be designed in accordance with Quality Group B requirements with the exception of any portion of leakage control piping between inner and outer containment isolation valves which should be designed in accordance with Quality Group A requirements supplemented by Appendix A of Reg. Guide 1.96.

This classification information from Reg. Guide 1.96 should be considered for incorporation in Appendix D of SRP Section 3.2.2.

**Potential Impacts/Documents supporting the Suggested Changes:**

12989/RG 1.96

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**Integrated Impact Number: 3      SRP Section Number: 3.2.2**

**Suggested Changes to the SRP Section:**

Incorporate Reg. Guide 1.7 staff position for quality group classification of combustible gas control systems in SRP Section 3.2.2.

Reg. Guide 1.7 provides staff positions related to post LOCA containment combustible gas control. Reg. Guide 1.7 position C.3 states that group B quality standards should be applied to containment combustible gas control systems.

This classification information from Reg. Guide 1.7 should be considered for incorporation in Appendices C and D of SRP Section 3.2.2.

**Potential Impacts/Documents supporting the Suggested Changes:**

12921/RG 1.7

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**Integrated Impact Number:** 4      **SRP Section Number:** 3.2.2

#### **Suggested Changes to the SRP Section:**

Incorporate Reg. Guide 1.143 staff positions related to selection and application of codes and standards for radioactive waste management systems in SRP Section 3.2.2.

Reg. Guide 1.143 provides design guidance acceptable to the NRC staff related to quality assurance for radioactive waste management systems, structures, and components. This Reg. Guide identifies the codes and standards applicable to these systems and provides additional positions regarding the design of these systems. Circular 80-18 states that modifications to radwaste management systems should be evaluated against Reg. Guide 1.143. SRP Sections 11.2, 11.3, and 11.4 state that the quality group classification review of radioactive waste management systems will be conducted within the scope of the SRP Section 3.2.2 review.

Reg. Guide 1.143 does not take a position with regard to a specific quality group classification described in Reg. Guide 1.26. Instead, Reg. Guide 1.143 establishes positions with regard to codes and standards applicability to radioactive waste management systems that are not directly comparable to any of the Reg. Guide 1.26 quality group classification, however, the general codes and standards applicability outlined for design and fabrication of these systems appears to correspond most closely to the codes and standards applied for quality group D.

In the ABWR and CE System 80+ FSERs, the staff applied Reg. Guide 1.143 in its review of radioactive waste management systems. In the ABWR FSER, the staff stated findings with respect to seismic design criteria and quality group classifications meeting the applicable criteria and guidelines specified in Reg. Guide 1.143.

Consideration should be given to including radioactive waste management systems in SRP Section 3.2.2 Appendices C and D including an identification of Reg. Guide 1.143 which provides design and quality assurance positions specific to radwaste management systems.

In addition, IPD 7.0 Form # 3.2.2-3 recommends revision of Reg. Guide 1.143 to specify a quality group classification of D for radioactive waste management systems.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

13006/RG 1.143; 13025/NRC CIRCULAR 80-18; 25503/FINAL SER CE80 CH 11;  
25505/FINAL SER ABWR CH 11

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**Integrated Impact Number: 5      SRP Section Number: 3.2.2**

#### **Suggested Changes to the SRP Section:**

Incorporate Reg. Guide 1.137 staff positions related to quality group classification and the applicability of codes and standards to diesel fuel oil storage and transfer systems in SRP Section 3.2.2 Appendices C and D.

Reg. Guide 1.137 describes the NRC staff position for complying with regulations regarding the design of fuel-oil systems for standby diesel generators. The Reg. Guide states that ANSI N195-1976, "Fuel Oil Systems for Standby Diesel-Generators," provides an adequate basis for design of fuel oil systems for diesel generators that provide standby electrical power for a nuclear power plant subject to the regulatory positions specified in the guide. The diesel fuel oil system, which is identified in Review Procedures subsection as a system not identified in Reg. Guide 1.26 that is considered by the staff to be a quality group C system. Reg. Guide 1.137 provides positions with regard to the pressure boundary design of this system that are supplemental to those established by the quality group C designation.

Consideration should be given to identifying this Reg. Guide in SRP Section 3.2.2 Appendices C and D.

Consideration should also be given to revising Reg. Guide 1.137 to reflect a quality group C classification of emergency diesel generator fuel oil systems. This future work item is tracked by IPD 7.0 Form 3.2.2-2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

13255/RG 1.137

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**Integrated Impact Number: 7      SRP Section Number: 3.2.2**

#### **Suggested Changes to the SRP Section:**

Revise the Areas of Review and Review Procedures subsections to reflect the Reactor Coolant Pressure Boundary (RCPB) review of SRP Section 5.2.1.1.

10 CFR 50 Sections 50.2, 50.55a(c), and Criterion 30 of Appendix A provide definitions and requirements related to the RCPB which are relevant to the quality group classification of RCPB components and the establishment of appropriate quality requirements. A detailed review of RCPB quality group classification and code applicability is performed in SRP Section 5.2.1.1 primarily to determine RCPB compliance with the relevant requirements of 10 CFR 50.55a.

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SRP Section 3.2.2 does not clearly differentiate the RCPB classification and quality requirements review from the balance of fluid systems important to safety.

Consideration should be given to revising the SRP 3.2.2 subsections with the exception of Evaluation Findings (already reflects the review performed in 5.2.1.1) to reflect the RCPB review of SRP Section 5.2.1.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

12859/CODE OF FED. REGS 10CFR50; 12915/CODE OF FED. REGS 10CFR50;  
12917/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 12      SRP Section Number: 9.3.1**

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure that addresses the potential for overpressurization of air actuated components.

NRC Bulletin 80-25 identified a concern with the potential for BWR safety-relief malfunctions due to overpressurization by the pneumatic supply system. The Bulletin requested BWR applicants and licensees to review the potential for and magnitude of an overpressure condition. NRC Information Notice 88-24 observes that safety-related components that depend on the air system are designed to assume a fail safe condition on loss of air; however, the converse condition of air overpressurization may not always be considered. Such a condition could render the affected safety-related components inoperable. Generic Letter 88-14 requested licensees to verify that the design of the entire instrument air system including air or other pneumatic accumulators is in accordance with its intended function.

Consideration should be given to adding a Review Procedure that addresses the potential for overpressurization of air actuated components.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1351/NUREG 0933; 12869/NRC BULLETIN 80-25; 12929/NRC NOTICE 80-40;  
12986/NRC NOTICE 88-24; 24321/NRC GENERIC LETTER 88-14

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**Integrated Impact Number: 13      SRP Section Number: 9.3.1**

**Suggested Changes to the SRP Section:**

Modify Review Interfaces and Review Procedures as necessary to appropriately reflect the CAS review performed for compliance with GDC 4.

SRP Section 9.3.1, Revision 1 Review Procedure III.3 describes a review of compatibility with adverse environmental phenomena and protection against dynamic effects including certain pipe breaks. This review is related to determining compliance with GDC 4, however, GDC 4 is not an acceptance criterion for SRP Section 9.3.1. The appropriate review procedures and criteria for performing these reviews are contained in the Chapter 3 SRP Sections. SRP Section 9.3.1, Revision 1 also addresses the review for compatibility with adverse environmental phenomena and protection against dynamic effects as review interfaces in Areas of Review (e.g., see item I.5).

Consideration should be given to modifying the Review Procedures related to verification of compliance with GDC 4 and revising the existing review interfaces as necessary to clarify the relationship of the 9.3.1 review with the Chapter 3 SRP Sections.

**Potential Impacts/Documents supporting the Suggested Changes:**

18861/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 14      SRP Section Number: 9.3.1**

**Suggested Changes to the SRP Section:**

Add a Review Procedure to verify compliance with GDC 5.

GDC- 5, which concerns the sharing of structures, systems and components important to safety, is currently an Acceptance Criterion (II.3 ) for SRP Section 9.3.1, and has an Evaluation Finding (IV.3). However, there are currently no implementing procedures to verify compliance with this requirement.

Consideration should be given to developing a Review Procedure to verify compliance with GDC 5.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

12867/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 22      SRP Section Number: 9.3.1**

#### **Suggested Changes to the SRP Section:**

Add Review Procedures for non-radioactive compressed air systems that may become contaminated through connections to radiologically contaminated systems.

Permanent and temporary connections to contaminated systems can result in the compressed air system becoming contaminated. This has affected both offsite and worker exposures, and has the potential for an unreviewed safety question. Bulletin 80-10 requested a review of systems that are considered nonradioactive for their potential to become contaminated and identified the instrument air as a system that should receive particular consideration.

Consideration should be given to incorporating a review of the issue identified in Bulletin 80-10 into the SRP section Review Procedures.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14166/NRC NOTICE 85-06; 14184/NRC BULLETIN 80-10; 14189/NRC NOTICE 79-08; 24324/FINAL SER ABWR CH 9; 24325/FINAL SER CE80 CH 11

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**Integrated Impact Number: 23      SRP Section Number: 9.3.1**

#### **Suggested Changes to the SRP Section:**

Add Areas of Review and Review Procedures related to air quality of interconnected compressed air systems.

NUREG 0933, Generic Issue 43 identified concerns with the reliability compressed air systems that could affect the ability of safety-related components to accomplish their intended safety functions. As indicated in NUREG-1275 and Generic Letter 88-14 related to this generic issue, contamination (e.g., oil, particulate, water, etc.) of compressed air was identified as significant contributor to unreliability in safety-related equipment controlled or actuated by compressed air. Generic Letter 88-14 requested licensees to verify that actual instrument air quality is consistent with the manufacturer's recommendations for individual components served. Regulatory Guide 1.68.3, Regulatory Position C.9, provides guidance that plant equipment designed to be supplied by the instrument and control air system not be supplied by other compressed air supplies (such



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as service air) that may have less restrictive air quality requirements.

The SRP provides guidance regarding the quality of air in safety-related compressed air systems, however, does not specifically address the air quality of compressed air sources that may provide backup supply to the safety-related compressed air system.

Consideration should be given to revising the SRP Section 9.3.1 as necessary to address the review of compressed air quality for backup compressed air sources to safety-related compressed air systems.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1351/NUREG 0933; 12870/NRC CIRCULAR 81-14; 12933/NRC NOTICE 81-38;  
12971/NRC NOTICE 86-57; 24322/RG 1.68.3; 24323/NRC GENERIC LETTER 88-14

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**Integrated Impact Number: 31      SRP Section Number: 3.2.2**

#### **Suggested Changes to the SRP Section:**

Add classification information for instrument sensing lines to SRP Section 3.2.2 Review Procedures and Appendices.

Reg. Guide 1.151 states that the requirements of ISA-S67.02, 'Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants', provide an acceptable basis to the staff for the design and installation of safety-related instrument sensing lines in nuclear power plants subject to the regulatory positions specified in the guide. Reg. Guide 1.151 includes positions (C.2 and C.3) regarding ASME Code applicability to instrument lines based on connection to safety-related systems and the safety significance of the instrumentation. Although this information does not specifically address quality group classification, this information would be useful when performing the detailed reviews of piping and instrumentation diagrams described in the Review Procedures subsection.

In the CE System 80+ FSER, in conjunction with its review of system quality group classification, the staff confirmed that instrument-sensing lines and their supports will be constructed per RG 1.151, in particular, positions C.2 and C.3 relating to applicable seismic and ASME Code criteria.

Consideration should be given to including this information in SRP Section 3.2.2.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

13012/RG 1.151; 25480/FINAL SER CE80 CH 3

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**Integrated Impact Number: 38      SRP Section Number: 3.2.2**

#### **Suggested Changes to the SRP Section:**

Revise citation of ANSI/ASME B31.1-1973 to reflect the latest version of the standard.

Paragraph 136.4 of ANSI/ASME B31.1-1973 is referenced in Appendices A and B of SRP Section 3.2.2 as providing an approach acceptable to the staff for pressure boundary examination and acceptance for portions of BWR main steam and feedwater systems designated as quality group "D+QA." ANSI/ASME B31.1-1992 is the latest version of this standard. A formal comparison was performed between the two versions of Paragraph 136.4 of ANSI/ASME B31.1. The comparison concluded that the changes involved evolutionary improvements to the standard and that there was no safety significant differences with respect to SRP Section 3.2.2. Under the SRP-UDP, however, standards citations existing in the SRP are not updated (nor are new citations added to address update of standards citations in other regulatory documents cited in the SRP) until NRC review and acceptance of standard comparisons.

The comparison performed by PNL recommends update of the SRP Section 3.2.2 Appendix A and B citations to ANSI/ASME B31.1-1992.

Pending NRC review and acceptance of the standard comparison recommendations described above, consideration should be given to revising the references to ANSI/ASME B31.1 in Appendices A and B to reflect the 1992 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14923/C&S: ANSI B.31.1

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**Integrated Impact Number: 39      SRP Section Number: 3.2.2**

#### **Suggested Changes to the SRP Section:**

Revise citation of ANSI/ASME B31.1-1980 in SRP Section 3.2.2 References subsection VI such that the citation is not version specific.

ANSI/ASME B31.1 is cited without specific version information in SRP Section 3.2.2 as an acceptable standard for Quality Group D piping and valve components. ANSI/ASME

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B31.1-1980 is listed in SRP Section 3.2.2 References subsection VI . The latest version of this standard evaluated under the SRP-UDP is ANSI/ASME B31.1-1992. Since the citation of ANSI/ASME B31.1 is for the purpose of identifying applicability only to Quality Group D components, no formal comparison was performed on the basis of the limited potential safety significance of any differences between the two versions of the standard. Under the SRP-UDP, however, standards citations existing in the SRP are not updated (nor are new citations added to address update of standards citations in other regulatory documents cited in the SRP) without evidence of NRC review and acceptance of updated standards and NRC authorization to update such citations.

Pending NRC review and authorization, consideration should be given to revising SRP section 3.2.2 such that citation of ANSI/ASME B31.1 in subsection VI is not specific to a particular version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14924/C&S: ANSI B.31.1

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**Integrated Impact Number:** 40      **SRP Section Number:** 3.2.2

#### **Suggested Changes to the SRP Section:**

Revise citation of ANSI/ASME B96.1-1980 in SRP Section 3.2.2 References subsection VI such that the citation is not version specific.

ANSI/ASME B96.1 is cited without specific version information in SRP Section 3.2.2 as an acceptable standard for Quality Group D atmospheric storage tanks. ANSI/ASME B96.1-1980 is listed in SRP Section 3.2.2 References subsection VI . The latest version of this standard evaluated under the SRP-UDP is ANSI/ASME B96.1-1989. Since the citation of ANSI/ASME B96.1 is for the purpose of identifying applicability only to Quality Group D components, no formal comparison was performed on the basis of the limited potential safety significance of any differences between the two versions of the standard. Under the SRP-UDP, however, standards citations existing in the SRP are not updated (nor are new citations added to address update of standards citations in other regulatory documents cited in the SRP) without evidence of NRC review and acceptance of updated standards and NRC authorization to update such citations.

Pending NRC review and authorization, consideration should be given to revising SRP section 3.2.2 such that citation of ANSI/ASME B96.1 in subsection VI is not specific to a particular version.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

14918/C&S: ANSI B.96.1

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**Integrated Impact Number:** 42      **SRP Section Number:** 3.2.2

#### **Suggested Changes to the SRP Section:**

Revise citation of API 620, sixth edition, 1977 in SRP Section 3.2.2 References subsection VI such that the citation is not version specific.

API 620 is cited without specific version information in SRP Section 3.2.2 as an acceptable standard for Quality Group D storage tanks with an operating pressure of 0-15 psig. API 620, sixth edition, 1977 is listed in SRP Section 3.2.2 References subsection VI. The latest version of this standard evaluated under the SRP-UDP is API 620-1990. Since the citation of API 620 is for the purpose of identifying applicability only to Quality Group D components, no formal comparison was performed on the basis of the limited potential safety significance of any differences between the two versions of the standard. Under the SRP-UDP, however, standards citations existing in the SRP are not updated (nor are new citations added to address update of standards citations in other regulatory documents cited in the SRP) without evidence of NRC review and acceptance of updated standards and NRC authorization to update such citations.

Pending NRC review and authorization, consideration should be given to revising SRP section 3.2.2 such that citation of API 620 in subsection VI is not specific to a particular version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14911/C&S: API 620

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**Integrated Impact Number:** 43      **SRP Section Number:** 3.2.2

#### **Suggested Changes to the SRP Section:**

Revise citation of API 650, sixth edition, Revision 1, 1978 in SRP Section 3.2.2 References subsection VI such that the citation is not version specific.

API 650 is cited without specific version information in SRP Section 3.2.2 as an acceptable standard for Quality Group D atmospheric storage tanks. API 650, sixth edition, Revision 1, 1978 is listed in SRP Section 3.2.2 References subsection VI. The latest version of this standard evaluated under the SRP-UDP is API 650 -1988. Since the citation of API 650 is for the purpose of identifying applicability only to Quality Group D components, no formal

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comparison was performed on the basis of the limited potential safety significance of any differences between the two versions of the standard. Under the SRP-UDP, however, standards citations existing in the SRP are not updated (nor are new citations added to address update of standards citations in other regulatory documents cited in the SRP) without evidence of NRC review and acceptance of updated standards and NRC authorization to update such citations.

Pending NRC review and authorization, consideration should be given to revising SRP section 3.2.2 such that citation of API 650 in subsection VI is not specific to a particular version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14913/C&S: API 650

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<b>Integrated Impact Number:</b> 44	<b>SRP Section Number:</b> 3.2.2
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#### **Suggested Changes to the SRP Section:**

Revise citations of the ASME Boiler and Pressure Vessel Code (BPVC), Section III, 1980 edition in SRP Section 3.2.2 References subsection VI such that the citations are not version specific.

Applicable provisions of the ASME BPVC, Section III, Division 1 are cited without specific version information in SRP Section 3.2.2 as acceptable standards for Quality Groups A, B, and C components. Although the latest editions of the Code identified under the SRP-UDP are the 1992 and 1995 editions, the latest edition of the Code, Sections III and XI addressed under the SRP-UDP for the purposes of reference verification/update is the 1989 edition based upon the current requirements of 10 CFR 50.55a. The ASME BPVC, Section III is referenced in 10 CFR 50.55a and in 10 CFR 50 Appendix G. No code comparison was performed between cited and the latest editions (including addenda) of the Code on the basis that the NRC staff is actively involved in the development of the Code and has a high degree of familiarity with the Code and its evolution. The editions and addenda of the Code, Sections III and XI, which are acceptable to the NRC for any particular application are established (incorporated by reference) in 10 CFR 50.55a. Staff positions regarding specific Code cases are included in Reg. Guides which are routinely updated. Any changes in identification or titles of applicable Code Sections, articles, etc. in later editions of the Code than available during the last major revision of the SRP (July 1981) are addressed under the SRP-UDP as reference verification items.

Consideration should be given to revising SRP Section 3.2.2 such that citation of the ASME BPVC, Section III in subsection VI is not specific to a particular edition.

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14925/C&S: ASME III; 14927/C&S: ASME XI; 19082/C&S: ASME III; 19083/C&S: ASME III

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**Integrated Impact Number: 45      SRP Section Number: 3.2.2**

#### **Suggested Changes to the SRP Section:**

Revise citations of the ASME Boiler and Pressure Vessel Code, Section VIII such that the SRP references are not version specific.

ASME Section VIII-1980, is referenced in the SRP as an acceptable code for Quality Group D pressure vessels. The current version of this code is ASME Section VIII-1995. Since this citation is limited to Quality Group D components, no formal comparison was performed on the basis of the limited potential safety significance of any differences between the two versions of the code. Under the SRP-UDP, code/standard citations existing in the SRP are not updated (nor are new citations added to address update of code/standard citations in other regulatory documents cited in the SRP) without evidence of NRC review and acceptance of the updated code/standard and NRC authorization to update such citations.

Pending NRC review and authorization, consideration should be given to revising SRP section 3.2.2 such that citation of ASME Section VIII-1980 are not specific to a particular version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

19096/C&S: ASME VIII

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**Integrated Impact Number: 60      SRP Section Number: 3.2.2**

#### **Suggested Changes to the SRP Section:**

Revise citation of AWWA D100-1979 in SRP Section 3.2.2 References subsection VI such that the citation is not version specific.

AWWA D100 is cited without specific version information in SRP Section 3.2.2 as an acceptable standard for Quality Group D atmospheric storage tanks. AWWA D100-1979 is listed in SRP Section 3.2.2 References subsection VI. The latest version of this standard evaluated under the SRP-UDP is AWWA D100-1984. Since the citation of AWWA D100 is for the purpose of identifying applicability only to Quality Group D components, no formal comparison was performed on the basis of the limited potential safety significance of any

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differences between the two versions of the standard. Under the SRP-UDP, however, standards citations existing in the SRP are not updated (nor are new citations added to address update of standards citations in other regulatory documents cited in the SRP) without evidence of NRC review and acceptance of updated standards and NRC authorization to update such citations.

Pending NRC review and authorization, consideration should be given to revising SRP section 3.2.2 such that citation of AWWA D100 in subsection VI is not specific to a particular version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14915/C&S: AWWA D100

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**Integrated Impact Number: 65      SRP Section Number: 9.3.1**

#### **Suggested Changes to the SRP Section:**

Add Acceptance Criteria and Review Procedures for station blackout.

10 CFR Part 50.63 requires that plants licensed to operate must be able to withstand for a specified duration and recover from a station blackout. Regulatory Guide 1.155 describes a method acceptable to the NRC staff for complying with the regulation. The SRP currently does not provide for review of this issue. Certain positions in C.3.2, C.3.3, and C.3.5 would apply to portions of the compressed air system necessary to provide core cooling and decay heat removal or maintain containment integrity following a station blackout. A review of the SBO event and the applicant's station blackout analysis is the subject of new SRP Section 8.4 developed under the SRP-UDP. A review of the compressed air system conformance with Regulatory Guide 1.155 positions, as applicable, would be coordinated with the review proposed for SRP Section 8.4.

Consider citing 10 CFR 50.63 and Regulatory Guide 1.155 requirements and guidance in Acceptance Criteria and developing a Review Procedure that coordinates the compressed air system review with the SBO review of SRP Section 8.4, as appropriate.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

13015/RG 1.155; 13016/RG 1.155; 24326/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 73      SRP Section Number: 9.3.1**

#### **Suggested Changes to the SRP Section:**

Add Areas of Review and Review Procedures related to Generic Letter 88-14 concerning the improvement of compressed air system reliability.

Numerous regulatory documents identify specific and general concerns regarding the reliability compressed air systems and equipment controlled or operated by compressed air. Generic Issue 43 was established to evaluate concerns regarding desiccant contamination of instrument air lines and was later broadened to include all causes of compressed air system unreliability. NUREG-1275 documents the staff's evaluation of Generic Issue 43. NUREG-1275 indicated that the performance of the air-operated safety-related components may not be in accordance with their intended safety function because of inadequacies in the design, installation, and maintenance of the instrument air system and recommendations were made to address these findings. This issue was resolved and Generic Letter 88-14 was issued that requested licensees and applicants to review NUREG-1275 and perform a design and operations verification of instrument air systems.

Consideration should be given to revising the SRP Section 9.3.1 as necessary to address the review of compressed air systems consistent with the positions indicated in Generic Letter 88-14.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1351/NUREG 0933; 12868/NRC BULLETIN 80-01; 12873/NRC GENERIC LETTER 88-14; 12918/NRC GENERIC LETTER 89-18; 12927/NRC NOTICE 80-30; 12939/NRC NOTICE 85-35; 12986/NRC NOTICE 88-24; 14772/NUREG 0737; 18862/NUREG 0737; 19059/FINAL SER EPRI CH 9; 19064/FINAL SER EPRI CH 9; 19066/CODE OF FED. REGS 10CFR50; 19067/CODE OF FED. REGS 10CFR50; 22498/NRC GENERIC LETTER 80-04; 24328/FINAL SER ABWR CH 9; 24329/FINAL SER CE80 CH 20

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**Integrated Impact Number: 80      SRP Section Number: 3.2.2**

#### **Suggested Changes to the SRP Section:**

Revise SRP to address the review of non-pressure-retaining component and structure classification.

The ABWR, CE System 80+, and EPRI Evolutionary Plant FSERs and the SRP acknowledge the use of the ANS safety classification methodology as an alternative to Reg. Guide 1.26. The



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FSERs explicitly state that the staff has not endorsed ANSI/ANS 51.1/52.1 and that the review of evolutionary plant applications is not dependant on such an endorsement.

Using the ANS methodology, all structures, systems, and components (SSCs) are classified into one of three safety classifications or into a non-safety classification. Applicable design and construction codes, standards, and regulatory requirements are identified based on these classifications. Although SRP Section 3.2.2 currently acknowledges that an applicant might use the ANS classification methodology, it does not provide acceptance criteria or procedures for reviewing the classification of non-pressure-retaining SSCs. The ABWR and CE System 80+ FSERs indicate that non-pressure-retaining components were reviewed by evaluating quality assurance in accordance with 10 CFR 50, Appendix B and seismic classification.

In the CE System 80+ FSER, in conjunction with its review of system quality group classification, the staff questioned the applicant's safety classification of new and spent fuel racks as non-nuclear-safety and the corresponding lack of quality assurance requirements to be applied. The applicant reclassified the fuel racks as ANS Safety Class 3 and thus subject to the quality assurance requirements of 10 CFR 50, Appendix B.

In the ABWR FSER, in conjunction with its review of system quality group classification, the staff also questioned the applicant's quality assurance requirements for new and spent fuel racks (and storage container for defective fuel). The applicant demonstrated that these components will be subject to the quality assurance requirements of 10 CFR 50, Appendix B commensurate with the importance of their function to safety.

The CE System 80+ and ABWR FSERs also describe staff acceptance of the design of numerous SSCs and related discussions of the ANS safety classifications and quality assurance requirements applied to SSCs for which a quality group classification is not required under current regulatory guidance. Examples of items for which safety/quality classifications were apparently accepted include buildings and structures, mechanical equipment such as cranes and fuel handling platform equipment, electric modules and cable performing safety-related functions, Class 1E electric equipment, piping supports and appurtenances, safety-related instrumentation, sump pumps and instrumentation associated with flood protection of safety-related areas, safety-related HVAC equipment and ductwork, electric heaters, fission product adsorption/filtration equipment, essential chilled water system components, cavity flooding system components, and diverse, non-safety-related reactor coolant pump seal injection components.

Consideration should be given to revising SRP Section 3.2.2 to address the review process and acceptance criteria of non-pressure-retaining components and structures and to indicate that Reg. Guide 1.26 quality group classifications are not applicable to those components. Revision of the SRP to address review of the classification of non-pressure retaining components and structures

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will be tracked as a candidate for future work by IPD 7.0 Form 3.2.2-4.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18945/FINAL SER EPRI CH 1; 20437/FINAL SER ABWR CH 3; 20438/FINAL SER ABWR CH 3; 25477/FINAL SER CE80 CH 3; 25479/FINAL SER CE80 CH 3; 25489/FINAL SER CE80 CH 6; 25490/FINAL SER CE80 CH 6; 25494/FINAL SER CE80 CH 9; 25495/FINAL SER CE80 CH 9; 25498/FINAL SER CE80 CH 9; 25500/FINAL SER CE80 CH 9; 25507/FINAL SER ABWR CH 20; 25518/FINAL SER ABWR CH 9; 25520/FINAL SER ABWR CH 9; 25534/FINAL SER ABWR CH 3; 25535/FINAL SER ABWR CH 3; 25541/FINAL SER ABWR CH 3

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**Integrated Impact Number:** 82      **SRP Section Number:** 3.2.1

#### **Suggested Changes to the SRP Section:**

Develop Technical Rationale for use of existing guidance document. RG 1.29 provides guidance for seismic classification of structures, systems and components and is currently cited in the SRP. Consider including technical rationale for use of RG 1.29 to determine compliance with GDC 2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

10607/CODE OF FED. REGS 10CFR100; 10707/RG 1.29

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**Integrated Impact Number:** 83      **SRP Section Number:** 3.2.1

#### **Suggested Changes to the SRP Section:**

Include an Acceptance Criterion for confirmation that seismic category items are within the scope of a QA Program. GDC-1 requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. An appropriate review procedure and guidance (RG 1.29) currently exist in this section to verify compliance with GDC-1, however no corresponding acceptance criterion is indicated. Consider incorporating GDC-1 and Appendix B to 10 CFR Part 50 as acceptance criteria, developing corresponding technical rationale for use of RG 1.29, and providing evaluation findings to establish that those structures, systems and components classified as seismic category I are within the scope of the applicant's QA Program.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

10707/RG 1.29; 21382/CODE OF FED. REGS 10CFR50; 21626/FINAL SER EPRI CH 1; 21634/NUREG 0800; 21893/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 84      SRP Section Number: 3.2.1**

#### **Suggested Changes to the SRP Section:**

Add Acceptance Criteria for existing and new guidance documents to identify non-seismic Category I structures, systems and components that must be designed and/or analyzed to seismic criteria. This would primarily involve radwaste management systems and items that could impact safety-related functions.

GDC-2 requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. GDC-61 requires that radwaste management systems be designed to assure adequate safety under normal and postulated accident conditions. Numerous documents, including RG 1.29, RG 1.143, the Staff's SER for the ABWR, and Generic Letter 86-10, provide examples of structures, systems and components that, although not seismic Category I, should be assured to maintain integrity under seismic conditions.

Consider adding GDC-61 as an acceptance criterion, listing RGs 1.29 and 1.143 as guidance documents, and providing review procedures and evaluation findings to cover seismic classification of systems, structures and components that are not required to remain functional following seismic events, but must be designed to not fail during specified seismic occurrences.

This integrated impact is referenced in IPD 7.0 Future Regulatory Action Needs forms 3.2.1-1 and 3.2.1-2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

10600/CODE OF FED. REGS 10CFR50; 10707/RG 1.29; 10898/RG 1.143; 10988/NRC GENERIC LETTER 86-10; 21621/FINAL SER ABWR CH 3; 21626/FINAL SER EPRI CH 1; 23014/SECY 93-087

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**Integrated Impact Number: 86      SRP Section Number: 3.2.1**

#### **Suggested Changes to the SRP Section:**

Identify guidance for classification, as seismic Category I, systems that contain or could contain

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radioactive material and whose failure could result in offsite exposures greater than 0.5 Rem.

The current SRP identifies three functions to be classed as safety-related: 1) integrity of the reactor coolant pressure boundary; 2) capability to achieve and maintain shutdown of the reactor; and, 3) capability to prevent or mitigate accidents with exposures exceeding the offsite dose limits of 10 CFR 100 (25 Rem). Existing guidance document, RG 1.29, includes an additional criterion for designation of systems as safety-related; systems that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses that are more than 0.5 rem to the whole body or its equivalent to any part of the body.

Consider revising the acceptance criteria subsection to indicate that, as specified in RG 1.29, those systems that could result in offsite doses greater than 0.5 rem should be classified as seismic Category I. Also consider listing RG 1.29 as the guidance document, adding technical rationale and revising the evaluation findings to address systems whose failure could result in offsite exposures greater than 0.5 Rem.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

10673/RG 1.26; 10707/RG 1.29

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**Integrated Impact Number:** 87      **SRP Section Number:** 3.6.2

#### **Suggested Changes to the SRP Section:**

Modify acceptance criteria to reflect an alternate approach to compliance with GDC-4. GDC-4 requires that structures, systems, and components important to safety be appropriately protected against dynamic effects, including the effects of pipe whipping and discharging fluids that may result from equipment failures. GDC-4 was modified in 1987 allowing dynamic effects associated with postulated pipe ruptures to be excluded from the plant design basis when analyses, referred to as "leak-before-break," demonstrate that the probability of fluid system piping rupture is extremely low. SRP 3.6.2 does not currently provide for application of such leak-before-break (LBB) criteria.

Consider revising the Acceptance Criteria and modifying the Review Procedures to acknowledge an alternate approach to meeting the requirements of GDC-4.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21441/NRC GENERIC LETTER 84-04; 21686/NUREG 1061 VOL 3; 21699/FINAL SER EPRI CH 1; 21708/DRAFT FINAL SER ABWR CH 3; 21721/CODE OF FED. REGS

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10CFR50; 21786/FINAL SER EPRI CH 1; 21834/NRC POLICY STATEMENT 54 FR 18649; 21835/DRAFT SER CE80 CH 3; 21836/DRAFT SER CE80 CH 3; 21837/DRAFT SER CE80 CH 3; 21839/DRAFT SER CE80 CH 3

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**Integrated Impact Number:** 89      **SRP Section Number:** 3.6.2

#### **Suggested Changes to the SRP Section:**

Replace existing Branch Technical Position.

GDC-4 requires that structures, systems, and components important to safety be appropriately protected against dynamic effects, including the effects of pipe whipping and discharging fluids that may result from equipment failures. BTP MEB 3-1, Revision 1, attached to SRP 3.6.2, provides guidance regarding determination of pipe rupture locations, including arbitrary locations in intermediate piping of high-energy systems. The NRC's Piping Review Committee recommended (in NUREG 1061, Vols. 3 & 5) that requirements in SRP 3.6.2 for arbitrary intermediate pipe breaks be relaxed. Generic Letter 87-11 was issued in response, transmitting BTP MEB 3-1, Revision 2, which relaxes the requirement to postulate and design for arbitrary pipe breaks in intermediate piping locations.

Consider replacing BTP MEB 3-1, Revision 1, with MEB 3-1, Revision 2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

15572/NUREG 0933; 20446/NRC GENERIC LETTER 87-11; 21689/NUREG 1061 VOL 3; 24578/FINAL SER CE80 CH 3; 24579/FINAL SER ABWR CH 3

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**Integrated Impact Number:** 93      **SRP Section Number:** 3.2.1

#### **Suggested Changes to the SRP Section:**

Add a review procedure for the seismic classification of certain BWR SSCs.

Current BWRs include a MSIV leakage control system to mitigate the radiological consequences of MSIV leakage following a DBA LOCA. EPRI and GE have proposed taking credit for fission product retention in main steam piping and the condenser as an alternative approach to fission product control following postulated accidents for the ABWR. In the EPRI FSER and the ABWR FSER the staff indicated that this approach is acceptable in principle, however to take credit for main steam system and condenser fission product retention, these components must be appropriately classified and shown to maintain their integrity following an SSE. Specific positions with regard to the seismic classification of these items are contained in the

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ABWR FSER. SECY 93-087 and its associated SRM provide additional guidance and staff positions on the classification of main steamlines in boiling water reactors.

Consider adding review procedures for the seismic classification of main steam piping, condenser, and turbine building as indicated in the ABWR FSER.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21621/FINAL SER ABWR CH 3; 21829/FINAL SER ABWR CH 3; 21830/FINAL SER EPRI CH 1; 23013/SECY 93-087

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**Integrated Impact Number:** 94      **SRP Section Number:** 3.2.1

#### **Suggested Changes to the SRP Section:**

Add a review procedure for analysis of piping beyond seismic Category I boundaries.

Currently, SRP 3.2.1 cites RG 1.29 as guidance for demonstrating compliance with GDC 2. RG 1.29 Regulatory Position C.3 states that seismic Category I design requirements should extend to the first seismic restraint beyond the defined boundaries. The ABWR FSER interprets the following statement to comply with RG 1.29: At the interface between seismic and non-seismic Category I piping systems, the seismic Category I dynamic analysis will be extended to either the first anchor point in the non-seismic system or to sufficient distance in the non-seismic system so as not to degrade the validity of the seismic Category I analysis.

Consider adding a review procedure that describes the application of seismic analysis at the interface between seismic Category I and non-seismic system boundaries.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21829/FINAL SER ABWR CH 3

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**Integrated Impact Number:** 95      **SRP Section Number:** 3.2.2

#### **Suggested Changes to the SRP Section:**

Add classification information for PWR pressurizer power operated relief valves (PORVs), block valves, and associated component equipment (e.g. PORV valve operators) to SRP Section 3.2.2 Review Procedures and Appendix C.

SRP Section 3.2.2 Review Procedures do not presently identify PORVs and associated block

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valves as important to safety.

Generic Letter 90-06 (resolution of Generic Issue 70), NUREG 0933 Generic Issue 70, and other NUREGs (1316, 1326, CR-4692, CF-5186) discuss the reliability and safety significance of PORVs and PORV block valves due to operational experience and the evolving safety-related role (SGTR mitigation, LTOPs, plant cooldown) of PORVs. Although Generic Issue 70 was resolved without requiring existing plants to upgrade PORV and block valves to full safety-grade requirements, NUREG 0933 and Generic Letter 90-06 indicate that this position should be reconsidered for new plants if safety functions are performed by these valves. Generic Letter 90-06 indicates that the staff has required a safety-grade classification for these valves in more recent licensing cases. NUREG 0933 states that recommended SRP changes have been transmitted to NRR with regard to the classification of these valves. An internal NRC Memorandum dated November 16, 1989 identifies which PWR applications should classify PORVs and related components as safety-related.

Consideration should be given to incorporating staff positions related to PORV and PORV block valve classification in Review Procedures and Appendix C of SRP Section 3.2.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

13085/NRC GENERIC LETTER 90-06; 13140/NRC GENERIC LETTER 90-06;  
14106/NUREG 1316; 14107/NUREG 1326; 14829/NUREG 0933; 18863/NUREG CR-5186;  
18864/NUREG CR-4692; 25588/NRC MEMORANDUM 11/16/89, from E.S. Beckjord to F.P. Gillespie

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**Integrated Impact Number:** 100      **SRP Section Number:** 3.7.2

#### **Suggested Changes to the SRP Section:**

Provide a review procedure to consider effects of concrete cracking on seismic analysis of seismic Category I structures.

Although small cracks in concrete structures are generally irrelevant in regard to carrying capacity of the structure, its dynamic response (frequency signature) may be altered by the cracks, because shear stiffness and friction coefficient might be affected. The CE System 80+ FSER documents a staff position that the applicant should address the effects of concrete cracking on seismic analysis of all seismic Category I structures.

Consider adding a review procedure to check if seismic analysis of seismic Category I structures includes the effect, where appropriate, of concrete cracking.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

21876/FINAL SER CE80 CH 3

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**Integrated Impact Number:** 101      **SRP Section Number:** 3.7.2

#### **Suggested Changes to the SRP Section:**

Revise section to provide for seismic analysis of certain non-Category I structures, systems, and components.

SRP 3.7.2 subsection II. Acceptance Criteria is primarily worded to apply to seismic Category I structures (Acceptance Criteria II.8 discusses interaction between seismic Category I and non-seismic Category I structures). RG 1.29 and other regulatory documents require that some non-seismic Category I structures be designed and constructed to withstand the SSE. In the EPRI Evolutionary Plant FSER, the staff concluded that a commitment to analyze non-seismic Category I structures and systems whose structural failure or interaction could degrade the functioning of a seismic Category I structure, system, or component, or could result in incapacitating injury to occupants of the control room using the same type of dynamic seismic analysis methodology as that used for seismic Category I structures and systems was consistent with the SRP. In the ABWR FSER, the staff established a position that the non-seismic Category I turbine building, condenser anchorages and main steam piping (relied upon for retention of fission products) be subjected to dynamic seismic analysis for the SSE. A staff position in the CE System 80+ FSER requires that the applicant specify analytical methods and the design criteria to be applied to non-safety-related structures to ensure protection of adjacent safety-related structures.

Consider revising Areas of Review, Acceptance Criteria and Review Procedures, where appropriate, to clarify that the review described in the SRP is not limited in application to Category I structures, systems, and components.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21627/FINAL SER EPRI CH 1; 21832/DRAFT FINAL SER ABWR CH 3; 21858/RG 1.29; 21891/FINAL SER CE80 CH 3; 25562/FINAL SER ABWR CH 3

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**Integrated Impact Number:** 102      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Add Review Procedures and Areas of Review to address intersystem LOCAs.

GSI 105 addressed Intersystem LOCAs at BWRs and was expanded to include PWRs by incorporating GSI 96. NUREG-933 recommended resolution of GSI 105 be pursued with a high priority. For current plants this issue is being addressed under individual plant examinations (IPE) and no new requirements specific to GSI-105 were established. Staff positions on this issue for evolutionary plants are found in Chapter 1, Annex A of the EPRI Evolutionary Plant FSER which contains documents related to SECY-90-016.

SECY-90-016 specifies the staff's position on protection against the possibility of LOCA occurring outside the containment for those systems linked to the RCS. In summary, the staff position is that evolutionary light-water reactor designs should reduce the possibility of LOCA outside the containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an ultimate rupture strength (URS) at least equal to full RCS pressure.

Consider revising Review Procedures and associated Areas of Review to incorporate staff positions on intersystem LOCAs.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

15414/NUREG 0933; 15469/NUREG 0933; 21815/FINAL SER EPRI CH 5; 23023/SECY 90-016; 25460/FINAL SER CE80 CH 5; 25461/FINAL SER ABWR CH 20; 25560/STAFF REQ. MEMO 9007160185; 25561/SECY 93-087

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**Integrated Impact Number:** 103      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Develop areas of review and review procedures to address staff guidance for mid-loop operations.

GSI 99 tracked a safety concern that decay heat removal could be lost in PWRs operating at reduced coolant inventory, referred to as "mid-loop operations". Mid-loop operation is not specifically covered in the current Code of Federal Regulations. Implementation of the mid-loop staff positions is intended to ensure that plants meet requirements associated with residual heat removal such as GDC 34, which is currently an acceptance criteria for SRP section 5.4.7.

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The staff positions related to resolution of this issue for current reactors are described in Generic Letter 88-17. Staff positions on this issue for evolutionary plants are found in Chapter 1, Annex A of the EPRI Evolutionary Plant FSER which contains documents related to SECY-90-016. These documents describe additional positions with regard to mid-loop operations for evolutionary reactors that exceed staff licensing practices for current reactors.

Consideration should be given to adding an area of review and review procedures for implementing staff positions related to mid-loop operations. Additionally, consideration should be given to developing review procedures that reflect positions related to mid-loop operations for evolutionary reactors.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11130/NRC GENERIC LETTER 87-12; 11131/NRC GENERIC LETTER 88-17;  
21828/FINAL SER EPRI CH 5; 22199/NUREG 0933; 23022/SECY 90-016; 25471/FINAL  
SER CE80 CH 5; 25559/STAFF REQ. MEMO 9007160185

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**Integrated Impact Number:** 105      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Generic Letters 87-12 and 88-17 describe concerns and staff positions with regard to DHR reliability particularly during reduced inventory operations. In the ABWR DFSE and CE 80+ DSE the staff requested that shutdown studies be performed that; (1) demonstrate adequate assessment of shutdown and low-power risk, identifying design-specific vulnerabilities and weaknesses, and (2) document consideration and incorporation of design features which minimize shutdown and low power vulnerabilities. NUREG 1449 has been issued which documents the staff's evaluation of the shutdown and low power issue. The staff indicated in the CE 80+ DSE that NUREG 1449 would be used as guidance in the review of the applicants shutdown study. The ABWR FSE and EPRI FSE cite this NUREG with respect to shutdown and low power operations. The staff also requested that these shutdown studies include a description of the applicants conformance to the positions described in Generic Letter 88-17 and justification for exceptions taken to these positions. SECY-93-087 (and referenced documents) summarizes the staff's position with regard to the mid-loop operations safety issue and also with regard to the larger issue of shutdown and low power operations.

To track the broader issue of generic requirements and guidance covering shutdown and low-power operations not directly applicable to SRP section 5.4.7, IPD-7.0 form number 5.4.7-1 has been initiated. Consideration should be given to developing a separate SRP section to address generic issues covering shutdown and low-power operations and the associated regulatory requirements and guidance.

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Certain positions in the documents indicated above may have specific applicability to the review described in section 5.4.7 of the SRP. Consideration should be given to performing a detailed evaluation of these documents to determine the extent to which these positions may be appropriate for consideration in SRP Section 5.4.7.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11120/NRC BULLETIN 80-12; 11130/NRC GENERIC LETTER 87-12; 11131/NRC GENERIC LETTER 88-17; 21905/FINAL SER EPRI CH 5; 25471/FINAL SER CE80 CH 5; 25472/FINAL SER ABWR CH 19; 25558/SECY 93-087

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**Integrated Impact Number:** 108      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to address RHR Auto-closure Interlock issues.

GSI 99 tracked and resolved a safety concern addressing the inadvertent closure of the RHR suction isolation valves when the RHR system is in operation. The initial auto-closure interlock related mode of failure was broadened to include the less frequent but higher risk mode of failure associated with mid-loop operations. The resolution of GSI 99 is contained in Generic Letter 88-17. In addition to the requirements for improved procedures and administrative controls, the removal of the auto-closure interlock was recommended.

To address the concerns of GSI 99 the CE80+ design included instrumentation consistent with the guidance in Generic Letter 88-17, in addition, the suction isolation valves do not have an auto-closure interlock.

Consider adding a Review Procedure and referencing Generic Letter 88-17 to address current staff guidance on the suction isolation valve auto-closure interlock.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11131/NRC GENERIC LETTER 88-17; 11322/NRC GENERIC LETTER 90-04; 15435/NUREG 0933; 25515/FINAL SER CE80 CH 5; 25516/FINAL SER CE80 CH 20

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**Integrated Impact Number:** 109      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Incorporate changes resulting from resolution of Generic Issue 70.

Branch Technical Position (BTP) RSB 5-1 (attached to section 5.4.7) establishes functional requirements for plant cooldown to cold shutdown using safety-grade systems. Generic Issue 70 was established to consider the reliability of PWR Power Operated Relief Valves (PORVs) and PORV Block Valves because these non-safety-grade components were relied upon to mitigate a design basis accident, including plant cooldown following a steam generator tube rupture. Generic Letter 90-06 forwarded staff positions as resolution of Generic Issue 70. NUREG-0933 indicates that the staff's resolution of Generic Issue 70 included development of a proposed revision to SRP 5.4.7. Additional information can be found in Memorandum for F. Gillespie from E. Beckjord, "Resolutions of Generic Issue 70, 'Power Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light Water Reactors,'" November 16, 1989.

Consider incorporating the revision to SRP 5.4.7 developed during resolution of Generic Issue 70.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11158/NRC GENERIC LETTER 90-06; 14934/NUREG 0933; 25618/NRC GENERIC LETTER 92-02; 25619/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 110      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Revise the Review Procedures to address proper design of the miniflow system required to ensure RHR pump protection.

One of the purposes of NRC Bulletin 88-04 was to ensure that operating plants did not have safety related centrifugal pumps which could interact during parallel pump operation under miniflow conditions and result in dead headed operation of the weaker pump. NRC Bulletin 86-01 identified and requested corrective action for a design deficiency in BWRs involving a single failure under certain accident sequences which could result in all RHR minimum flow bypass valves being signaled to close while all other pump discharge valves are also closed. Review Procedure step 4 only addresses loss of miniflow during pump testing.

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Consideration should be given to revising the Review Procedures to include the miniflow design considerations addressed by NRC Bulletins 88-04 and 86-01.

On further analysis it has been determined that Generic Letter 89-04 also covers related concerns on miniflow issues and has been added a related reference document.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11308/NRC BULLETIN 86-01; 11310/NRC BULLETIN 88-04; 25607/NRC GENERIC LETTER 89-04

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**Integrated Impact Number:** 111      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Add an Area of Review (review interface) to address the coordination and review of the RHR suction intake in the containment sump. USI A-43 "Containment Emergency Sump Performance" was resolved when the staff issued a revision to SRP section 6.2.2 and Regulatory Guide 1.82 to reflect the staff's technical findings reported in NUREG 0897. The primary safety concern of this issue was potential debris blockage and recirculation failure due to inadequate NPSH caused by vortex formation. Consideration should be given to writing an Area of Review (review interface) to SRP section 6.2.2 which contains the requirements and guidance for proper design of the containment sump and suction intakes.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

6488/NUREG 0933; 7271/NUREG 0933; 8023/NUREG 0933

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**Integrated Impact Number:** 112      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to ensure that the RHR system design has provisions to ensure that thermal stratification and thermal stresses, which may occur in the unisolable portions of piping connected to the RCS, are properly accounted for.

Thermal stratification and the resultant thermal stresses, which can occur when cold water leaks into or hot water leaks out of the RCS, can cause premature failure of the unisolable sections of connected piping subject to the stresses. NRC Bulletin 88-08 provides staff positions and guidance directed at identifying, preventing and correcting this problem. The ABWR FSER requested that applicants review their design in accordance with NRC Bulletin 88-08 to

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determine if any sections of unisolable piping connected to the RCS could be subject to unacceptable thermal stresses. In the FSER for the ABB-CE System 80+ the staff requested, in accordance with NRC Bulletin 88-08, that licensees and applicants review systems connected to the RCS to determine whether any sections of such piping that cannot be isolated can be subject to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves.

Consideration should be given to adding a Review Procedure to address provisions in the RHR system design to account for potential thermal stratification and resultant thermal stresses.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

19837/NRC BULLETIN 88-08 Sup 3; 21917/NRC BULLETIN 88-08; 25539/FINAL SER ABWR CH 3; 25540/FINAL SER CE80 CH 3

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**Integrated Impact Number:** 113      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Add an Area of Review (review interface) to address the coordination and review of the capability of the RHR system to provide core cooling and decay heat removal following a station blackout. Regulatory Guide 1.155 describes a means acceptable to the NRC staff for implementing the requirements of 10 CFR 50.63.

Consideration should be given to writing an Area of Review (review interface) which will direct the reviewer to SRP Section 8.4 which will contain guidance on meeting the station blackout requirements.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

19955/RG 1.155

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**Integrated Impact Number:** 115      **SRP Section Number:** 3.7.1

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria, Areas of Review and Review Procedures to be consistent with current regulatory position on Operating Basis Earthquakes (OBE).

Design response spectra, time histories, and damping values are reviewed in SRP 3.7.1 and accepted for both OBE and SSE. At present, 10 CFR Part 100 Appendix A and SRP 2.5.2

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require the magnitude of the OBE to be at least 1/2 SSE. Industry experience has shown such an OBE controls the design of some safety systems. The staff agreed in SECY-90-016, Section IV.A, that the OBE should not control safety system design. Additionally, the EPRI Evolutionary Plant FSER documents a staff position indicating that elimination of the OBE is acceptable under certain circumstances.

As discussed in the ABWR and CE 80+ FSERs and in SECY 93-087, evolutionary plants may eliminate the OBE analysis from the design basis for structures, systems and components (SSCs). In such cases, the OBE will then serve as an inspection level earthquake above which the licensee would shut down the plant and inspect for potential damage to SSCs important to safety. Design certification based upon this single-earthquake design approach is predicated, in part, on the ability to ascertain the need to take action to shut down the plant in case of occurrence of the OBE, and to evaluate the performance of SSCs important to safety. In the ABWR and CE80+ FSERs, the NRC staff reviewed criteria developed by the Electric Power Research Institute (EPRI) for evaluating whether or not the plant needs to be shut down following an earthquake.

Consider changes to the Acceptance Criteria, Areas of Review and Review Procedures to reflect that the staff will consider an applicant's proposal to eliminate or reduce the OBE on a case-by-case basis, as an exemption to 10 CFR 100.

NOTE: The staff's review of the ABWR and the CE80+ designs appears to be based largely upon approaches described in the proposed Appendix S to 10 CFR 50 and on several draft Regulatory Guides. Pending final rulemaking and issuance of the Regulatory Guides for use, the changes proposed above could potentially be considered a Type II change. On this basis, this impact will not be processed further at this time. (Also see Integrated Impact # 1137 and Integrated Impact # 1221.)

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11153/CODE OF FED. REGS 10CFR100; 11160/CODE OF FED. REGS 10CFR100;  
15754/NUREG 0933; 22050/FINAL SER EPRI CH 1; 22053/FINAL SER CE80 CH 2;  
22054/FINAL SER ABWR CH 3; 22055/FINAL SER ABWR CH 3; 22323/SECY 90-016;  
22325/SECY 93-087; 23019/SECY 93-087; 23020/SECY 93-087

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**Integrated Impact Number: 116      SRP Section Number: 3.7.1**

#### **Suggested Changes to the SRP Section:**

Revise Areas of Review and Acceptance Criteria for allowable damping values.

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SRP Section 3.7.1 reviews overall critical damping values using Regulatory Guide 1.61 as guidance. The EPRI Evolutionary Plant FSER, the ABWR FSER and the CE80+ FSER all allow alternate damping values derived from the Regulatory Guide 1.84 conditional acceptance of ASME Code Case N-411-1. This code case allows damping values which are somewhat less restrictive than the values provided by Appendix N to Section III, Division 1 of the ASME Code or by Regulatory Guide 1.61.

Modify the Areas of Review and Acceptance Criteria to accept ASME Code Case N-411-1, as conditioned by Regulatory Guide 1.84, as a source of damping values.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

15575/NUREG 0933; 22052/FINAL SER EPRI CH 1; 22058/C&S: ASME N-411;  
22091/FINAL SER ABWR CH 3; 22092/FINAL SER CE80 CH 3

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**Integrated Impact Number:** 118      **SRP Section Number:** 6.7

#### **Suggested Changes to the SRP Section:**

Revise subsections of the SRP as necessary to address BWR designs that do not have a Main Steam Isolation Valve Leakage Control System (MSIVLCS). The SRP currently identifies Regulatory Guide 1.96 as the principal source of guidance with regard to the control of MSIV leakage in BWRs following a LOCA. The Regulatory Guide indicates that "any leakage of contaminated steam through these valves is controlled by a leakage control system" and describes staff positions defining acceptable characteristics of such a system. In the EPRI FSER and ABWR FSER, the staff has taken the position that proposals to eliminate the MSIVLCS by taking credit for fission product plate-out and holdup in main steam lines and the condenser would be acceptable, subject to complying with additional requirements. In the ABWR FSER, the staff described positions in Sections 10.3, 10.4, and 3.2.1, regarding design approaches to provide assurance that necessary portions of the main steam system and condenser maintain their integrity following an SSE. In Section 15.4.4.2, the staff addresses several considerations affecting the ability of this alternative design approach to comply with 10 CFR 100 requirements.

Consideration should be given to revising Section 6.7 to indicate that a design without an MSIVLCS may be acceptable and to indicate the appropriate Sections of the SRP (to be identified following integration of the balance of the SRP Sections) where specific regulatory positions regarding this approach are located.



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#### **Potential Impacts/Documents supporting the Suggested Changes:**

15082/NRC GENERIC LETTER 86-17; 22102/FINAL SER EPRI CH 1; 22103/DRAFT FINAL SER ABWR CH 15; 22104/DRAFT FINAL SER ABWR CH 10; 22471/SECY 93-087; 22473/DRAFT FINAL SER ABWR CH 3; 22476/DRAFT FINAL SER ABWR CH 14

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**Integrated Impact Number:** 119      **SRP Section Number:** 9.3.5

#### **Suggested Changes to the SRP Section:**

Add 10 CFR 50.62(c)(4) as Acceptance Criteria and develop associated Review Procedures for review of BWR standby liquid control system (SLCS) compliance with the ATWS Rule.

GSI A-9, 'Anticipated Transient Without Scram (ATWS),' was resolved with the promulgation of the ATWS Rule, 10 CFR 50.62. 10 CFR 50.62(c)(4) establishes requirements for a reliable, automatically initiated SLCS capable of specified minimum reactivity insertion. NRC Generic Letter 85-03 provides clarification of the reactivity control equivalency requirements of 10 CFR 50.62(c)(4).

Consideration should be given to incorporating 10 CFR 50.62(c)(4) as Acceptance Criteria for SRP Section 9.3.5 and developing associated Review Procedures. Consideration should also be given to identifying Generic Letter 85-03 in the proposed Review Procedures.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

2029/CODE OF FED. REGS 10CFR50; 2073/NRC GENERIC LETTER 85-03; 22278/FINAL SER ABWR CH 9; 22280/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 120      **SRP Section Number:** 9.3.5

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures to verify inclusion of specific standby liquid control system (SLCS) components in the applicant's Reliability Assurance Program (RAP).

The RAP was identified by the staff, in SECY-89-13, as an advanced light water reactor (ALWR) issue. The current interim staff position with respect to the RAP for future reactors is described in SECY-93-87. Consideration of an SRP review associated with the RAP will be addressed in an integrated impact associated with Chapter 17 of the SRP.

SLCS storage tank discharge valves are a new feature of the ABWR SLCS incorporated as part

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of the design approach taken to eliminate the explosive actuated injection valves of earlier BWR designs. The ABWR DFSE describes a staff position that ABWR SLCS storage tank discharge valves must be included in the applicant's RAP because of the importance of reliable operation of these valves.

Consideration should be given to developing a Review Procedures to verify the inclusion of SLCS storage tank discharge valves in the applicant's Reliability Assurance Program.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22279/FINAL SER ABWR CH 9

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**Integrated Impact Number:** 121      **SRP Section Number:** 15.6.4

#### **Suggested Changes to the SRP Section:**

Perform detailed comparison of ICRP 2, 1959 and ICRP 30, 1989.

SRP Section 15.6.4 cites Regulatory Guide 1.5 in Acceptance Criterion 3. Regulatory Guide 1.5 cites ICRP Publication 2, 1959, in Position 6. ICRP Publication 2 has been replaced by ICRP Publication 30. The current version is ICRP Publication 30, 1989.

Consideration should be given to performing a detailed side by side comparison between ICRP 2, 1959 and ICRP 30, 1989 to allow SRP reviewers to use the more current version of the standard. (Such a comparison would also support revision efforts for RG 1.5. IPD 7.0 form number 15.6.4-1 has been prepared to address the need to revise RG 1.5.)

#### **INSPECTION PROGRAM BRANCH COMMENT:**

No comparison needed. ICRP 2 was changed to ICRP 30 which is used in the new 10 CFR Part 20. ICRP 30 was replaced by ICRP 61-1990 which the Office of Research is reviewing.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20674/RG 1.5; 24136/C&S: ICRP 2 1959

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**Integrated Impact Number:** 122      **SRP Section Number:** 9.2.1

#### **Suggested Changes to the SRP Section:**

Revise Areas of Review to reflect the Design Certification/Combined License (COL) process.

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SRP Section 9.2.1 does not specifically address the process provided in 10CFR52. 10CFR52.47(b), 10CFR52.79(b) and the NRC Policy Statement on Nuclear Power Plant Standardization make clear that site specific elements of plant design such as the service water intake structure and the ultimate heat sink need not be complete for certification of a nuclear power plant design. Design features that depend upon the site are reviewed at the time of an individual COL application. In Section 9.2.15 of the ABWR FSER, the scope of service water system included in the ABWR design certification application was described as those service water components located in the control building. Equipment outside the control building, including the service water pumps, were not included.

Consider clarifying statements in Areas of Review to the effect that (1) at the Design Certification stage, site-specific features of the design are not reviewed and (2) that site-specific design features are reviewed at the time of a COL application.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

2277/CODE OF FED. REGS 10CFR52; 2279/NRC POLICY STATEMENT 52 FR 34884;  
22665/FINAL SER ABWR CH 9

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**Integrated Impact Number:** 123      **SRP Section Number:** 9.2.1

#### **Suggested Changes to the SRP Section:**

Add Review Procedures for confirmation that an ongoing program of surveillance and control is implemented and maintained to reduce fouling problems in open-cycle service water systems.

Reports of serious fouling events caused by mud, silt, corrosion products, or aquatic bivalve organisms in open-cycle service water systems led the NRC to establish Generic Issue 51, "Improving the Reliability of Open-Cycle Service Water Systems." To resolve this issue, the NRC initiated a research program to compare alternative surveillance and control programs to minimize the effects of fouling on plant safety. Initially, the program was restricted to a study of biofouling, but in 1987 the program was expanded to also address fouling by mud, silt, and corrosion products. The research program was completed in 1989 and the NRC issued Generic Letter 89-13 (including Supplement 1) directing licensees and applicants to implement a surveillance and control program. Such a program also addresses Generic Issues 32 and 52.

The EPRI Evolutionary Plant FSER and the ABWR FSER describe provisions to address GI 51, and indicate the staff's expectation of compliance with Generic Letter 89-13. The EPRI Evolutionary Plant FSER describes the staff's application of Generic Letter 89-13 guidance regarding design provisions for heat exchanger heat load testing for ALWRs.

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Consider adding Review Procedures to confirm that appropriate design provisions and inspection programs will be implemented to detect and control fouling in the service water system.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1109/NUREG 0933; 1424/NUREG 0933; 1430/NUREG 0933; 2412/NRC BULLETIN 81-03; 2520/NRC GENERIC LETTER 89-13; 2521/NRC GENERIC LETTER 89-13 Sup 1; 22110/FINAL SER ABWR CH 20; 22112/FINAL SER EPRI CH 1; 22155/FINAL SER EPRI CH 8; 22421/FINAL SER ABWR CH 20; 22422/FINAL SER CE80 CH 20; 22664/FINAL SER EPRI CH 8

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**Integrated Impact Number:** 124      **SRP Section Number:** 9.2.1

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure related to cross-connected service water systems at multi-unit sites.

GI-130 identified a concern with multi-unit sites that have only two service water pumps per plant with crosstie capability between plants. Service water pumps may not always be available during all possible operating conditions increasing vulnerability to core-melt sequences at the site. GI-130 has been resolved as described in Generic Letter 91-13. Generic Letter 91-13 describes the staff's position with regard to resolution of GI-130 for multi-unit sites with service water crosstie capability, including Technical Specification changes and improvements to emergency procedures for a loss of essential service water.

In addition to the issue identified in GI-130, an applicant for a multi-unit site with service water crosstie capability must address the requirements of GDC 5 related to system sharing. GDC-5 is currently identified as an Acceptance Criteria in SRP Section 9.2.1, however, verification of compliance with this requirement is not specifically addressed in Review Procedures.

Consider adding a Review Procedure, applicable to multi-unit sites with service water crosstie capability, which; (1) confirms that the applicant has addressed the concern identified in GI-130 based on the positions described in Generic Letter 91-13, and (2) verifies compliance with the requirements of GDC-5.

It should be noted that part of the issue considered in GI-130 was addressed in the resolution of GI-51 (described in Integrated Impact number 123). Specifically, the resolution of GI-130 assumes that the flushing and flow testing provisions of Generic Letter 89-13 will be applied to crosstie lines. Consideration should be given to including this position either in the Review Procedure proposed in this integrated impact or as part of the implementation of the Review Procedures proposed in Integrated Impact 123.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

2263/CODE OF FED. REGS 10CFR50; 15509/NUREG 0933; 22409/FINAL SER ABWR CH 20; 22410/FINAL SER CE80 CH 20; 22411/FINAL SER EPRI CH 1; 22412/FINAL SER EPRI CH 8; 22444/NRC GENERIC LETTER 91-13

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**Integrated Impact Number:** 125      **SRP Section Number:** 9.2.1

#### **Suggested Changes to the SRP Section:**

Revise current Acceptance Criteria and associated Review Procedures related to TMI Item II.K.1.22.

NRC Bulletin 79-08 item 3 required BWR licensees to describe the actions, both automatic and manual, necessary for proper functioning of the auxiliary heat removal systems (e.g., RCIC) that are used when the main feedwater system is not operable. For any manual action necessary, licensees were required to describe in summary form the procedure by which this action is taken in a timely sense.

This Bulletin was evaluated in Task II.K.1 as described in NUREG 0660. This Bulletin item was identified as II.K.1.22 and was approved for implementation without further clarification in NUREG 0737 as an item applicable to BWR applicants. II.K.1.22 was also identified in NUREG 0718 as an item applicable to BWRs. A BWR license application requirement related to this issue was incorporated in the CFR as 10 CFR 50.34(f)(2)(xxi). The SRP currently identifies TMI items related to this issue in Acceptance Criteria II.4.d (applicable to BWRs) and II.4.e (applicable to B&W plants) and in associated Review Procedures and Evaluation Findings. Aside from the SRP citation (II.4.e), no position could be identified that establishes applicability of TMI Action Plan Item II.K.1.22 to B&W applicants.

Consideration should be given to updating Acceptance Criteria and other related portions of SRP Section 9.2.1 to reflect the 10 CFR 50.34(f)(2)(xxi) requirement and the position indicated in Bulletin 79-08 item 3 and to identify II.K.1.22 as a NUREG 0737 item applicable to BWRs.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22413/CODE OF FED. REGS 10CFR50; 22414/FINAL SER ABWR CH 20

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**Integrated Impact Number:** 193      **SRP Section Number:** 15.6.3

#### **Suggested Changes to the SRP Section:**

Develop a Review Procedure that addresses steam generator overfill due to tube rupture.

Staff concerns regarding the potential for steam generator overfill due to steam generator tube rupture (SGTR) are discussed as part of the steam generator integrity issues under GI-67 (items 67.3.1, 67.5.1, and 67.5.2) and GI-135. In the CE80+ Draft SER review, the staff requested that the applicant address the staff's concern about preventing steam generator overfilling during an SGTR, including the technical basis that demonstrates the effectiveness of the operating strategy and compliance with 10 CFR Part 100 requirements.

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 an SGTR.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14683/NUREG 0933; 15072/NUREG 0933; 15133/NUREG 0933; 15137/NUREG 0933;  
22345/FINAL SER CE80 CH 15; 22357/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 194      **SRP Section Number:** 15.6.3

#### **Suggested Changes to the SRP Section:**

Revise existing Review Procedure to incorporate the improved Standard Technical Specifications (STS).

SRP Section 15.6.3 references the STS for the three PWR vendors (W, B&W, and CE) to obtain primary-to-secondary leakage limits, and primary and secondary coolant iodine concentration limits. Improved STSs (NUREGs -1430, -1431, and -1432) were developed based on the criteria in the interim Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, dated February 6, 1987.

Consider incorporating the improved Standard Technical Specifications in the Review Procedures. Further review at draft revision stage will determine if the improved STS limits are equivalent to the existing STS limits for primary-to-secondary leakage limits, and primary and

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secondary coolant iodine concentration limits.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22263/NUREG 1430; 22265/NUREG 1431; 22266/NUREG 1432; 22356/NRC POLICY STATEMENT 52 FR 3788

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**Integrated Impact Number:** 195      **SRP Section Number:** 15.6.3

#### **Suggested Changes to the SRP Section:**

Develop a Review Procedure related to the potential for containment bypass due to steam generator tube rupture (SGTR).

In SECY-90-016, Issue III.D, and SECY-93-087, Issue I.J, the staff discussed containment performance emphasizing the importance of maintaining containment integrity following a postulated severe accident. In SECY-93-087, Issue II.R, the staff described an additional containment performance issue involving a rupture of one or more steam generator tubes that could lead to a bypass of containment. The staff indicated that containment bypass due to SGTR could violate containment integrity and hamper attainment of severe accident goals.

In response to this issue, the staff recommended in SECY-93-087 that the Commission approve the position to require that the applicant for design certification for an evolutionary PWR assess design features to mitigate the amount of containment bypass leakage that could result from an SGTR. In the CE80+ DSER, the staff requested the applicant to consider potential design features that would reduce the amount of containment bypass leakage from an SGTR and to justify rejection of any mitigating options on the basis of low risk, taking into account the uncertainties in these calculations.

Consider incorporating a Review Procedure to assess the applicant's consideration of potential design features that would reduce the amount of containment bypass that could result from SGTR.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22346/FINAL SER CE80 CH 15; 22358/FINAL SER EPRI CH 1; 22359/SECY 90-016; 22383/SECY 93-087

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**Integrated Impact Number:** 196      **SRP Section Number:** 15.6.3

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures and Areas of Review to include verification that the most limiting single failure was considered in the SGTR analysis.

The current SRP review assures that the most severe case be considered with respect to the release of fission products and calculated doses. The assumption of a concurrent, most limiting single failure is not stated explicitly in SRP Section 15.6.3 (except for calculation of fuel failures in which the most reactive control rod is assumed to stick in the fully withdrawn position). In the CE80+ Draft SER, the staff indicated that it reviewed CE's determination that failure of a atmospheric dump valve on the affected steam generator to close constituted the most limiting failure.

Consideration should be given to revising the Review Procedures and Areas of Review to assure the most limiting single failure is considered for the SGTR analysis.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22347/FINAL SER CE80 CH 15

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**Integrated Impact Number:** 197      **SRP Section Number:** 3.11

#### **Suggested Changes to the SRP Section:**

Add a discussion of 10 CFR 50.49, the "EQ Rule," to Acceptance Criteria and Evaluation Findings.

The staff published 10 CFR 50.49 and RG 1.89, Rev. 1, to establish EQ requirements and provide more detailed guidance on methods and procedures acceptable to the staff for demonstrating the capability of electric equipment to perform necessary safety functions. Additional guidance on equipment to be included in an EQ program is also presented (e.g. safety-related equipment, selected non-safety-related equipment, and certain items of post-accident monitoring equipment). The guidance and requirements in these documents complement the guidance and requirements in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" and IEEE Std 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." NUREG-0588 and IEEE Std 323 are already addressed in SRP Section 3.11.

It should be noted that much of the information and many of the requirements addressed in 10



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CFR 50.49 are already addressed in documents discussed in SRP Section 3.11 such as NUREG-0588, and other Regulatory Guides and IEEE Standards. In addition, 10 CFR 50.49 is limited to electric equipment whereas SRP Section 3.11 indicates that NUREG-0588 and IEEE Std 323 are also applicable to mechanical equipment. Further evaluation is needed to determine to what extent these documents are interrelated and how their requirements apply to mechanical equipment as well as electrical equipment.

Consider modifying Areas of Review to include a discussion of safety-related equipment, non-safety-related equipment, and certain items of post-accident monitoring equipment addressed in 10 CFR 50.49(b)(1), (b)(2) and (b)(3). Examples of equipment in these categories is also discussed in RG 1.89, Rev.1, Appendix A, "Typical Safety-Related Electric Equipment or Systems;" and Appendix B, "Typical Examples of Non-Safety-Related Equipment."

Consider modifying Acceptance Criteria to include a discussion of the RG 1.89, Rev. 1, staff positions for endorsing IEEE Std 323-1974.

Consider incorporating discussions of RG 1.89, Rev. 1 and 10 CFR 50.49 into Acceptance Criteria as these documents provide elaboration on methods and procedures acceptable to the staff for meeting the general requirements for environmental design and qualification addressed in GDC 4. These documents provide additional guidance that complement the guidance and requirements of NUREG-0588 for Category 1 equipment, and IEEE Std 323-1974.

Consider deleting reference to NUREG-0588, Category II equipment, because all replacement and future equipment is required to be qualified in accordance with 10 CFR 50.49, NUREG-0588 Category 1, RG 1.89 Rev. 1, IEEE 323-1974, and other applicable standards and documents discussed in SRP Section 3.11.

Consider modifying Implementation and delete reference to the June 30, 1982 deadline because 10 CFR 50.49 paragraph (g) changed the June 30, 1982 deadline to November 30, 1985. 10 CFR 50.49 paragraphs (h) and (i) elaborate on the change.

Consider including reference to 10 CFR 50.49 in Evaluation Findings.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

6471/RG 1.89; 12546/CODE OF FED. REGS 10CFR50; 21374/NRC POLICY STATEMENT  
49 FR 8422

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**Integrated Impact Number:** 198      **SRP Section Number:** 3.11

#### **Suggested Changes to the SRP Section:**

Add a discussion of RG 1.158's endorsement of IEEE Std 535 to Acceptance Criteria.

The staff published RG 1.158, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants," in February 1989. RG 1.158 endorses IEEE Std 535-1986, "IEEE Standard for Qualification of Class 1E Lead Storage batteries for Nuclear Power Generating Stations" and also provides regulatory guidance for replacement batteries. IEEE Std 535 was previously addressed in SRP Section 3.11 as a standard to be used for guidance even though it was not endorsed by a RG.

Consider updating Acceptance Criteria and Evaluation Findings to reflect the publication of RG 1.158 and its endorsement of IEEE Std 535-1986 including the additional guidance on replacement batteries.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

7240/RG 1.158

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**Integrated Impact Number:** 199      **SRP Section Number:** 3.11

#### **Suggested Changes to the SRP Section:**

Add a discussion of RG 1.156's endorsement of IEEE Std 572 to Acceptance Criteria.

The NRC published RG 1.156, "Environmental Qualification of Connection Assemblies for Nuclear Power Plants," in November 1987. RG 1.156 endorses ANSI/IEEE Std 572-1985 as satisfying the Commission's regulations pertaining to the environmental qualification of quick-disconnect connection assemblies and environmental seals in combination with cables or wires.

It is noted that SRP Section 3.11 currently addresses IEEE Std 383 and RG 1.131 which deal with cables, field splices, and connections. Further evaluation is needed to determine the effect of adding RG 1.156 and IEEE Std 572 to the current write-up on RG 1.131 and IEEE Std 383.

Consider updating Acceptance Criteria and Evaluation Findings to reflect the publication of RG 1.156 and its endorsement of IEEE Std 572-1985.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

7225/RG 1.131; 7239/RG 1.156

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**Integrated Impact Number:** 202      **SRP Section Number:** 3.11

#### **Suggested Changes to the SRP Section:**

Add a discussion of RG 1.97 to Areas of Review.

The NRC published RG 1.97, Rev. 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," in May 1983.

Consider including a discussion of RG 1.97, Rev. 3 into Areas of Review under electrical equipment covered by SRP Section 3.11.

It should be noted that RG 1.97, Rev. 2 is discussed in RG 1.89, Rev. 1 and 10 CFR 50.49, which are also the subject of other integrated impacts for SRP Section 3.11. Further discussion of the RG 1.97, Rev. 2 instrumentation will be automatically included in Acceptance Criteria and Evaluation Findings once RG 1.89, Rev. 1 and 10 CFR 50.49 are incorporated into SRP Section 3.11. However, RG 1.97, Rev. 3 needs to be specifically discussed in SRP Section 3.11 until 10 CFR 50.49 and RG 1.89 are revised.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

6912/RG 1.97; 25653/FINAL SER ABWR CH 3; 25654/FINAL SER CE80 CH 3

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**Integrated Impact Number:** 204      **SRP Section Number:** 3.11

#### **Suggested Changes to the SRP Section:**

Add a discussion and provide examples of non-metallic mechanical equipment sensitive to environmental conditions into Acceptance Criteria.

Address Topical Report CENPD-255-A, Rev. 3, "Qualification of Class 1-E Electrical Equipment," October 1985 in Acceptance Criteria or Review Procedures.

Add a discussion of the NRC staff position on radiation levels that constitute a mild environment for electronic equipment and other electrical equipment to Acceptance Criteria.

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In the DSER, 3.11.3.2.2, "Safety-Related Mechanical Equipment in a Harsh Environment," the staff provides examples of non-metallic components that are subject to environmental degradation, e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms.

The staff has approved Topical Report CENPD-255-A, Rev. 3, "Qualification of Class 1 E Electrical Equipment," October 1985. CENPD-255-A, Rev. 3, is a generic document written by the applicant that provides a methodology to be used by CE applicants to qualify safety-related electrical equipment located in a harsh environment. Because this document is generic in nature and does not address qualification requirements with the level of detail deemed adequate by the staff the staff requires that applicants provide actual plant-specific values that are in compliance with the requirements of 10 CFR 50.49.

RG 1.89, Rev.1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" states that organic compounds in electrical equipment with the least radiation resistance have damage thresholds greater than 1 E4 rads and that electronic components with metal-oxide semiconductor devices may have a lower exposure threshold. NUREG-1462 provides more specific guidance and states, the staff position is that a mild radiation environment for electronic components has a total integrated dose less than 1 E3 rads.

Consider modifying Areas of Review and Acceptance Criteria to provide examples of non-metallic parts that are subject to environmental degradation.

Consider including reference to Topical Report CENPD-255-A, Rev. 3, as the staff has approved this document. Also include in the discussion that the document is generic in nature and that the applicant must provide plant-specific values that are in compliance with the requirements of 10 CFR 50.49.

Consider modifying Acceptance Criteria to include a discussion of the NRC staff position on mild environment radiation threshold limits of 1 E3 rads for electronic equipment and 1 E4 rads for other electrical equipment.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23070/FINAL SER CE80 CH 3

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**Integrated Impact Number:** 205      **SRP Section Number:** 3.11

#### **Suggested Changes to the SRP Section:**

Add a discussion and provide examples of non-metallic mechanical equipment sensitive to environmental conditions into Acceptance Criteria.

Address GE Topical Report NEDE-24326-1 (proprietary) and GESSAR II design, "General Electric Standard Safety Analysis Report," in Acceptance Criteria or Review Procedures.

Add a discussion of the NRC staff position on radiation levels that constitute a mild environment for electronic equipment and other electrical equipment to Acceptance Criteria.

Add a discussion of Information Notice, IN 89-63, which discusses equipment wetting, i.e., flooding above the flood level.

Add a discussion addressing design life as it relates to equipment qualification into Acceptance Criteria.

In the DFSER, NUREG-1469, 3.11.2.2, "Safety-Related Mechanical Equipment in a Harsh Environment," the staff provides examples of non-metallic components subject to environmental degradation, e.g., greases, gaskets, and lubricants.

The staff has approved the environmental qualification program as discussed in GE Topical Report NEDE-24326-1 (proprietary) with exception of the GE position on time margin. This topical outlines the methodology GE used to qualify nuclear steam supply system safety-related electrical and mechanical equipment that is subject to a harsh environment. Qualification of mechanical equipment is also addressed in the GESSAR II design, "General Electric Standard Safety Analysis Report," . GESSAR II has been approved by the NRC staff. Because these documents are generic in nature and do not address all qualification requirements with the level of detail deemed adequate by the staff, the staff requires that applicants provide actual plant-specific values that are in compliance with the requirements of 10 CFR 50.49.

RG 1.89, Rev.1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" states that organic compounds in electrical equipment with the least radiation resistance have damage thresholds greater than  $1 \text{ E}4$  rads and that electronic components with metal-oxide semiconductor devices may have a lower exposure threshold. NUREG-1469 provides more specific guidance and states, the staff position is that a mild radiation environment for electronic components has a total integrated dose less than  $1 \text{ E}3$  rads.

IN 89-63 provides additional NRC staff guidance on the requirements to evaluate equipment

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wetting, i.e., flooding above the flood level. IN 89-63 provides new information that does not appear to be provided in other referenced NRC staff approved documents on equipment qualification.

NUREG-1469 states that normal and accident doses will be based on 60 years of operation. SRP Section 3.11 does not specifically address qualified life in years. Qualification of installed equipment must take into account the proposed plant life time.

Consider modifying Areas of Review and Acceptance Criteria to include more detailed discussions of mechanical equipment that are subject to environmental degradation and provide examples of non-metallic components.

Consider including reference to GE Topical Report NEDE-24326-1 and GESSAR II, as staff approved documents that provide generic guidance relative to qualification of electrical and mechanical equipment specific to GE plants and the ABWR. Also include in the discussion that the documents are generic in nature and that the applicant must provide plant-specific values that are in compliance with the requirements of 10 CFR 50.49.

Consider modifying Acceptance Criteria to include a discussion of the NRC staff position on mild environment radiation thresholds limits of 1 E3 rads for electronic equipment and 1 E4 rads for other electrical equipment.

Consider modifying Acceptance Criteria to address IN 89-63, equipment wetting concerns, i.e., flooding above the flood level,

Consider modifying Acceptance Criteria by adding a statement that equipment qualified life should be based on the proposed licensing period.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23071/FINAL SER ABWR CH 3

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**Integrated Impact Number: 206      SRP Section Number: 3.11**

#### **Suggested Changes to the SRP Section:**

Add a discussion of the NRC Staff approval of the new EPRI-proposed source term into Acceptance Criteria.

Add a discussion addressing design life as it relates to equipment qualification into Acceptance Criteria.

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Source terms are currently addressed in RG 1.89, Rev. 1, C.2.c., NUREG-0737, Section II.B.2, and Atomic Energy Commission Technical Information Document (TID) 14844. Source terms are used in determining the radiation environment associated with a design basis LOCA. In NUREG-1242, Section 4.8.2, use of a new EPRI developed source term for the ALWR is addressed. The new ALWR source term is a physically based source term in lieu of the source term specified in TID 14844. Following a NRC staff evaluation of the new EPRI-proposed source term, the staff is developing a revised source term for use during equipment qualification.

In addition, it should be noted that RG 1.89, Rev. 1, C.2.c, provides more specific guidance on qualification source term calculations than that provided in NUREG-0737, Section II.B.2.

NUREG-1242 states that ALWRs will be designed for 60 years of operation. The current SRP Section 3.11 does not specifically address qualified life in years. Qualification of installed equipment must take into account the proposed plant life time.

Consider modifying Acceptance Criteria to include a discussion of the proposed physically based source term that will be used in determining the radiation environment associated with a design basis LOCA for the ALWR. Also consider modifying Acceptance Criteria to address source terms as they are discussed in RG 1.89, Rev.1.

Consider modifying Acceptance Criteria by adding a statement that equipment qualified life should be based on the proposed licensing period.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23072/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 209      **SRP Section Number:** 3.9.2

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to include references for detailed test specifications for piping vibrations, thermal expansion, and dynamic effect pre-operational and startup testing.

Pre-operational and startup testing for piping vibrations, thermal expansion, and dynamic effects are reviewed in SRP Section 3.9.2. At present, an overview of the required testing is provided in the Acceptance Criteria and Review Procedures. However detailed test specifications are not included and no references are provided for additional guidance. In the ABWR DFSER, the staff found that the SSAR criteria (which included test specifications to be prepared in accordance with ASME/ANSI OM part 3 and draft Part 7) conformed with the guidelines of this SRP Section. In the CE80+ DSER the staff asked the applicant to commit to conduct this testing

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program in accordance with ASME/ANSI OM-1987 Part 3 and OM-1986 Part 7.

Consider changing the Review Procedures to include ASME/ANSI OM Parts 3 and 7 as references for developing the detailed test specifications required for piping vibrations, thermal expansion, and dynamic effects testing.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22218/FINAL SER ABWR CH 3; 22233/FINAL SER CE80 CH 3; 22276/C&S: ANSI OM;  
22277/C&S: ANSI OM

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**Integrated Impact Number: 211      SRP Section Number: 3.9.2**

#### **Suggested Changes to the SRP Section:**

Revise Areas of Review, Acceptance Criteria and Review Procedures pertaining to Seismic System Analysis to reference the corresponding portions of SRP Section 3.7.3.

SRP Section 3.9.2 Item 2 deals with seismic system analysis. This closely parallels the similarly labeled SRP Section 3.7.3. (It should be noted the piping systems referenced in 3.9.2 are considered a "seismic subsystem" in 3.7.3.) In the ABWR DFSE, the staff's review was done in accordance with SRP Section 3.9.2, but consisted of an evaluation of SSAR Section 3.7.3. In the CE80+ DSE, the staff's review of SSAR Section 3.9.2.2 was conducted in accordance with the requirements of SRP Section 3.7.3.

In the ABWR DFSE, CE80+ DSE and EPRI Evolutionary Plant FSE, the staff allowed techniques for considering independent support movements which were not discussed in SRP Section 3.9.2. In both the ABWR DFSE and CE80+ DSE, the staff provided clarification of RG 1.29, beyond that included in SRP Section 3.9.2, regarding the analysis of non-Category I systems' effects on Category I systems. At present, the number of earthquake cycles is based at least partially on the OBE, however, as discussed in SECY 90-016, the staff will consider on a case-by-case basis the decoupling of the OBE from the SSE. SECY 93-087 discusses the proposed positions related to this elimination of the OBE from design consideration, including two alternatives for accounting for the effect of eliminating the OBE on determining the number of earthquake cycles. Proposed Integrated Impacts to SRP Section 3.7.3 already address these issues.

In the ABWR DFSE, the staff requested a description and justification, of modal damping for composite structures be provided in SSAR Section 3.9.2.2. SRP Section 3.9.2's Acceptance Criteria and Review Procedures for damping do not now address analysis procedures for damping as does SRP Section 3.7.3 Item 5.



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The 1989 revisions to SRP Sections 3.7.2 and 3.7.3 provide guidance for modal combination of high frequency modes (3.7.2 Item 7 referenced in 3.7.3 Item 7), and additional clarification for methods of analysis for Category I buried piping.

Consider changes to the seismic system analysis Areas of Review, Acceptance Criteria, and Review Parameters in Section 3.9.2 to reference the appropriate Items from SRP Section 3.7.3.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

5954/RG 1.29; 22229/DRAFT FINAL SER ABWR CH 3; 22230/DRAFT FINAL SER ABWR CH 3; 22232/DRAFT FINAL SER ABWR CH 3; 22234/DRAFT SER CE80 CH 3; 22236/DRAFT SER CE80 CH 3; 22237/DRAFT SER CE80 CH 3; 22238/DRAFT FINAL SER ABWR CH 3; 22242/FINAL SER EPRI CH 1; 22262/NUREG 1061 VOL 4; 22268/NUREG 0800; 22274/NUREG 0800; 22352/SECY 93-087; 22354/SECY 90-016

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**Integrated Impact Number:** 212      **SRP Section Number:** 3.9.2

#### **Suggested Changes to the SRP Section:**

Revise existing discussion in the Acceptance Criteria for allowable damping values to reference SRP Section 3.7.1.

SRP Section 3.9.2 Item 2 reviews piping damping values using Regulatory Guide (RG) 1.61 as guidance. However, in the EPRI Evolutionary Plant FSER, the ABWR DFSE, and the CE80 + DSER, the staff allowed alternate damping values derived from the RG 1.84 conditional acceptance of ASME Code Case N-411.

In the ABWR DFSE and CE80 + DSER, the staff noted their review of SSAR Section 3.9.2 was based on the information contained in SSAR Section 3.7.3. For the CE80 + DSER, the staff also conducted the review in accordance with the requirements of SRP Section 3.7.3. Although SRP Section 3.7.3 provides analysis procedures for damping, it does not provide critical damping values. Section 3.9.2 could either provide independent encompassing references for damping values, or reference an SRP Section that does. RG 1.84 conditional acceptance of ASME Code Case N-411 is already included in a proposed Integrated Impact to SRP Section 3.7.1. SRP Section 3.7.1 Item 2 reviews overall critical damping values for the entire plant, including piping Systems.

In the existing discussion in the Acceptance Criteria for "Criteria Used for Damping," consider replacing the direct reference to RG 1.61 with a reference to SRP Section 3.7.1's discussion on "Percentage of Critical Damping Values."

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22192/RG 1.84; 22229/DRAFT FINAL SER ABWR CH 3; 22231/DRAFT FINAL SER ABWR CH 3; 22234/DRAFT SER CE80 CH 3; 22248/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 214      **SRP Section Number:** 3.9.2

#### **Suggested Changes to the SRP Section:**

Revise existing discussion of reactor internals and reactor coolant piping dynamic analysis loads to reflect asymmetric loads and Leak-Before-Break (LBB) policies.

Dynamic system analyses of the reactor internals and reactor coolant piping under the most severe LOCA, in combination with the SSE, are reviewed in SRP Section 3.9.2. The most severe LOCA is defined, in this SRP Section, as generally being a double-ended guillotine break of a primary coolant loop near a reactor vessel nozzle in the most critical normal operating mode. In GSI A-2, "Asymmetric Blowdown Loads on Reactor primary Coolant Systems," the NRC determined that asymmetric LOCA loadings have generic implications for all PWRs. GDC 4 was modified in 1986 to allow use of LBB analyses. It was noted in Generic Letter 84-04 that the use of LBB could exclude double-ended guillotine breaks as a design basis for protection against local dynamic effects. Application of LBB can also eliminate requirements to consider asymmetric loads per NUREG-0609 (Resolution to GSI A-2). The existing SRP Section 3.9.2 does not specifically include asymmetric loads as discussed in GSI A-2 or its resolution, nor does it allow LBB related reductions/ elimination of LOCA-related loads. The application of LBB technology to eliminate local dynamic effects of postulated pipe ruptures from the design basis was discussed by the staff in the EPRI Evolutionary Plant FSER, the ABWR DFSE and the CE80 + DSE. The required conditions for approval of LBB applications are included in draft SRP Section 3.6.3's implementation of NUREG-1061 Volume 3 requirements for LBB.

Consider adding discussions of asymmetric loads and LBB to the existing Acceptance Criteria discussion for dynamic system analysis. A cross reference to SRP Section 3.6.3 and/or NUREG 1061 Volume 3 should also be considered.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1463/NUREG 0933; 21736/CODE OF FED. REGS 10CFR50; 22235/DRAFT SER CE80 CH 3; 22252/DRAFT FINAL SER ABWR CH 3; 22254/FINAL SER EPRI CH 1; 22275/NUREG 1061 VOL 3; 22287/NRC GENERIC LETTER 84-04

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**Integrated Impact Number:** 215      **SRP Section Number:** 3.9.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to address the stiffness of anchorage of heavy equipment and components.

Generic Issue 146, "Support Flexibility of Equipment and Components," addressed the common industry practice of assuming rigid and fixed attachments between equipment, components, and piping and the supporting structural system. In some cases, particularly in anchorage of heavy equipment and components, this assumption can potentially lead to underestimation of seismic loadings. In NUREG 0933, Generic Issue 146, "Support Flexibility of Equipment and Components," the staff recommends revising pertinent sections of the SRP to ensure adequate treatment of anchorage stiffness assumptions.

Consider revising Acceptance Criteria to ensure adequate treatment of anchorage stiffness assumptions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23123/NRC MEMORANDUM 09/30/93, from E. Beckjord to F. Gillespie; 23124/NUREG 0933

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**Integrated Impact Number:** 216      **SRP Section Number:** 10.4.9

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures to address the potential for auxiliary feedwater system (AFWS) pump steam binding.

Steam binding, with its potential as a common-mode failure mechanism of the auxiliary feedwater system, was identified as Generic Issue 93. Bulletin 85-01 initially addressed the issue by proposing procedures for the identification of steam binding and restoration of the AFWS to operable status. Generic Letter 88-03 documented the resolution of GSI 93 and reinforced the positions provided in Bulletin 85-01. In the EPRI Evolutionary Plant FSER, the staff noted that AFWS pump steam binding would be addressed by the provision of temperature monitors that indicate and alarm in the Control Room, and vents/drains for removing steam following detection. Similar observations were also documented in the CE System 80+ FSER.

Consider developing Review Procedures to address the detection and correction of AFWS pump steam binding.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

4155/NRC BULLETIN 85-01; 15399/NUREG 0933; 20830/NRC GENERIC LETTER 88-03; 22416/FINAL SER EPRI CH 5; 22417/FINAL SER CE80 CH 10; 22418/FINAL SER CE80 CH 20

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**Integrated Impact Number:** 217      **SRP Section Number:** 10.4.9

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to include station blackout requirements and guidance applicable to the Auxiliary Feedwater System (AFWS).

10 CFR 50.63, the Station Blackout (SBO) rule, requires that each light-water-cooled nuclear power plant be able to withstand and recover from an SBO. Regulatory Guide 1.155 contains staff positions related to systems and components required for decay heat removal during an SBO. Certain positions in C.3.2, C.3.3, and C.3.5 would apply to portions of the AFWS necessary for decay heat removal for the required station blackout duration and for the associated instrumentation and controls.

A review of the SBO event and the applicant's associated analysis is the subject of new SRP Section 8.4 developed under the SRP-UDP. A review of the AFWS conformance with Regulatory Guide 1.155 positions, as applicable, would be coordinated with the review proposed for SRP Section 8.4.

Consider citing 10 CFR 50.63 and Regulatory Guide 1.155 requirements and guidance in Acceptance Criteria and developing a Review Procedure that coordinates the AFWS review with the SBO review of SRP Section 8.4, as appropriate.

NOTE: This Integrated Impact has been revised to reflect the content of new SRP Section 8.4 developed under the SRP-UDP.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4138/RG 1.155; 22488/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 218      **SRP Section Number:** 10.4.9

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to identify alternative methods for establishing auxiliary feedwater system (AFWS) unreliability.

TMI Action Plan Item II.E.1.1 and 10 CFR 50.34(f)(1)(ii) require a reliability analysis of AFW systems. The Acceptance Criteria of SRP Section 10.4.9, Paragraph 5.c, includes an acceptable unreliability for the AFWS in the range of 1E-04 to 1E-05 per demand based on an analysis using methods and data presented in NUREG 0611 and NUREG 0635. Conformance with this criteria is cited as the basis for a finding of acceptability with regard to TMI Action Plan Item II.E.1.1 in Evaluation Findings of the SRP.

As described in the CE 80+ FSER, CE evaluated system reliability based on overall system unavailability determined as part of the PRA using the methodology described in NUREG/CR-2815 and NUREG/CR-2300, and using component failure rates from several generally accepted generic data bases. In the CE 80+ FSER, the staff also indicated that AFWS unreliability was greater than 1E-04 for the station blackout scenario considered in the PRA and this was acceptable.

Consider revising Review Procedures to address the use of methods, different from those in NUREG-0611 and NUREG-0635, for determining AFWS unreliability. Consideration should also be given to revising Acceptance Criteria 5.c to cite TMI Action Plan Item II.E.1.1 and 10 CFR 50.34(f)(1)(ii) and to indicate that the unavailability criteria is applicable to demands from other than station blackout conditions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4094/CODE OF FED. REGS 10CFR50; 20826/NUREG 0737; 22564/FINAL SER CE80 CH 10

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**Integrated Impact Number:** 219      **SRP Section Number:** 10.4.9

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to include anticipated transient without scram (ATWS) considerations relative to the Auxiliary Feedwater System (AFWS).

Review Procedure III.2.g currently describes a review of control features associated with the AFWS. 10 CFR 50.62(C)(1) requires that each PWR must have a means to automatically

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initiate the AFWS diverse from the reactor trip system under conditions indicative of an ATWS. In the CE 80+ FSER, the staff described its review of the AFWS which included a discussion of ATWS actuation.

Consider incorporating 10 CFR 50.62 into Acceptance Criteria and revising Review Procedure III.2.g to address diverse AFWS initiation under conditions indicative of an ATWS.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4098/CODE OF FED. REGS 10CFR50; 22608/FINAL SER CE80 CH 15; 22624/FINAL SER CE80 CH 20; 22626/FINAL SER CE80 CH 10

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**Integrated Impact Number: 220      SRP Section Number: 4.4**

#### **Suggested Changes to the SRP Section:**

Provide additional Acceptance Criteria and add Review Procedures to address BWR flow instabilities.

SRP 4.4 addresses thermal-hydraulic stability during normal operation and anticipated operational occurrences. BWR flow instabilities, with the potential for power oscillations and violation of fuel design limits, was studied by the NRC staff as Generic Issue B-19. This issue was resolved for current plants as described in GL 86-02. For BWRs that did not preclude thermal-hydraulic instabilities by design, GL 86-02 discussed implementation of operating limitations, through modifications of their Technical Specifications, which provide for the detection and suppression of flux oscillations in operating regions of potential instability consistent with the recommendations of General Electric SIL-380, to assure compliance with GDC 12. As a result of additional BWR operating experience, and further study of issue by the BWROG and the NRC staff, Bulletin 88-07 and Supplement 1 thereto were issued to provide current licensees with additional guidance (related to operating procedures, training and instrumentation) on BWR oscillations and thermal-hydraulic stability.

The BWR Owners Group (BWROG) has developed several long-term solutions for thermal-hydraulic stability which have been approved by the NRC for application to operating reactors. This methodology, documented in Topical Reports NEDO-31960 and NEDO-31960 Supplement 1, "BWR Owners Group Long-Term Stability Solution Licensing Methodology," has been accepted by the staff. The ABWR ASER states that, in addition to the region exclusion system, the oscillation power range monitor, which is based upon the detection of oscillation signals by the Low Power Range Monitors, will be implemented in the ABWR design. The staff considered it to be acceptable for the ABWR. Generic Letter 94-02 also references the GE Topical Reports (as approved by the NRC in their related safety evaluation report) as acceptable

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long term corrective actions to BWR instability issues.

Consider incorporating GDC 12 into the Acceptance Criteria, modifying the "specific criteria" to refer to NEDO-31960 and NEDO-31960 Supplement 1 as approved by the related NRC safety evaluation report, and by augmenting the Review Procedures to address the staff positions related to BWR stability including those associated with Generic Letter 94-02.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

6797/NUREG 0933; 9888/NRC BULLETIN 88-07; 14884/NRC BULLETIN 88-07 Sup 1; 18779/NRC GENERIC LETTER 86-02; 18789/CODE OF FED. REGS 10CFR50; 22402/FINAL SER EPRI CH 4; 23492/NRC LETTER From A.C. THADANI to L.A. ENGLAND; 23493/FINAL SER ABWR CH 4; 24361/NRC GENERIC LETTER 94-02

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**Integrated Impact Number:** 221      **SRP Section Number:** 4.4

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures to address BWR thermal-hydraulic instability during ATWS events.

As part of an ongoing evaluation of BWR thermal-hydraulic instability issues, the NRC staff has expressed concerns about such phenomena during ATWS events. In the EPRI Evolutionary Plant FSER, the staff stated that they will review FDA/DC applications to assure that designers demonstrate T-H stability of the core during ATWS events. In documenting the review of GE Topical Reports NEDO-32047, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability," and NEDO-32164, Revision 0, "Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS," the staff found that the subject documents were acceptable for referencing in license applications.

Consider developing Review Procedures to incorporate the resolution of thermal-hydraulic instability concerns during ATWS events, as described in NEDO-32047, NEDO-32164, Revision 0 and the staff safety evaluation for these topical reports.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22404/FINAL SER EPRI CH 4; 22405/FINAL SER ABWR CH 4; 22558/FINAL SER ABWR CH 15; 23494/NRC LETTER From A.C. THADANI to L. ENGLISH

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**Integrated Impact Number:** 222      **SRP Section Number:** 4.4

#### **Suggested Changes to the SRP Section:**

Add Review Procedures for the assessment of BWR thermal-hydraulic (T-H) instability associated with extended cycle operation.

For evolutionary plant designs, EPRI specified that the core designs provide for extended cycle operation at reduced power or with reduced feedwater temperature. Since reducing feedwater temperature for extended operation would result in reduced T-H stability, the NRC staff stated that the benefits of extending cycle operation by reducing the feedwater temperature must be weighed against the undesirability of decreasing core T-H stability.

Consider modifying the Review Procedures to address the potential for T-H instabilities during extended cycle operation.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22557/FINAL SER EPRI CH 4

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**Integrated Impact Number:** 223      **SRP Section Number:** 4.4

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures to address verification of PWR thermal-hydraulic stability.

In the EPRI Evolutionary Plant FSER, the NRC staff stated their intent to make certain that the thermal-hydraulic stability and radial xenon stability characteristics of new designs have been established by testing. While the SRP currently describes a review of thermal-hydraulic stability, it does not describe a specific review of verification of thermal-hydraulic stability by testing.

Consider developing Review Procedures to address the verification of thermal-hydraulic stability in new PWR designs.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22560/FINAL SER EPRI CH 4; 22563/FINAL SER CE80 CH 4

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**Integrated Impact Number:** 224      **SRP Section Number:** 4.4

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for (N-1) Loop Operation in PWRs and BWRs, and for inoperable reactor internal pumps (RIPs) in evolutionary BWRs.

The SRP 4.4 Acceptance Criteria includes "specific criteria" for thermal-hydraulic stability during part loop operation. Generic Issue B-59 initially noted that (N-1) Loop Operation was not permitted in BWRs and PWRs, pending further analytical justification. The NRC provided the technical resolution of Generic Issue B-59 in Generic Letter 86-09, which also provided guidance for justifying part loop operation in PWRs and BWRs.

In the EPRI Evolutionary Plant FSER the NRC staff reiterated the position articulated in Generic Letter 86-09. In the ABWR FSER, the staff discussed technical specifications related to operation with less than ten RIPs.

Consider incorporating review guidance into Review Procedures for (N-1) loop operation from Generic Letter 86-09 and for inoperable RIPs in the evolutionary BWRs.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14875/NRC GENERIC LETTER 86-09; 17996/NUREG 0933; 22403/FINAL SER EPRI CH 1; 22559/FINAL SER ABWR CH 4

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**Integrated Impact Number:** 225      **SRP Section Number:** 2.3.1

#### **Suggested Changes to the SRP Section:**

Update the reference to ANSI A58.1 to the current version.

Standard ANSI A58.1 is cited (1972 version) in SRP Section 2.3.1, Acceptance Criteria, as guidance for determining operating basis wind velocity (fastest mile of wind). The standard is also cited in Review Procedures as guidance for review of snow and ice loadings, extreme winds, wind vertical velocity distributions, and gust factors. The current version of ANSI A58.1 has been revised and redesignated as ANSI/ASCE-7-1988. A detailed side-by-side comparison between the cited and current standard is currently being performed by PNL to allow SRP reviewers to use the more current version of the standard.

Consider updating the reference to ANSI A58.1 to the current version pending completion of the detailed standard comparison.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22915/C&S: ANSI A58.1; 22916/C&S: ASCE 7-88

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**Integrated Impact Number: 226      SRP Section Number: 2.3.1**

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to incorporate new staff positions on the design basis tornado.

SRP Section 2.3.1 currently cites Reg. Guide 1.76 as guidance for review of proposed design basis tornado parameters. In SECY 93-087, the staff concluded that it was acceptable to reduce the design-basis tornado wind speed to 322 km/hr (200 mph) west of the Rocky Mountains and to 482 km/hr (300 mph) east of the Rocky Mountains. In an SRM dated July 21, 1993, the Commission approved the staff's position, proposed in SECY-93-087, that a maximum wind speed of 482 km/hr (300 mph) be used as the design-basis tornado employed in the design of evolutionary plants. These staff positions resulted from a reevaluation of current regulatory guidance in Reg. Guide. 1.76, using more recent tornado strike data than was available at the time of issuance of Reg. Guide 1.76. In the ABWR FSER the staff accepted GE's use of a maximum tornado wind speed of 482 km/hr (300 mph). In the System 80+ FSER the staff accepted CE's use of the Reg. Guide 1.76 standards for max tornado wind speed.

Consideration should be given to revising Acceptance Criteria and Review Procedures to incorporate new staff positions on the design basis tornado. A recommendation to revise Reg. Guide 1.76 will be submitted for impacts related to SRP Section 3.5.1.4.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

15591/RG 1.76; 22995/SECY 93-087; 25766/FINAL SER ABWR CH 3; 25767/FINAL SER CE80 CH 3

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**Integrated Impact Number: 227      SRP Section Number: 2.3.1**

#### **Suggested Changes to the SRP Section:**

SRP Section 2.3.1 provides review of the CP/OL applicant's historical regional climatological research and analysis which demonstrates the applicability of regional data to the plant site so that the adequacy of meteorological-related design bases may be independently confirmed by the staff.

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10 CFR Part 52 provides requirements for site parameter envelopes that are to be included in applications for design certification and manufacturing licenses for standard plant designs. Applications which reference standard plant designs approved under 10 CFR 52 are required to address the conformance of site specific parameters with the site parameter envelope for the approved, certified standard design. In SERs documenting staff review of evolutionary plant applications for design certification, the staff addressed the requirements related to the site parameter envelope in SER Section 2.6.

Consideration should be given to developing a new SRP Section (See IPD-7.0 Form No. 2.3.1-1) for review of the site parameter envelope associated with standard plant applications as a candidate for future work. Consideration should also be given to revising existing SRP Sections, including SRP Section 2.3.1 for review of site-specific parameters to reflect the site parameter-related requirements of 10 CFR 52, for applications referencing a standard plant design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18877/CODE OF FED. REGS 10CFR52; 18878/CODE OF FED. REGS 10CFR52;  
18879/CODE OF FED. REGS 10CFR52; 18880/CODE OF FED. REGS 10CFR52;  
23085/DRAFT SER CE80 CH 2

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**Integrated Impact Number: 228      SRP Section Number: 2.3.2**

#### **Suggested Changes to the SRP Section:**

SRP Section 2.3.2 provides review of the CP/OL applicant's submittal relating to local topography, meteorological parameters, and assessment of the influence of the plant on local meteorological conditions so that the acceptability for meteorological-related environmental impacts and design considerations may be independently confirmed by the staff.

10 CFR Part 52 provides requirements for site parameter envelopes that are to be included in applications for design certification and manufacturing licenses for standard plant designs. Applications which reference standard plant designs approved under 10 CFR 52 are required to address the conformance of site specific parameters with the site parameter envelope for the approved, certified standard design. In SERs documenting staff review of evolutionary plant applications for design certification, the staff addressed the requirements related to the site parameter envelope in SER Section 2.6.

Consideration should be given to developing a new SRP Section for review of the site parameter envelope associated with standard plant applications, as a candidate for future work. Consideration should also be given to revising existing SRP Sections, including SRP Section

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2.3.2 for review of parameters to reflect the site parameter-related requirements of 10 CFR 52, for applications referencing a standard plant design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18855/CODE OF FED. REGS 10CFR52; 18856/CODE OF FED. REGS 10CFR52;  
18857/CODE OF FED. REGS 10CFR52; 18858/CODE OF FED. REGS 10CFR52;  
18859/CODE OF FED. REGS 10CFR52; 23116/DRAFT SER CE80 CH 2

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**Integrated Impact Number:** 229      **SRP Section Number:** 2.3.3

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria and Review Procedures to address the requirements and guidance for the meteorological measurement program contained in 10 CFR 50.47 and Appendix E of 10 CFR Part 50.

Section 50.47 of 10 CFR Part 50 requires nuclear power plant licensees to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. In developing the onsite and offsite emergency response plans, licensees should provide that "adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use." Appendix E to 10 CFR Part 50 requires that applicants for an operating license develop plans for coping with radiological emergencies. The plans must include criteria for determining when protective measures should be considered within and outside the site boundary, including capabilities for dose projection using real-time meteorological information. Therefore, it is necessary for the applicant to establish and maintain a meteorological program capable of rapidly assessing critical meteorological parameters.

Regulatory Guide 1.97 contains guidance on meteorological instrumentation design requirements. Table 2 (BWRs) and Table 3 (PWRs) of Regulatory Guide 1.97 provides specific guidance on design requirements for the instruments used to measure wind direction, wind speed, and atmospheric stability. This guidance on meteorological measurements is provided in the context of emergency response for the purpose of assessing radiological releases. In the EPRI FSER the NRC staff found that the use of Regulatory Guide 1.23 and 1.97 for defining the system's capability to monitor wind speed, wind direction and to determine atmospheric stability is acceptable. The EPRI FSER also documented the use of other design requirements such as the use of noninterruptible power sources, backup battery power for offsite meteorological equipment, and use of proper instrument ranges to comply with the guidance of Regulatory Guide 1.97. Supplement 1 to NUREG-0737 replaced the requirement of NUREG-0737 item III.A.2.2 on meteorological data and contains guidance on meteorological information required

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to ensure emergency response capability.

Consideration should be given to revising the Acceptance Criteria and Review Procedures to incorporate the requirements and guidance of 10 CFR 50.47 and Appendix E of 10 CFR Part 50 to ensure that the meteorological measurement program at nuclear power plants is capable of providing the meteorological information required for emergency response purposes.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

17332/CODE OF FED. REGS 10CFR50; 17355/RG 1.97; 17679/CODE OF FED. REGS 10CFR50; 22923/FINAL SER EPRI CH 9; 22932/NUREG 0737 SUPPLEMENT 1

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**Integrated Impact Number: 230      SRP Section Number: 3.5.1.3**

#### **Suggested Changes to the SRP Section:**

Revise SRP Section 3.5.1.3 as necessary to reflect the staff's current approach for reviewing the adequacy of turbine missile protection. The SRP currently identifies RG 1.115 as the principal source of guidance for the protection of safety-related structures, systems, and components from the effects of potential turbine missiles. RG 1.115 guidance is based upon limiting the total probability of unacceptable damage from turbine missiles to less than  $1.0E-7$  per reactor year.

In consideration of uncertainties in calculating the total probability of unacceptable damage from turbine missiles, the staff indicated in the EPRI FSER, ABWR DFSE, and CE80+ DSE, that low-pressure turbines should meet the specific failure probability of turbine-missile generation related to a specific turbine orientation. Specifically, the staff indicated that the failure probability of missile generation for a favorably oriented turbine should be less than  $1.0E-4$  per reactor year and for an unfavorably oriented turbine,  $1.0E-5$  per reactor-year.

In connection with the above position the staff indicated in the ABWR DFSE and CE80+ DSE that COL applicants should submit, for NRC approval, a turbine system maintenance program including probability calculations of turbine missile generation based on a methodology approved by the NRC. In the ABWR DFSE the staff described an appropriate approach for COL applicants with regard to completing turbine missile generation probability calculations and implementation of the turbine maintenance program. The ABWR DFSE also identified recommended actions for situations when the probability of turbine missile generation does not meet minimum values over the life of the plant.

Consideration should be given to revising SRP Section 3.5.1.3 to reflect the staff's current

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approach for reviewing the adequacy of turbine missile protection, as described in the SERs. Consideration should also be given to adding those positions to the current guidance for turbine missile protection in RG 1.115 as a future work item. This future work recommendation will be tracked with IPD 7.0 form 3.5.1.3-1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21110/RG 1.115; 22620/DRAFT FINAL SER ABWR CH 3; 22621/DRAFT FINAL SER ABWR CH 3; 22622/FINAL SER EPRI CH 13; 22623/FINAL SER EPRI CH 13; 22625/DRAFT SER CE80 CH 3; 22627/NUREG 1048 SUP 6

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**Integrated Impact Number:** 231      **SRP Section Number:** 5.2.1.1

#### **Suggested Changes to the SRP Section:**

Revise code references such that the SRP citations are not version specific.

The ASME BPV Code, 1980 edition, Section III, "Nuclear Power Plant Components," is cited in SRP Section 5.2.1.1 with respect to compliance with 10 CFR 50.55a. This citation describes the applicability of ASME Code Section III subsections to ASME Class 1, 2, 3, and MC components and systems. The NRC staff is actively involved in the development of the Code and has a high degree of familiarity with the Code and its evolution. The version of the Code acceptable to the NRC for a particular application is established in 10 CFR 50.55a.

Consideration should be given to revising SRP Section 5.2.1.1 such that the reference to ASME Code Section III is not specific to a particular version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20478/CODE OF FED. REGS 10CFR50; 22335/C&S: ASME III

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**Integrated Impact Number:** 232      **SRP Section Number:** 5.2.1.1

#### **Suggested Changes to the SRP Section:**

Revise the SRP Section to reflect a review of component code, code edition, applicable addenda, and component order date (where applicable) for all ASME Class 1, 2, and 3 components.

SRP Section 5.2.1.1 describes a review for compliance with 10 CFR 50.55a requirements related to component code, code edition, applicable addenda, and component order date (where applicable) for ASME Code Class 1 and 2 RCPB components. At present, the Section does not

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apply this review to non-RCPB Class 2 or any Class 3 components. It is noted that the classification of Quality Group B (ASME Class 2) and Quality Group C (ASME Class 3) components of other fluid systems important to safety is performed under SRP Section 3.2.2. However, SRP Section 3.2.2 does not describe a review of applicable component code, code edition, addenda or component order dates. In the CE80+ DSER, the staff requested that this information be included in Table 5.2-1 and Section 3.2.2 or 3.9.3. In the ABWR DFSE, the staff appears to have conducted their review of this information for all ASME Class 1, 2, and 3 components and their supports in SRP Section 5.2.1.1.

Consider expanding the present review of component code, code edition, applicable addenda and component order date, to include all ASME Class 1, 2, and 3 components.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22331/DRAFT SER CE80 CH 3; 22338/DRAFT FINAL SER ABWR CH 5

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**Integrated Impact Number:** 235      **SRP Section Number:** 5.2.1.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria, Review Procedures and Evaluation Findings to reflect design certification and combined license (COL) applicant related code case reviews.

10 CFR 50.55a requires that the code edition, addenda, and optional code cases to be applied to components and systems be determined by the provisions of Paragraph NCA-1140, Subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code. Applicable code cases to which NCA-1140 criteria are applied are determined in RGs 1.84, 1.85, and 1.147. SRP Section 5.2.1.2 implies a single review of the code cases with a design's SAR. However, in the ABWR DFSE and CE80+ DSER, the staff stated that the COL applicant may submit, with its COL application, future code cases that are endorsed in regulatory guides at the time of the COL application, provided they do not alter the staff's safety findings on the certified design.

Consider adding a review of a COL applicant's revised code case applicabilities (if any) to the Acceptance Criteria, Review Procedures and Evaluation Findings.

Regarding the above, it should be noted that this approach and the corresponding limits on the use of code cases are not necessarily reflected in NCA-1140. A detailed discussion of this issue is provided in the ABWR DFSE's Section 5.2.1.1. This issue will be tracked using IPD-7.0 form 5.2.1.1-1 concerning consideration of a revision to 10 CFR 50.55a to allow deviations from NCA-1140 for COL applicants.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

4298/CODE OF FED. REGS 10CFR50; 22387/C&S: ASME III; 22388/DRAFT FINAL SER ABWR CH 5; 22389/DRAFT FINAL SER ABWR CH 5; 22390/DRAFT SER CE80 CH 5

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**Integrated Impact Number: 236      SRP Section Number: 5.2.1.2**

#### **Suggested Changes to the SRP Section:**

Revise code references such that the SRP citations are not version specific.

The ASME BPV Code, 1980 edition, Section III, "Nuclear Power Plant Components," and ASME BPV Code, 1980 edition "Code Cases, Nuclear Components," are cited in SRP Section 5.2.1.2 with respect to compliance with 10 CFR 50.55a. These citations describe the applicability of ASME Code, Section III Subsections and ASME code cases to ASME Class 1, 2, 3, and MC components, component supports, core support structures and concrete containments. NRC staff is actively involved in the development of the Code and has a high degree of familiarity with the Code and its evolution. The version of the Code and code cases acceptable to the NRC for any particular application is established in 10 CFR 50.55a and in various regulatory guides.

Consideration should be given to revising SRP Section 5.2.1.2 such that references to ASME Code Section III and ASME code cases are not specific to a particular version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4298/CODE OF FED. REGS 10CFR50; 22384/C&S: ASME III

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**Integrated Impact Number: 237      SRP Section Number: 5.2.1.2**

#### **Suggested Changes to the SRP Section:**

Revise the Review Procedures to identify the scope of Regulatory Guides 1.84, 1.85, and 1.147 in a consistent manner.

SRP Section 5.2.1.2 Review Procedures ensure that the applicant's identified ASME code cases for ASME Section III, Division 1 and Division 2 components are checked for compliance with the acceptable code cases identified in RG 1.84, 1.85, and 1.147. (As noted in their titles, RG 1.84 and 1.85 are specifically applicable to only Division 1 of Section III Code Cases. Division 1 has a different meaning in Section XI and applies to LWRs in general. Therefore, RG 1.147 Code Cases could include Code Cases applicable to Section III, Division 1 and 2.) No Section



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III, Division 2 code cases are uniquely identified in these Regulatory Guides. The next paragraph of the review procedures notes that Section III, Division 2 code cases are reviewed on a case-by-case basis.

Consider updating the Review Procedures discussion to reflect that only ASME Code, Section III, Division 1, code cases are covered in RG 1.84 and 1.85.

IPD 7.0 Form INEL 5.2.1.2 R/R A - 1 was prepared regarding modifications to Regulatory Guide 1.70, Section 5.2.1.2 to incorporate reference to Regulatory Guide 1.147.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22125/RG 1.84; 22132/RG 1.85; 22139/RG 1.147

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**Integrated Impact Number:** 238      **SRP Section Number:** 5.2.2

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures and Branch Technical Position (BTP) RSB 5-2 to incorporate staff positions that address the resolution of Generic Issues (GI) 70 and 94. This SRP reviews the overpressure protection of the Reactor Coolant Pressure Boundary (RCPB) during all modes of operation. Low temperature overpressure protection (LTOP) is currently addressed in BTP RSB 5-2 by requiring the provision of a LTOP system to prevent the overpressurization of the RCPB under low-temperature, water-solid conditions.

GI 70 arose from staff concerns about the need to improve the reliability of power-operated relief valves (PORVs) and associated block valves. As a result of the event at TMI-2, Item II.D.1 of NUREG-0737 was issued to define functional requirements for both the PORVs and the block valves. The staff later questioned the acceptability of relying on non-safety grade PORVs to mitigate a design-basis accident, such as a steam generator tube rupture, and identified GI-70 to address the need for improving the reliability of PORVs and block valves. Related issues concerning the use of PORVs in LTOP systems during startup/shutdown operations, and the potential for brittle fracture of the reactor pressure vessel because of inoperable LTOP systems, were identified in GI 94.

GIs 70 and 94 were resolved as described in Generic Letter 90-06, which establishes positions for current plants and states that the safety classification of PORVs and associated block valves in new plants should be reconsidered when these valves are used to perform safety-related functions. The letter acknowledges that a number of improvements have been implemented at operating plants in response to the event at TMI-2, and defined additional requirements for improvements in quality assurance, procurement, maintenance practices, inservice inspection of

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valves, and modifications to the Technical Specifications in the area of limiting conditions for operation of PORVs and block valves.

Consider modifying the Review Procedures and BTP RSB 5-2 to incorporate staff positions that address the resolution of GIs 70 and 94.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14932/NUREG 0933; 15400/NUREG 0933; 17029/NRC GENERIC LETTER 90-06;  
22568/FINAL SER EPRI CH 1; 22570/FINAL SER CE80 CH 20

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**Integrated Impact Number:** 239      **SRP Section Number:** 2.3.4

#### **Suggested Changes to the SRP Section:**

SRP Section 2.3.4 provides guidance for review of atmospheric dispersion estimates for postulated accidental releases of effluents to the atmosphere.

10 CFR Part 52 contains requirements for site parameter envelopes that are to be included in applications for design certifications and manufacturing licenses for standard plant designs. Applications which reference standard plant designs approved under 10 CFR Part 52 are required to address the conformance of site specific parameters with the site parameter envelope for the approved certified standard design. In SERs documenting staff review of evolutionary plant applications for design certification, the staff addressed the requirements related to the site parameter envelope in Section 2.6.

Consideration should be given to developing a new SRP Section (See IPD-7.0 Form # 2.3.1-1) for review of the site parameter envelope associated with standard plant applications, as a candidate for future work. Consideration should also be given to revising existing SRP Sections, including SRP 2.3.4 for review of site-specific parameters to reflect the site parameter-related requirements of 10 CFR 52, for applications referencing a standard plant design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21297/CODE OF FED. REGS 10CFR52; 21298/CODE OF FED. REGS 10CFR52;  
23113/ADVANCE SER ABWR CH 2; 23115/DRAFT SER CE80 CH 2

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**Integrated Impact Number:** 240      **SRP Section Number:** 2.3.5

#### **Suggested Changes to the SRP Section:**

SRP Section 2.3.5 reviews atmospheric dispersion estimates for routine releases to the atmosphere, namely; atmospheric dispersion models, meteorological data used, diffusion parameters, and relative concentration and deposition values used for assessment of consequences of routine airborne radioactive releases.

10 CFR Part 52 contains requirements for site parameter envelopes that are to be included in applications for design certifications and manufacturing licenses for standard plant designs. Applications that reference standard plant designs approved under 10 CFR Part 52 are required to address the conformance of site specific parameters with the site parameter envelope for the approved certified standard design. In SERs documenting staff review of evolutionary plant applications for design certification, the staff addressed the requirements related to the site parameter envelope in Section 2.6.

Consideration should be given to developing a new SRP Section (See IPD-7.0 Form No. 2.3.1-1) for review of the site parameter envelope associated with standard plant applications, as a candidate for future work. Consideration should also be given to revising existing SRP Sections, including SRP Section 2.3.5 for review of site-specific parameters to reflect the site parameter-related requirements of 10 CFR 52, for applications referencing a standard plant design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21316/CODE OF FED. REGS 10CFR52; 21317/CODE OF FED. REGS 10CFR52;  
21318/CODE OF FED. REGS 10CFR52; 22948/DRAFT SER CE80 CH 2; 22961/ADVANCE  
SER ABWR CH 2

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**Integrated Impact Number:** 241      **SRP Section Number:** 2.3.5

#### **Suggested Changes to the SRP Section:**

Replace references to NUREG-0324 by reference to NUREG/CR-2919 with respect to the diffusion model used by the staff. NUREG-0324 is currently cited in Acceptance Criteria and Review Procedures as describing the staff's standard long-term diffusion model.

NUREG/CR-2919 is a user guide for the NRC computer program XOQDOQ, which supersedes NUREG-0324, published as a draft in September 1977. This program is used by NRC meteorology staff in their independent meteorological evaluation of routine or anticipated

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intermittent releases at nuclear power stations.

Consider modifying Acceptance Criteria and Review Procedures to replace reference to NUREG-0324 by reference to NUREG/CR-2919.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22844/NUREG CR-2919

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**Integrated Impact Number:** 242      **SRP Section Number:** 9.3.2

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to provide for review of Post-Accident Sampling System (PASS) capabilities for advanced light water reactors (ALWRs).

Currently, SRP Section 9.3.2 identifies the clarifications of NUREG 0737, TMI Action Plan Item II.B.3, and Reg. Guide 1.97 as part of the Acceptance Criteria for review of PASS designs. 10 CFR 50.34(f)(2)(viii) establishes post accident sampling requirements related to TMI Action Plan Item II.B.3.

In SECY-93-087 the staff recommended that the Commission approve alternatives to certain aspects of current PASS design criteria. The following items, applicable to ALWRs, are discussed in SECY-93-087 related to SRP Section 9.3.2 PASS criteria; 1) elimination of hydrogen sampling capability, 2) elimination of BWR dissolved gas sampling capability, 3) modification of PWR dissolved gas sampling criteria, and 4) modifications to boron and activity sampling criteria. Positions related to these issues were approved by the Commission with certain clarifications in an SRM dated July 21, 1993. The EPRI Evolutionary FSER discusses the staff's review of proposed deviations to current PASS design criteria.

In Section 9.3.2 of the ABB-CE 80+ FSER the staff documented that the System 80+ design has a PASS that complies with the requirements of 10 CFR 50.34(f)(2)(viii) and item II.B.3 of NUREG-0737 with the modifications described in SECY-93-087. The ABWR FSER also documents the fact that the ABWR design will have PASS which meets the requirements of 10 CFR 50.34(f)(2)(viii) and item II.B.3 of NUREG-0737 with the modifications described in SECY 93-087. In section 1.6 of the ABB-CE80+ FSER and the ABWR FSER the staff included a list of exemptions to certain regulations, the exemption from postaccident sampling discussed under section 9.3.2 was included in these lists.

Consideration should be given to revising Review Procedures to incorporate current guidance regarding PASS design reviews for ALWRs.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22296/FINAL SER EPRI CH 1; 22547/CODE OF FED. REGS 10CFR50; 22548/SECY 93-087; 22549/DRAFT FINAL SER ABWR CH 9; 22550/DRAFT SER CE80 CH 9; 24256/FINAL SER ABWR CH 9; 24263/FINAL SER CE80 CH 9; 24380/FINAL SER CE80 CH 1; 24381/FINAL SER ABWR CH 1

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**Integrated Impact Number:** 243      **SRP Section Number:** 9.3.2

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of the applicant's administrative program for post-accident sampling. The current Areas of Review for SRP Section 9.3.2 include OL stage reviews of the adequacy of the applicant's Technical Specifications and operating procedures related to post-accident sampling. NRC Generic Letter 83-36 includes a staff position that an administrative program should be established, implemented and maintained to ensure the capability to obtain and analyze samples under accident conditions. The program should include training, procedures for sampling/analysis, and provisions for maintenance of sample/analysis equipment. Generic Letter 83-36 states that it is acceptable to reference this program in the Administrative Controls section of Technical Specifications and include a detailed description in the plant operations manuals.

Consideration should be given to developing Review Procedures for review of the applicant's administrative program for post-accident sampling, based upon information provided in Generic Letter 83-36.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4396/NRC GENERIC LETTER 83-36; 24271/NRC GENERIC LETTER 83-37

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**Integrated Impact Number:** 244      **SRP Section Number:** 9.3.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to identify post-accident sampling system design basis personnel exposure limits. Currently, SRP Section 9.3.2 identifies NUREG-0737, TMI Action Plan Item II.B.3 as acceptance criteria for review of PASS systems.

Individual exposure limits are applied as design basis assumptions for post-accident sampling systems in TMI Action Plan Requirement II.B.3, Clarification (6). Clarification (6) specifies design basis limits of 5 rem to the whole body and 75 rem to the extremities based upon GDC

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10 CFR 50.34(f)(2)(viii), relating to TMI Item II.B.3, was amended to reflect new occupational exposure requirements of 10 CFR 20.1001-20.2401. The limits reflected in 10 CFR 50.34(f)(2)(viii) are 5 rem to the whole body and 50 rem to the extremities.

Consideration should be given to revising Acceptance Criteria to identify post-accident sampling system design basis personnel exposure limits, based upon limits identified in 10 CFR 50.34(f)(2)(viii).

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4435/CODE OF FED. REGS 10CFR50; 22547/CODE OF FED. REGS 10CFR50;  
23692/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 246      **SRP Section Number:** 9.5.4

#### **Suggested Changes to the SRP Section:**

Delete reference to BTP ICSB-17.

SRP Section 9.5.4 currently identifies BTP ICSB-17 as a criterion for the review of fuel oil storage and transfer system protective interlocks during accident conditions. Appendix 8-A of the SRP indicates that BTP ICSB 17 was superseded by Reg. Guide 1.9 (Rev. 2) Regulatory Position C.7. It should be noted that Reg. Guide 1.9 was recently revised (Rev. 3) and positions related to interlocks are now located in Regulatory Position C.1.8.

Consider substituting the reference to BTP ICSB-17 with a reference to Regulatory Guide 1.9 as related to the emergency diesel engine fuel oil storage and transfer system protective interlocks during accident conditions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20564/RG 1.9

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**Integrated Impact Number:** 249      **SRP Section Number:** 9.5.4

#### **Suggested Changes to the SRP Section:**

Revise review process to incorporate Generic Letter 83-26 provisions.

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Generic letter 83-26 requested that specific surveillance requirements for diesel fuel impurity level tests be included in the plant's Technical Specifications, and in the STS for all nuclear power reactor vendors.

Acceptance Criteria II.4.b of SRP Section 9.5.4 identifies Reg. Guide 1.137 as guidance related to the diesel fuel oil system design, fuel oil quality and testing.

Reg. Guide 1.137 identifies Appendix B to ANSI N195-1976, as supplemented by Regulatory Position 2 of the guide, as an acceptable basis for a program to maintain the quality of fuel oil. ANSI N195-1976 was superseded by ANSI/ANS 59.51. An augmented version of Appendix B to ANSI N195-1976 is included as Appendix C to ANSI/ANS 59.51-1989. A comparison between these standards is currently being performed by PNL.

The requirements in position C.2 of Reg. Guide 1.137 and Appendix C to ANSI/ANS 59.51-1989 are not as specific as the requirements of Generic Letter 83-26. Consider incorporating the provisions of Generic Letter 83-26 into the review procedures.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

84/RG 1.137; 7377/NRC GENERIC LETTER 83-26; 24277/NUREG 1430

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**Integrated Impact Number:** 250      **SRP Section Number:** 9.5.4

#### **Suggested Changes to the SRP Section:**

Add acceptance criteria and review procedures to address station blackout requirements.

For those plants that will use an emergency diesel generator as the alternate AC source in response to a station blackout event, the adequacy of the fuel oil supply and recharging capability, independent of preferred and onsite emergency ac power, of the emergency diesel engine fuel oil storage and transfer system need to be reviewed. Regulatory Position C.3 of RG 1.155 provides review guidance for this evaluation.

Consider adding 10 CFR 50.63 and RG 1.155 as acceptance criteria and developing applicable review procedures for the emergency diesel engine fuel oil storage and transfer system.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23197/CODE OF FED. REGS 10CFR50; 23198/RG 1.155

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**Integrated Impact Number:** 251      **SRP Section Number:** 9.5.7

**Suggested Changes to the SRP Section:**

Delete reference to BTP ICSB-17.

SRP Section 9.5.7 currently identifies BTP ICSB-17 as a criterion for the review of the lubricating system protective interlocks during accident conditions. Appendix 8-A of the SRP indicates that BTP ICSB 17 was superseded by Reg. Guide 1.9 (Rev. 2) Regulatory Position C.7. It should be noted that Reg. Guide 1.9 was recently revised (Rev. 3) and positions related to interlocks are now located in Regulatory Position C.1.8.

Consider substituting the reference to BTP ICSB-17 with a reference to Regulatory Guide 1.9 as related to the emergency diesel engine lubricating system protective interlocks during accident conditions.

**Potential Impacts/Documents supporting the Suggested Changes:**

20628/RG 1.9; 22690/FINAL SER ABWR CH 9

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**Integrated Impact Number:** 253      **SRP Section Number:** 9.5.7

**Suggested Changes to the SRP Section:**

Add acceptance criteria and review procedures to address station blackout requirements.

For those plants that will use an emergency diesel generator as the alternate AC source in response to a station blackout event, the capacity and capability, independent of preferred and onsite emergency ac power, of the emergency diesel engine lubrication system need to be reviewed. Regulatory Position C.3 of RG 1.155 provides review guidance for this evaluation.

Consider adding 10 CFR 50.63 and RG 1.155 as acceptance criteria and developing applicable review procedures for the emergency diesel engine lubrication system.

**Potential Impacts/Documents supporting the Suggested Changes:**

23201/CODE OF FED. REGS 10CFR50; 23202/RG 1.155

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**Integrated Impact Number:** 254      **SRP Section Number:** 9.5.8

**Suggested Changes to the SRP Section:**

Delete reference to BTP ICSB-17.

SRP Section 9.5.8 currently identifies BTP ICSB-17 as a criterion for the review of the combustion air intake and exhaust system protective interlocks during accident conditions. Appendix 8-A of the SRP indicates that BTP ICSB 17 was superseded by Reg. Guide 1.9 (Rev. 2) Regulatory Position C.7. It should be noted that Reg. Guide 1.9 was recently revised (Rev. 3) and positions related to interlocks are now located in Regulatory Position C.1.8.

Consider substituting the reference to BTP ICSB-17 with a reference to Regulatory Guide 1.9 as related to the diesel engine combustion air intake and exhaust system protective interlocks during accident conditions.

**Potential Impacts/Documents supporting the Suggested Changes:**

20640/RG 1.9; 22694/FINAL SER ABWR CH 9

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**Integrated Impact Number:** 256      **SRP Section Number:** 9.5.8

**Suggested Changes to the SRP Section:**

Add acceptance criteria and review procedures to address station blackout requirements.

For those plants that will use an emergency diesel generator as the alternate AC source in response to a station blackout event, the capacity and capability, independent of preferred and onsite emergency ac power, of the emergency diesel engine air intake and exhaust system need to be reviewed. Regulatory Position C.3 of RG 1.155 provides review guidance for this evaluation.

Consider adding 10 CFR 50.63 and RG 1.155 as acceptance criteria and developing applicable review procedures for the emergency diesel engine air intake and exhaust system.

**Potential Impacts/Documents supporting the Suggested Changes:**

23199/CODE OF FED. REGS 10CFR50; 23200/RG 1.155

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**Integrated Impact Number:** 257      **SRP Section Number:** 9.5.5

**Suggested Changes to the SRP Section:**

Substitute reference to Reg. Guide 1.9 for the reference to BTP ICSB-17.

SRP Section 9.5.5 currently identifies BTP ICSB-17 as a criterion for the review of cooling water system protective interlocks during accident conditions. Appendix 8-A of the SRP indicates that BTP ICSB 17 was superseded by Reg. Guide 1.9 (Rev. 2) Regulatory Position C.7. It should be noted that Reg. Guide 1.9 was recently revised (Rev. 3) and positions related to interlocks are now located in Regulatory Position C.1.8.

Consider substituting the reference to BTP ICSB-17 with a reference to Reg. Guide 1.9 as related to the emergency diesel engine cooling water system protective interlocks during accident conditions.

**Potential Impacts/Documents supporting the Suggested Changes:**

15583/RG 1.9; 22565/FINAL SER ABWR CH 9

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**Integrated Impact Number:** 259      **SRP Section Number:** 9.5.5

**Suggested Changes to the SRP Section:**

Add Acceptance Criteria and Review Procedures to address station blackout requirements.

For those plants that will use an emergency diesel generator as the alternate AC source in response to a station blackout event, the capacity and capability of the emergency diesel engine cooling water system to provide adequate cooling need to be reviewed. Regulatory Position C.3 of Reg. Guide 1.155 provides guidance for this evaluation.

Consider adding 10CFR50.63 and Reg. Guide 1.155 as acceptance criteria and developing applicable review procedures for the emergency diesel engine cooling water system.

**Potential Impacts/Documents supporting the Suggested Changes:**

23181/CODE OF FED. REGS 10CFR50; 23182/RG 1.155

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### **Integrated Impacts**

**Integrated Impact Number:** 260      **SRP Section Number:** 9.5.6

**Suggested Changes to the SRP Section:**

Substitute reference to Reg. Guide 1.9 for the reference to BTP ICSB-17.

SRP Section 9.5.6 currently identifies BTP ICSB-17 as a criterion for the review of the air starting system protective interlocks during accident conditions. Appendix 8-A of the SRP indicates that BTP ICSB 17 was superseded by Reg. Guide 1.9 (Rev. 2) Regulatory Position C.7. It should be noted that Reg. Guide 1.9 was recently revised (Rev. 3) and positions related to interlocks are now located in Regulatory Position C.1.8.

Consider substituting the reference to BTP ICSB-17 with a reference to Reg. Guide 1.9 as related to the emergency diesel engine starting system protective interlocks during accident conditions.

**Potential Impacts/Documents supporting the Suggested Changes:**

22395/RG 1.9

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**Integrated Impact Number:** 262      **SRP Section Number:** 9.5.6

**Suggested Changes to the SRP Section:**

Add Acceptance Criteria and Review Procedures to address compliance with the station blackout rule.

For those plants that will use an emergency diesel generator as the alternate AC source in response to a station blackout event, the capability of the emergency diesel engine starting system to start the alternate AC source without reliance on the preferred or emergency on-site power sources needs to be reviewed. Regulatory Position C.3 of Reg. Guide 1.155 provides guidance for this evaluation.

Consider adding 10CFR50.63 and Reg. Guide 1.155 as Acceptance Criteria and developing applicable Review Procedures for the emergency diesel engine starting system.

**Potential Impacts/Documents supporting the Suggested Changes:**

23183/CODE OF FED. REGS 10CFR50; 23184/RG 1.155

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### **Integrated Impacts**

**Integrated Impact Number:** 263      **SRP Section Number:** 3.10

#### **Suggested Changes to the SRP Section:**

Add a discussion of RG 1.100, Rev. 2, and its endorsement of IEEE Std. 344- 1987 to Acceptance Criteria.

Exceptions to IEEE Std. 344-1975 that were noted in RG 1.100, Rev. 1, have been resolved in the 1987 version. Also, IEEE Std. 344-1987 extends the applications of the standard to include discussion on topics such as the qualification of mechanical equipment, thermal distortion effects on equipment operability, and variations in dynamic and static testing. In addition, IEEE Std. 344-1987 recognizes the use of justified "experience data" for seismic qualification of mechanical and electrical equipment.

Consider updating Acceptance Criteria and Evaluation Findings to reflect the up-to-date staff positions, guidance and requirements presented in RG 1.100, Rev. 2 and its endorsement of IEEE Std. 344-1987.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

9245/RG 1.100

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**Integrated Impact Number:** 264      **SRP Section Number:** 3.10

#### **Suggested Changes to the SRP Section:**

Add a discussion of RG 1.97 instrumentation to Areas of Review and Acceptance Criteria.

This Integrated Impact recommends further research/regulatory action per IPD 7.0 form number 3.10-1.

The NRC staff published RG 1.97, Rev. 3, May 1983, which discusses plant variables required to be monitored and provides guidance on instrumentation that should be qualified. RG 1.97 addresses the qualification of instrumentation by stating that Category 1 instrumentation should be qualified in accordance with: RG 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants," and NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," which are discussed in SRP 3.11; and, RG 1.100, Seismic Qualification of Electric Equipment for Nuclear Power Plants, which is discussed in SRP 3.10.

RG 1.97, Section B, sixth paragraph, states: it is desirable that accident- monitoring instrumentation components and their mounts that cannot be located in seismically qualified

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### **Integrated Impacts**

buildings be designed to continue to function, to the extent feasible, following seismic events.

Consider adding a discussion of RG 1.97 instrumentation into Areas of Review and Acceptance Criteria. Accident monitoring instrumentation that cannot be located in seismically qualified buildings should be specifically addressed.

A review of the information provided in RG 1.89, Rev. 1; 10 CFR Part 50.49; IEEE Std. 323-1974; RG 1.100, Rev. 2; and IEEE Std. 344-1987 is needed to update Table 1 of RG 1.97 to reference these documents. An IPD 7.0 Form has been initiated.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

5173/RG 1.97; 5174/RG 1.97

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**Integrated Impact Number:** 266      **SRP Section Number:** 3.10

#### **Suggested Changes to the SRP Section:**

Replace all reference to 10 CFR Part 100, Appendix A, VI(a)(2) and the Operating Basis Earthquake with appropriate checks at a fraction of the safe shutdown earthquake (SSE).

10 CFR Part 100, A, Vibratory Ground Motion, VI(a)(2), addresses the Operating Basis Earthquake and is discussed in Acceptance Criteria and in Evaluation Findings.

SECY-93-087 - "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water-Reactor (ALWR) Designs," Item I.M, "Elimination of Operating Basis Earthquake (OBE)" discusses elimination of the OBE from seismic qualification and discusses alternative qualification requirements.

Consider modifying Acceptance Criteria and Evaluation Finding by replacing all reference to VI(a)(2) and the OBE with appropriate checks at a fraction of the SSE.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

5152/CODE OF FED. REGS 10CFR100; 23053/SECY 93-087; 23210/SECY 93-087; 25746/FINAL SER CE80 CH 1; 25747/FINAL SER ABWR CH 1

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**Integrated Impact Number:** 269      **SRP Section Number:** 3.10

#### **Suggested Changes to the SRP Section:**

This integrated impact has been partially processed only (see Part D, Cons:). Future action for the unprocessed part will be tracked by IPD 7.0 Form No. 3.10-2.

Revise Acceptance Criteria, 1.a., of SRP 3.10, to address the concerns of Generic Issue 113.

Generic Issue 113 addresses the need for requirements for dynamic qualification and testing of large bore hydraulic snubbers (LBHS). In a memorandum for J. Norberg from R. Baer, "Recommendations for SRP Revisions Related to Snubbers," May 5, 1992, RES proposed changes as part of the resolution of the issue. In summary, testing should be performed to determine the operational characteristics of the snubber control valve associated with LBHS. This testing is in addition to the dynamic tests required for mechanical and hydraulic snubbers.

Consider revising 1.a. of Acceptance Criteria to incorporate RES recommendations on design and testing of snubbers including LBHS.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23211/NRC MEMORANDUM 05/05/92, from Robert L. Baer to James A. Norberg

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**Integrated Impact Number:** 273      **SRP Section Number:** 3.10

#### **Suggested Changes to the SRP Section:**

Add a discussion of stress limits relative to active pump and valve operability to Acceptance Criteria.

The staff presented additional criteria for ensuring the operability of active pumps and valves by limiting the allowable stresses to material elastic limits when the component is subjected to the combination of normal operating loads, seismic loads and dynamic system loads. Implementation of the criteria will provide assurance that valve bodies or pump cases will not distort to the extent that operability of the component is impaired.

Consider modifying Acceptance Criteria to include a discussion of the staff's position on stress limit criteria relative to active pump and valve operability.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

23215/DRAFT FINAL SER ABWR CH 3

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**Integrated Impact Number:** 278      **SRP Section Number:** 3.5.1.4

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to support review of the design-basis tornado missiles for evolutionary plants.

SRP Section 3.5.1.4 currently identifies Reg. Guide 1.76 as guidance for the selection of design-basis tornado parameters and identifies acceptable design-basis missile spectra for current reactors. Design-basis tornado maximum wind speeds are used in the calculation of missile velocities to be assumed for design-basis missiles. In SECY 93-087, the staff concluded that it was acceptable to reduce the design-basis tornado wind speed to 322 km/hr (200 mph) west of the Rocky Mountains and to 482 km/hr (300 mph) east of the Rocky Mountains. The EPRI Evolutionary Plant FSER and the ABWR DFSE also describe staff positions reducing the maximum wind speeds which must be assumed for the design-basis tornado. In an SRM dated July 21, 1993, the Commission approved the staff's position that a maximum wind speed of 482 km/hr (300 mph) be used in the design-basis tornado employed in the design of evolutionary plants. The reduced design-basis tornado wind speeds result in reduced velocities to be assumed for evolutionary plant design-basis missiles.

Consideration should be given to revising Acceptance Criteria and Review Procedures to support review of the design-basis tornado missiles for evolutionary plants, based upon the above staff positions. Consideration should also be given to revising the guidance for design-basis tornado selection in RG 1.76 as a future work item. This future work recommendation will be tracked with IPD 7.0 form numbers 3.5.1.4-1 and 3.5.1.4-2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20729/RG 1.76; 22633/FINAL SER ABWR CH 3; 22635/SECY 93-087; 22636/FINAL SER EPRI CH 1; 24591/NRC MEMORANDUM 03/18/94, from G. Bagchi to L. Cunningham

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**Integrated Impact Number:** 279      **SRP Section Number:** 3.5.1.4

#### **Suggested Changes to the SRP Section:**

Add Review Procedures for Regulatory Guide 1.76.

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The current Review Procedures do not discuss the use of Regulatory Guide 1.76 for the staff's review. In addition, the Evaluation Findings should be modified to state that applicants should meet the guidelines of Position C.1 or C.2 of Regulatory Guide 1.76, but not both since they are mutually exclusive.

Consider changes to Review Procedures and Evaluation Findings to address Regulatory Guide 1.76 positions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20729/RG 1.76

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**Integrated Impact Number:** 280      **SRP Section Number:** 3.5.1.4

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to address the use of probabilistic criteria in evaluating the need to provide missile protection.

The ABWR FSER documents a review method utilizing a combined probability to determine the statistical significance of an identified missile. Once a potential missile is identified its statistical significance is determined by calculating the combined probability of missile occurrence, impacting a significant target, and causing significant damage. If the combined probability is less than  $10^{-7}$  per year, the missile is not considered significant, if the combined probability is greater than  $10^{-7}$  per year, missile protection of safety-related SSCs is provided by one or more of the following: (1) locating the system or component in a missile-proof structure, (2) separating redundant systems or components for the missile path or range, (3) providing local shields and barriers for systems and components, (4) designing the equipment to withstand the impact of the most damaging missile, (5) providing design features to prevent the generation of missiles, (6) orienting missile sources to prevent missiles from striking equipment important to safety.

Consideration should be given to adding a new Review Procedure to address the use of probabilistic criteria in evaluating the need to provide missile protection.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23252/NRC MEMORANDUM 11/07/83, from Harold Denton to Victor Stello; 24295/FINAL SER ABWR CH 3

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### **Integrated Impacts**

**Integrated Impact Number:** 281      **SRP Section Number:** 3.5.2

#### **Suggested Changes to the SRP Section:**

Revise SRP Section 3.5.2 as necessary to reflect additional guidance associated with the protection of safety-related equipment from failures of non-safety-related structures, systems, and components (SSCs) due to tornado effects.

The SRP currently cites Reg. Guide 1.117, "Tornado Design Classification," as guidance and describes a review of non-safety-related SSCs to be provided missile protection to address the potential for secondary missiles. Regulatory Guide 1.117 provides guidance for identifying those SSCs that should have tornado protection. The Regulatory Guide's list of SSCs to be protected includes those portions of SSCs whose continued function is not required but whose failure could reduce to an unacceptable safety level the functional capability of any plant features which should be tornado protected or could result in incapacitating injury to occupants of the control room. The SRP does address a review of non-safety-related SSCs to be protected from externally generated missiles if as a result of their failure by a missile the consequences could prevent safety-related SSCs from performing their intended function.

The ABWR FSER provides information on external missiles resulting from the failure of non-safety-related SSCs not housed in tornado-resistant buildings or structures. Any failure of non-safety-related SSCs that may result in external missile generation should not prevent safety-related SSCs from performing their intended safety function.

As indicated in the ABWR FSER those applicants referencing a certified design are responsible for the design of SSCs outside of the design certification scope. Any failure of SSCs outside of the design certification scope that may result in external missile generation should not prevent safety-related SSCs from performing their intended safety function.

Consideration should be given to revising SRP Section 3.5.2 as necessary to reflect additional guidance associated with the protection of safety-related equipment from missiles generated as a result of failures of non-safety-related SSC due to tornado effects.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20749/RG 1.117; 22684/FINAL SER ABWR CH 3

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### **Integrated Impacts**

**Integrated Impact Number:** 282      **SRP Section Number:** 4.5.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for austenitic stainless steel to address staff positions more restrictive than RG 1.44.

RG 1.44 is cited in SRP 4.5.1 for control of the use of sensitized stainless steel. The SRP cites RG 1.44 for protection from contamination during handling and storage, verification of non-sensitization of the material, and qualification of welding processes employed in production using ASTM A-262.

NRC Generic Letter 88-01 regarded implementation of new staff positions covering technical areas related to intergranular stress corrosion cracking in BWR austenitic stainless steel piping. GL 88-01 also transmitted Revision 2 to NUREG-0313, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." The resolution to New Generic Issue 119.4, "BWR Piping Materials," notes that updating of RG 1.44 to reflect the staff's findings in NUREG-0313, Rev. 2 is recommended.

Consider revising Review Procedures for austenitic stainless steel to address staff positions more restrictive than RG 1.44 as outlined in NUREG 0313, Rev. 2. In addition, consideration should be given to a revision of RG 1.44 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-6.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1086/RG 1.44; 23363/NRC GENERIC LETTER 88-01; 23374/NUREG 0933; 23412/NUREG 0313

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**Integrated Impact Number:** 283      **SRP Section Number:** 6.2.7

#### **Suggested Changes to the SRP Section:**

Revise code references such that the SRP citations are not version specific and reflect the appropriate subsection articles. SRP Section 6.2.7 reviews fracture prevention of the reactor containment pressure boundary materials, which consists of those ferritic steel parts of the reactor containment that sustain loading and provide a pressure boundary in the performance of the containment function.

Section III of the ASME Boiler and Pressure Vessel Code, Summer 1977 Addenda, Subsection

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NC, is cited in SRP Section 6.2.7 to address mandatory fracture toughness testing of ASME Code, Section III Class 2 materials. The NRC staff is actively involved in the development of the Code and has a high degree of familiarity with the Code and its evolution. The version of the Code acceptable to the NRC for any particular application is established in 10 CFR 50.55a.

Subsection NE, "Class MC Components," establishes the ASME Code Section III requirements for metal containment vessels. Within Subsection NE, Article NE-2300 provides the fracture toughness requirements for Class MC component materials.

In the CE80+ FSER, the staff states that the containment vessel materials are in accordance with Article NE-2000.

Consideration should be given to revising SRP Section 6.2.7 such that the references to ASME Code Section III are not specific to a particular version and reflect the appropriate subsection articles.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22860/C&S: ASME III; 23092/FINAL SER CE80 CH 3; 24542/NUREG 0800

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**Integrated Impact Number:** 284      **SRP Section Number:** 3.5.1.6

#### **Suggested Changes to the SRP Section:**

SRP Section 3.5.1.6 provides review of the CP/OL applicant's assessment of aircraft hazards to assure that the risks for the site are sufficiently low.

10 CFR 52 provides requirements for site parameter envelopes that are to be included in applications for design certifications and manufacturing licenses for standard plant designs. Applications which reference standard plant designs approved under 10 CFR 52 are required to address the conformance of site specific parameters with the site parameter envelope for the approved, certified standard design. In SERs documenting staff review of evolutionary plant applications for design certification, the staff addressed the requirements related to the site parameter envelope in SER Section 2.6.

Consideration should be given to developing a new SRP Section (See IPD-7.0 Form # 2.3.1-1) for review of the site parameter envelope associated with standard plant applications, as a candidate for future work. Consideration should also be given to revising existing SRP Sections, including SRP Section 3.5.1.6, for review of site-specific parameters to reflect the site parameter-related requirements of 10 CFR 52, for applications referencing a standard plant design.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

23261/CODE OF FED. REGS 10CFR52; 23262/FINAL SER CE80 CH 2; 25621/FINAL SER ABWR CH 3; 25622/FINAL SER ABWR CH 2

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**Integrated Impact Number:** 286      **SRP Section Number:** 6.2.1.1.A

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to indicate that construction permit stage containment design margin criteria should be applied to reviews performed for design certifications.

Specific criterion "a" of the Acceptance Criteria subsection states that the requirements of GDC 16 and 50 regarding sufficient design margin are satisfied, for plants at the construction permit (CP) stage of review, if the containment design pressure provides at least a 10% margin above the accepted peak calculated containment pressure following a loss-of-coolant accident, or a steam or feedwater line break. It also allows that, for plants at the operating license (OL) stage of review, revised or upgraded analytical models or minor changes in the as-built design of the plant may result in a decrease in that margin.

In the EPRI Evolutionary Plant FSER, the staff approved EPRI's deletion of the 10% margin criterion for peak containment pressure on the basis that this margin is only applicable to the CP stage of a licensing review and the information required for the ALWR submittal will be that required in a final safety analysis report (OL) stage review. In the later CE80+ DSER, the staff did not accept the applicant's statement that the single step licensing process for System 80+ precludes considering this review to be at the CP stage. The staff stated that CE's proposed margin (1%) in calculated peak internal primary containment pressure did not meet SRP Section 6.2.1.1.A and GDC 50 and that the applicant must justify the adequacy of this margin. (In the ABWR ASER for the review related to SRP Section 6.2.1.1.C, the staff reviewed and approved the final ABWR containment design to the SRP 6.2.1.1.C CP stage criteria for Mark III plants.)

Specific criterion "f" of the Acceptance Criteria subsection states that, to satisfy the requirements of GDC 38 and 50, if the containment is designed to withstand the maximum external pressure, such a design should provide adequate margin above that pressure. In the CE80+ DSER, the staff stated that the applicant must justify the adequacy of its proposed margin (8.5%) in calculated peak external primary containment pressure because this margin did not meet a 10% criterion.

Consider modifying Review Procedures to indicate that construction permit stage containment design margin criteria should be applied to reviews performed for design certifications.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22985/FINAL SER EPRI CH 5; 22986/FINAL SER CE80 CH 6; 23279/FINAL SER ABWR CH 6

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**Integrated Impact Number:** 287      **SRP Section Number:** 6.2.1.1.B

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to indicate that construction permit stage containment design margin criteria should be applied to reviews performed for design certifications.

Specific criterion II.1 of SRP Section 6.2.1.1.B states that, at the construction permit (CP) stage, the containment design pressure should provide at least a 20% margin above the highest calculated accident pressure; and, for plants under review for Operating Licenses (OL), the highest calculated accident pressure should not exceed the design pressure of the containment. SRP Sections 6.2.1.1.A and 6.2.1.1.C contain similar distinctions between CP stage and OL requirements. However, SRP 6.2.1.1.A and 6.2.1.1.C also state that, at the OL stage, it is expected that the peak calculated pressure should be approximately the same as at the construction permit stage.

In the EPRI Evolutionary Plant FSER, the staff approved EPRI's deletion of the 10% criterion for peak containment pressure on the basis that this margin is only applicable to the CP stage of a licensing review and the information required for the ALWR submittal will be that required in a final safety analysis report (OL) stage review. In the later CE80+ FSER, the staff did not accept the applicant's statement that the single step licensing process for System 80+ precludes considering this review to be at the CP stage. The staff stated that CE's proposed 1% margin in calculated peak internal primary containment pressure did not meet SRP Section 6.2.1.1.A and GDC 50 and that the applicant must justify the adequacy of this margin. In the ABWR FSER, the staff reviewed and approved the final ABWR containment design to the SRP 6.2.1.1.C CP stage criteria for Mark III plants.

Consider modifying Review Procedures to indicate that construction permit stage containment design margin criteria should be applied to reviews performed for design certifications.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22864/FINAL SER EPRI CH 5; 22865/FINAL SER CE80 CH 6; 23280/FINAL SER ABWR CH 6

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### **Integrated Impacts**

**Integrated Impact Number:** 288      **SRP Section Number:** 6.2.1.1.C

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to indicate that construction permit stage containment design margin criteria should be applied to reviews performed for design certifications.

Specific criterion "1" of the Acceptance Criteria subsection states that, for Mark III plants at the construction permit stage, the containment design should provide at least a 15% margin above the peak calculated containment pressure, and the design differential pressure between drywell and containment should provide at least a 30% margin above the peak calculated differential pressure. For Mark I, II and III plants at the operating license stage, the peak calculated containment pressure and differential pressure should be less than the design pressure. In general, it is expected that the peak calculated pressure should be approximately the same as at the construction permit stage. However, it is possible that the margins may be affected by revised or improved analytical models, test results or minor changes in the as-built design of the plant.

In the EPRI Evolutionary Plant FSER, the staff approved EPRI's deletion of the construction permit criterion on the basis that this margin is only applicable to the CP stage of a licensing review and the information required for the ALWR submittal will be that required in a final safety analysis report (OL) stage review. In the later ABWR FSER, the staff reviewed and approved the final ABWR containment design to the SRP 6.2.1.1.C CP stage criteria for Mark III plants. (It should also be noted that in the review of the CE80+ design which is covered by SRP Section 6.2.1.1.A, the staff did not accept the applicant's statement that the single step licensing process for System 80+ precludes considering this review to be at the CP stage. The staff stated that CE's proposed 1% margin in calculated peak internal primary containment pressure did not meet SRP Section 6.2.1.1.A and GDC 50 and that the applicant must justify the adequacy of this margin.)

Consider modifying Review Procedures to indicate that construction permit stage containment design margin criteria should be applied to reviews performed for design certifications.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22992/FINAL SER EPRI CH 5; 22993/FINAL SER ABWR CH 6; 23090/FINAL SER CE80 CH 6

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**Integrated Impact Number:** 291      **SRP Section Number:** 3.9.5

#### **Suggested Changes to the SRP Section:**

Add Review Procedures for verifying that design fatigue curves for reactor vessel internals include adequate conservatism to address environmental effects upon materials. SRP Section 3.9.5 Areas of Review includes the design bases for the mechanical design of the reactor vessel internals, including fatigue limits.

In the EPRI Evolutionary Plant FSER, the staff identified concerns relating to possible detrimental environmental effects not currently reflected in ASME Code design fatigue curves. Recent data indicate that ASME Code design fatigue curves may not be as conservative as originally intended. The staff stated a position that, until these curves are revised, ALWR applicants should propose appropriate fatigue design curves that will be reviewed by the staff.

Subsequently, in the FSERs documenting staff review of the ABWR and CE System 80+ applications for design certification, the staff accepted proposed design fatigue curves for reactor internals which were based upon current ASME Code design fatigue curves and commitments for further analysis of environmental effects upon the fatigue resistance of materials. For the CE System 80+ design, the applicant demonstrated to the satisfaction of the staff that the environmental conditions of greatest concern were generally not applicable to the design and were thus not likely to result in significant environmental degradation of components, including reactor vessel internals.

Consideration should be given to adding Review Procedures for verifying that design fatigue curves for reactor vessel internals include adequate conservatism to address environmental effects upon materials.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22562/FINAL SER EPRI CH 1; 22822/FINAL SER CE80 CH 3; 24294/FINAL SER ABWR CH 3

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**Integrated Impact Number:** 292      **SRP Section Number:** 3.9.5

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to clarify that asymmetric loading conditions should be considered in the design of the RPV internals. SRP 3.9.5 Areas of Review includes the loading conditions that provide the basis for the design of the RPV internals to sustain postulated accidents.

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USI A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant Systems," was established to determine the impact of a postulated LOCA at the vessel nozzle that results in an asymmetric loading on the reactor internals, and was resolved in 1981 with the publication of NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems." NUREG-0609 specifies that the asymmetric loads on the RPV internals should not exceed the limits imposed by the applicable codes and standards. As discussed in Generic Letter 84-04 and allowed by a 1986 revision of GDC 4, plants meeting specified criteria may utilize leak-before-break (LBB) analyses to justify exclusion from design bases of certain asymmetric loadings on the reactor vessel and internals which would result from pipe breaks at locations where such postulated breaks are demonstrated to be extremely unlikely to occur. In the CE 80+ FSER, the staff accepted the applicant's LBB approach and the applicant's approach to accommodating effects of asymmetric blowdown loads associated with the resultant reduced range of break locations/sizes postulated for the design.

Consider modifying Review Procedures to clarify that asymmetric loading conditions should be considered in the design of the RPV internals.

This Integrated Impact was revised to reflect the staff's acceptance of the resolution of Issue A-2 for the CE System 80+ design as documented in the FSER. This Integrated Impact was also revised to reflect the correct title of USI A-2, based upon NUREG-0933. The title of NUREG-0609 was added, primarily to reflect that the issue resolution only affected PWR designs.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1465/NUREG 0933; 21035/CODE OF FED. REGS 10CFR50; 22824/FINAL SER CE80 CH 20; 22825/NRC GENERIC LETTER 84-04; 22826/NUREG 0609

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**Integrated Impact Number: 293      SRP Section Number: 6.6**

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to provide guidance for application of ASME Section XI, Subsection IWH for examination of piping susceptible to wall thinning due to erosion/corrosion.

Serious pipe ruptures at two plants (one leading to fatalities) were attributed to pipe wall thinning due to erosion/corrosion in single phase fluid systems. The subsequent investigation determined that this issue should be treated generically and Generic Issue 139 was established. Bulletin 87-01 requested that licensees describe the scope and extent of their programs for ensuring that pipe wall thicknesses are not reduced below the minimum allowable thickness. Generic Letter 89-08 discussed a long term erosion/corrosion monitoring program that provides assurance that procedures or administrative controls are in place to assure that the structural



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integrity of all high-energy carbon steel systems (containing two phase as well as single phase flow) is maintained.

In the CE 80+ FSER, the staff indicated that ASME Class 1, 2, and 3 carbon and low-alloy-steel piping items that are susceptible to wall thinning due to the single-phase (water) erosion-corrosion phenomenon will be subject to examination in accordance with Subsection IWH of ASME Section XI. The staff noted that Subsection IWH had not been incorporated into the Code at the time of their review. Where the design will not practically accommodate full adherence to IWH, alternatives will be proposed in accordance with 10 CFR 50.55a(a)(3).

In the ABWR FSER, the staff acknowledged that the applicant has committed to a monitoring program for erosion-corrosion that provides assurances that procedures or administrative controls are in place to assure that the structural integrity of all high-energy carbon-steel systems (containing two-phase as well as single-phase flow) is maintained as discussed in Generic Letter (GL) 89-08 and NUREG-1344, "Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants," April 1989.

Consider modifying Review Procedures to incorporate the provisions of Bulletin 87-01 and Generic Letter 89-08, and provide guidance for application of ASME Section XI, Subsection IWH for examination of piping susceptible to wall thinning due to erosion/corrosion.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14705/NUREG 0933; 22544/FINAL SER CE80 CH 6; 22772/NRC BULLETIN 87-01;  
22774/NRC GENERIC LETTER 89-08; 23305/FINAL SER ABWR CH 3

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**Integrated Impact Number:** 294      **SRP Section Number:** 6.6

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to provide guidance for application of ASME Section XI, Appendices VII and VIII for qualification of personnel and performance demonstration of ultrasonic examination systems.

The ASME has published in ASME Section XI, (Division 1), Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." In both the ABWR FSER and the CE 80+ FSER, the staff indicates that the NRC has published in the Federal Register its intent to reference in 10 CFR 50.55a(b) the ASME Section XI edition that includes the published Appendix VII. In addition, the NRC staff has established a technical contact to coordinate the implementation of Appendix VIII. In the ABWR FSER, the staff requested that the applicant

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include provisions that ultrasonic testing be performed in accordance with Appendices VII and VIII pursuant to 10 CFR 50.55a(g)(3). In the CE 80+ FSER, the staff indicated that the applicant should indicate that the Section XI requirements are to be augmented with the requirements in Appendices VII and VIII.

Consider modifying Review Procedures to review application of ASME Section XI, Appendices VII and VIII for qualification of personnel and performance demonstration of ultrasonic examination systems.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22531/FINAL SER ABWR CH 6; 22543/FINAL SER CE80 CH 6

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**Integrated Impact Number: 295      SRP Section Number: 6.6**

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures to address augmented inspection of austenitic stainless steel piping in BWRs.

Generic Safety Issue A-42 was established to study the safety implications of the significant number of cracks being discovered in heat-affected zones of austenitic stainless steel piping in BWRs. The major problem was recognized to be intergranular stress corrosion cracking (IGSCC). This issue was resolved in February, 1981 and the technical results of this task were published in NUREG-0313, Revision 1. Generic Letter 81-03 transmitted NUREG-0313, Revision 1 to all holders of and applicants for BWR operating licenses or construction permits. Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," transmits NUREG-0313, Revision 2. The letter states that varying amounts of augmented inspections are specified for piping with a greater susceptibility to cracking, where there is less certainty about the effectiveness of mitigation measures used, or in cases where repairs have been performed. Generic Letter 88-01, Supplement 1 provides acceptable alternative staff positions to some of those delineated in Generic Letter (GL) 88-01.

Consider developing Review Procedures to address augmented inspection of austenitic stainless steel piping in BWRs. Consider adding a reference to NUREG-0313, Revision 2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

7648/NRC GENERIC LETTER 81-03; 7662/NRC GENERIC LETTER 88-01; 11084/NUREG 0933; 21799/NRC GENERIC LETTER 88-01 SUP 1; 22778/NUREG 0313

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### **Integrated Impacts**

**Integrated Impact Number:** 296      **SRP Section Number:** 5.4.12

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and References to add a citation for 10 CFR 50.44(c) as a source of Acceptance Criteria for Reactor Coolant System High Point Vents.

10 CFR 50.44 establishes standards for combustible gas control system in light-water-cooled power reactors. Paragraph (c)(3)(iii) describes requirements relative to reactor coolant system high point vents.

SRP Section 5.4.12 currently cites TMI Action Plan Item II.B.1, related to 10CFR50.34(f)(2)(vi), which establishes requirements similar to 10CFR50.44. As a result of comparing the statements in 10CFR50.44(c)(3)(iii) with those in SRP 5.4.12 acceptance criteria and TMI Action Plan Item II.B.1, it does not appear that changes to existing review procedures would be necessary.

Consideration should be given to revising the Acceptance Criteria and References of Section 5.4.12 to add appropriate citations of 10 CFR 50.44(c).

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20016/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 297      **SRP Section Number:** 5.4.12

#### **Suggested Changes to the SRP Section:**

Modify Acceptance Criteria, Review Procedures, and References to cite 10 CFR 50.49 and to cite IEEE-344 as endorsed by Regulatory Guide 1.100.

SRP 5.4.12, specific Acceptance Criterion 10 and Review Procedures step 10 cite IEEE Std 344-1975, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," as supplemented by Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants." Revision 2 of Regulatory Guide 1.100 endorses a later version of the standard, IEEE Std 344-1987.

Specific Acceptance Criterion 10 also states that environmental qualifications are in accordance with the May 23, 1980 Commission Order and Memorandum (CLI-80-21). 10 CFR 50.49 which was published in the Federal Register 48 FR 2733, Jan. 21, 1983, appears to be a more appropriate reference.

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Consideration should be given to modifying specific Acceptance Criterion 10 and Review Procedures step 10 to cite IEEE-344-1987 as endorsed by Regulatory Guide 1.100. Consideration should also be given to replacing the reference to CLI-80-21 with a reference to 10 CFR 50.49.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20020/CODE OF FED. REGS 10CFR50; 22906/C&S: IEEE 344

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**Integrated Impact Number: 298      SRP Section Number: 15.8**

#### **Suggested Changes to the SRP Section:**

Include 10 CFR 50.62 as Acceptance Criteria.

GSI A-9, "Anticipated Transient Without Scram (ATWS)", was resolved (for reactor designs evaluated at that time) with the promulgation of the ATWS Rule, 10 CFR 50.62, on June 26, 1984. 10 CFR 50.62 establishes requirements applicable to the reviews currently contemplated for SRP Section 15.8.

Consideration should be given to incorporating 10 CFR 50.62 as Acceptance Criteria.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1499/NUREG 0933; 11159/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 299      SRP Section Number: 15.8**

#### **Suggested Changes to the SRP Section:**

Include Review Procedures based upon the staff position on ATWS consequences for new plants/designs.

In SECY 90-016, the staff requested that the Commission approve a staff position that diverse scram systems be provided for evolutionary ALWRs. In the SRM for SECY 90-016, the Commission approved the staff's position but indicated that the staff should accept either a diverse scram system or alternatively a demonstration that the consequences of an ATWS event are acceptable. The position that would require a diverse scram system relates primarily to Westinghouse PWR designs which are not explicitly required to provide diverse scram systems under the ATWS rule.

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During review of the ABWR and CE System 80+ designs, the staff determined that diverse scram systems were provided in accordance with the ATWS rule. The staff also encouraged the applicants to provide new ATWS analyses. In the FSERs for the ABWR and CE System 80+ designs, the staff found that the designs comply with the ATWS rule, include added preventive and mitigative features exceeding the requirements of the rule, and are supported by ATWS analyses. CE System 80+ analyzed consequences were found by the staff to be bound by previous generic evaluations upon which the ATWS rule was based (detailed in SECY 83-293), even when no credit is taken for required diverse scram systems.

Consideration should be given to incorporating the staff position on evolutionary plant ATWS consequences in Review Procedures.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21792/FINAL SER EPRI CH 10; 23028/SECY 90-016; 24426/FINAL SER ABWR CH 15; 24428/FINAL SER CE80 CH 15; 24570/SECY 83-293

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**Integrated Impact Number:** 300      **SRP Section Number:** 15.8

#### **Suggested Changes to the SRP Section:**

Include Review Procedures for BWR stability during/following an ATWS event.

Loss of recirculation flow resulting from recirculation pump trips (in BWRs) creates potential for operation in thermal-hydraulically unstable regions. Bulletin 88-07 describes an event involving power oscillations following a loss of recirculation flow and indicates that the implications of the event require further study with respect to the effects of ATWS recirculation pump trips. Chapter 4, Section 4.2 of the EPRI Evolutionary Plant FSER indicates that the staff will review BWR applications to assure that the thermal-hydraulic stability performance of the core during an ATWS is acceptable. In the ABWR FSER, the staff documented its stability review in Sections 4.4 and 15.5.2 and found that based upon several new design features, the ABWR is more stable than most currently operating BWRs during/following an ATWS event.

Analyses serving as the basis for the staff's proposed ATWS rule (which was not adopted) had predicted significant oscillations following an ATWS event as described in NUREG-0460. Acceptable BWR stability performance (core coolability and containment integrity can be acceptably maintained in the event large oscillations occur) was apparently demonstrated in analyses serving as the basis for final the ATWS rule (See 59 FR 42182) and acceptable, qualified computer codes (e.g. TRACG) are available for estimating the global behavior of reactors during transients that may result in large power oscillations. The power oscillation event described in Bulletin 88-07 has resulted in further staff and BWR Owners Group

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(BWROG) evaluation of stability for all BWR operating conditions, including postulated ATWS events.

It should also be noted that the BWROG has provided an acceptable resolution of BWR stability issues associated with ATWS events. Acceptable resolution for power oscillations during/following ATWS events will be addressed in SRP Section 4.4 (see Integrated Impact 221).

Consideration should be given to incorporating Review Procedures for reviewing BWR stability during an ATWS.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11220/NRC BULLETIN 88-07; 21789/FINAL SER EPRI CH 4; 21793/NUREG 0460 VOL 1; 21794/NUREG 0460 VOL 2; 21795/NUREG 0460 VOL 3; 21796/NUREG 0460 VOL 4; 24426/FINAL SER ABWR CH 15

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**Integrated Impact Number:** 301      **SRP Section Number:** 15.8

#### **Suggested Changes to the SRP Section:**

Include Review Procedures and criteria as appropriate for determining compliance with the ATWS Rule and its underlying bases.

GSI A-9, ATWS, was resolved for current reactors with the issuance of the ATWS Rule, 10 CFR 50.62. The final ATWS Rule was based on NRC task force recommendations provided in a report to the Commission (SECY 83-293). The task force considered several final rule options, including a proposed rule based upon staff evaluation of ATWS and recommendations provided in NUREG-0460, and a proposed rule sponsored by a Utility Group which related to a generic PRA assessment of the ATWS issue for each NSSS vendor. Based upon task force recommendations and value/impact evaluations enclosed in SECY 83-293, the Commission adopted a modified version of the Utility Group proposed rule (prescriptive, vendor-specific mitigative design requirements) in the final ATWS rule. The task force report and NUREG-0460 provide information relevant to the review of proposed design features and information submitted pursuant to the ATWS Rule for current reactors.

Consideration should be given to including Review Procedures and criteria as appropriate for determining compliance with the ATWS Rule and its underlying bases.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

1499/NUREG 0933; 11159/CODE OF FED. REGS 10CFR50; 21793/NUREG 0460 VOL 1; 21794/NUREG 0460 VOL 2; 21795/NUREG 0460 VOL 3; 21796/NUREG 0460 VOL 4; 24570/SECY 83-293

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**Integrated Impact Number: 302      SRP Section Number: 15.8**

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures and associated Areas of Review regarding ATWS analysis.

GSI A-9, ATWS, was resolved for current reactors with the issuance of the prescriptive ATWS Rule, 10 CFR 50.62. The final ATWS Rule was based on NRC task force recommendations enclosed in a report to the Commission (SECY 83-293). The task force considered several final rule options, including a proposed rule based upon staff evaluation of ATWS and recommendations provided in NUREG-0460, and a proposed rule sponsored by a Utility Group which related to a generic PRA assessment of the ATWS issue for each NSSS vendor. Based upon task force recommendations and value/impact evaluations, the Commission adopted a modified version of the Utility Group proposed rule (prescriptive, vendor-specific mitigative design requirements) in the final ATWS rule.

The EPRI Evolutionary Plant FSER describes a staff position that the designer demonstrate the bases on which the ATWS rule was developed remain valid for the proposed design and that the new plant will not experience ATWS behavior more severe than in current plants. This position implies that the staff will review specific ATWS analyses for new design applications. ATWS analyses were requested by the staff and were reviewed for both the CE80+ and the ABWR design certification applications. Currently, the SRP does not describe a review of ATWS analyses.

The task force report and NUREG-0460 describe PRAs and analyses considered during the development of the ATWS Rule including the severity of ATWS behavior for current plants and the evaluations performed for these plants. It appears that the task force report and NUREG-0460 establish bases for reviewing the acceptability of ATWS analyses for new designs.

Consideration should be given to developing ATWS analyses Review Procedures for new designs including a description of the applicability of the task force report and NUREG-0460 to this review. If this review is incorporated, Areas of Review should also be revised to reflect a review of ATWS analyses.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

11159/CODE OF FED. REGS 10CFR50; 21787/FINAL SER EPRI CH 1; 21792/FINAL SER EPRI CH 10; 21793/NUREG 0460 VOL 1; 21794/NUREG 0460 VOL 2; 21795/NUREG 0460 VOL 3; 21796/NUREG 0460 VOL 4; 24425/FINAL SER ABWR CH 15; 24426/FINAL SER ABWR CH 15; 24428/FINAL SER CE80 CH 15; 24570/SECY 83-293

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**Integrated Impact Number:** 303      **SRP Section Number:** 4.5.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address staff positions that are more restrictive than RG 1.31.

The SRP uses RG 1.31 as specific criteria for assuring integrity of welds in stainless steel components. Review Procedures specify RG 1.31 for measuring and controlling the ferrite content in stainless steel weld metal. RG 1.31 specifies ferrite content limits between 5 and 20% for weld metal.

Generic Letter 88-01 transmitted NUREG-0313, Rev. 2 and provides positions related to welding which are applicable to current generation BWRs based upon the recommendations of NUREG-0313, Rev. 2.

In the CE System 80+ FSER, in conjunction with reviews of control rod drive structural and ESF materials, the staff compared the ferrite content limits for austenitic stainless steel castings and weld metal against industry guidelines and NUREG-0313, Rev. 2. NUREG-0313, Rev. 2 identifies a lower ferrite content limit of 7.5% which is more restrictive than the lower limit specified in RG 1.31.

Consider revising Review Procedures for austenitic stainless steel to address staff positions outlined in NUREG-0313, Rev. 2 and industry guidelines. In addition, consideration should be given to revision of RG 1.31 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-7.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1083/RG 1.31; 22225/FINAL SER CE80 CH 4; 23363/NRC GENERIC LETTER 88-01; 23412/NUREG 0313

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**Integrated Impact Number:** 304      **SRP Section Number:** 4.5.1

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of the acceptability of Nickel-Chromium-Iron alloys (e.g., Inconel) as materials for control rod drive mechanism components.

The EPRI Evolutionary Plant FSER provides a staff position that alloys such as Inconel 600 are susceptible to primary water stress corrosion cracking and should generally not be used in applications where stress corrosion cracking is a concern. The staff also identified alternative alloys considered acceptable with respect to stress corrosion cracking in conjunction with these positions. In addition, the staff accepted EPRI proposed requirements that the use of Inconel Alloy-X-750 be in accordance with EPRI NP-6202, "Material specification for Alloy-X-750 in LWR Internal Components." The staff indicated that improperly heat treated Alloy-X-750 is susceptible to cracking and that EPRI NP-6202 provides the technical basis for the use of this alloy.

In the CE80+ FSER, the staff notes that the applicant plans to use Inconel 690 in lieu of 600 in the fabrication of the motor housing assembly. The staff views the Inconel 690 alloy as the preferred nickel base alloy in the primary and secondary coolant loops because of its improved corrosion resistance compared to Inconel 600. Inconel X-750 (AMS 5698B and AMS 5699B) will be used in the CE System 80+ design for CEDM springs that will be subjected to elevated temperatures and must resist relaxation. Prototype testing of the CEDM and extensive previous experience in existing reactors have demonstrated the acceptability of these materials for their intended use. The staff reviewed the processing controls for these materials and found that the materials should perform their function when supported by the maintenance and inspections prescribed by the applicant.

Consider developing Review Procedures for review of the acceptability of Nickel-Chromium-Iron alloys (e.g., Inconel) as materials for control rod drive mechanism components.

Also consider revising the Review Procedures to identify that EPRI NP-6202 provides an acceptable basis for the use of Inconel X-750.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22222/FINAL SER CE80 CH 4; 22290/FINAL SER EPRI CH 1; 25344/FINAL SER CE80 CH 4

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**Integrated Impact Number:** 305      **SRP Section Number:** 4.5.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for the control rod drive component and system cleanliness and cleanliness controls.

The SRP currently cites RG 1.37 with regard to cleanliness and cleanliness controls. RG 1.37 cites ANSI N45.2.1 in describing the staff's position with regard to cleanliness and cleanliness controls. In the CE 80+ FSER, the staff indicated that ANSI/ASME NQA-2 supersedes ANSI N45.2.1, cited RG 1.37, and indicated that they had reviewed ANSI/ASME NQA-2-1983 and found it acceptable.

Consideration should be given to revising existing Review Procedures related to cleanliness and cleanliness controls to cite ANSI/ASME NQA-2-1983 in addition to RG 1.37. In addition, consideration should be given to revision of RG 1.37 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-3.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

Per 11/94 conversation with Quality Assurance and Maintenance Branch staff, N45.2.1 requirements are being incorporated into NQA-1 and NQA-2. RG 1.28, Revision 3 endorsed NQA-1. NRC has a program to revise the endorsement based on the results of an evaluation of the graded QA program. Also NQA is going through a review of both standards.

Per 11-10-94 telecon with Office of Research staff, two draft regulatory guides were prepared to endorse NQA-1 and NQA-2 through their 1993 addenda. Both regulatory guides were put on hold due to NRC/NEI work on the graded QA program. In the interim, NQA-1 and NQA-2 were consolidated into a new NQA-1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1081/RG 1.37; 22221/FINAL SER CE80 CH 4; 22285/C&S: ANSI N45.2.1; 23428/C&S: ASME NQA-2

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**Integrated Impact Number:** 307      **SRP Section Number:** 4.5.1

#### **Suggested Changes to the SRP Section:**

ASTM A-262, pertaining to detecting susceptibility to intergranular attack in stainless steel, is cited in conjunction with RG 1.44 in SRP Section 4.5.1 Review Procedure III.2. RG 1.37 is also

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cited in SRP Section 4.5.1 for cleaning and cleanliness control. ASTM A-262-68 is endorsed by RG 1.37 and ASTM A-262-70 is endorsed by RG 1.44, with exceptions. The current version of this standard is ASTM A262-1993. PNL has performed a detailed side-by-side comparison between cited and current versions of this standard.

Consider revising the reference to ASTM A-262 to specify the current 1993 version pending NRC review of the side-by-side comparison. Consideration should also be given to revising RGs 1.44 and 1.37 as future work items. The future work recommendations will be tracked with IPD 7.0 Forms number 4.5.1-1 (RG 1.44) and 4.5.1-4 (RG 1.37).

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1081/RG 1.37; 1086/RG 1.44; 22206/C&S: ASTM A262

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**Integrated Impact Number:** 310      **SRP Section Number:** 2.5.1

#### **Suggested Changes to the SRP Section:**

Upon final approval of the amendment to Appendix A, add a discussion of the amended version of 10 CFR Part 100 to Areas of Review, Acceptance Criteria, and Evaluation Findings.

The Regulation is presently being amended and has been entered into the Federal Register as a proposed rule on October 20, 1992. Incorporation of the changes in the amended version of 10 CFR Part 100 will require major modifications to all areas of SRP Section 2.5.1.

Consider updating SRP Section 2.5.1 to include the amended version of sections of 10 CFR Part 100 when the new rule is published.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

6331/CODE OF FED. REGS 10CFR100; 6332/CODE OF FED. REGS 10CFR100;  
6333/CODE OF FED. REGS 10CFR100; 6556/CODE OF FED. REGS 10CFR100;  
6560/CODE OF FED. REGS 10CFR100; 6561/CODE OF FED. REGS 10CFR100;  
6562/CODE OF FED. REGS 10CFR100; 6563/CODE OF FED. REGS 10CFR100;  
6567/CODE OF FED. REGS 10CFR100; 6568/CODE OF FED. REGS 10CFR100;  
6580/CODE OF FED. REGS 10CFR100; 6583/CODE OF FED. REGS 10CFR100;  
6600/CODE OF FED. REGS 10CFR100

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**Integrated Impact Number:** 312      **SRP Section Number:** 3.4.1

#### **Suggested Changes to the SRP Section:**

Modify Areas of Review, Acceptance Criteria, Evaluation Findings and References to include GDC 4.

SRP Section 3.4.1 addresses the review of a plant for flood protection from both internal and external causes. However, the Acceptance Criteria subsection does not specify acceptance criteria for protection from internal flooding.

GDC 4 includes requirements that structures, systems, and components important to safety be appropriately protected against dynamic effects, including the effects of discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit.

Consider modifying the Areas of Review, Acceptance Criteria, Evaluation Findings and References subsections to include GDC 4.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14890/NRC BULLETIN 80-24; 22952/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 313      **SRP Section Number:** 3.4.1

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to address the concerns of Unresolved Safety Issue A-17.

Unresolved Safety Issue A-17 is concerned with adverse systems interactions. In addition to providing systems-interaction-related input to NRC guidance on the content of PRAs for future LWRs, the staff recommended in NUREG-1229 that the SRP for future plants include specific guidance regarding protection from flooding and water intrusion from internal sources. Section 3.4.1 mentions internal flooding but does not provide specific guidance for considering it in the review.

Consider revising Acceptance Criteria and Review Procedures to provide specific guidance regarding protection from internal flooding and water intrusion events.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

1971/NUREG 0933; 7509/NUREG 0933; 15208/NRC GENERIC LETTER 89-18;  
16864/NUREG 0933; 23321/NUREG 1229

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**Integrated Impact Number:** 315      **SRP Section Number:** 3.9.1

#### **Suggested Changes to the SRP Section:**

Develop the 60-Year Design Life Review sub-section of SRP Section 3.9.1. SRP Section 3.9.1 Areas of Review includes transients, including the number of these transients expected to occur over the entire design life of the plant, which are used in the design and fatigue analysis of all Code Class 1 and CS components, and supports and reactor internals.

In section 3.9.1 of the DSER for CE 80+, the staff asked CE to justify the use of the same number of cycles for the System 80+ design (60-year design life) as was used for the System 80 design (40-year design life). CE committed to revise the CESSAR to include the events and frequency of occurrence expected during the 60-year design life.

In section 3.9.1 of the ABWR DFSE, the staff observed that the list of transients listed in the ABWR SSAR appeared to be based on a 40-year life. The staff required that, for a design life of 60 years, the number of cycles for each transient be increased by a factor of 1.5 and be applied to all safety-related systems and components.

Consideration should be given to developing the 60-Year Design Life Review sub-section of SRP Section 3.9.1 based on the positions contained in the current SRP (Acceptance Criteria, specific criterion 1) and the nature of the 60-year design life reviews described in the CE 80+ DSER and the ABWR DFSE.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22660/DRAFT FINAL SER ABWR CH 3; 22662/DRAFT SER CE80 CH 3; 22820/DRAFT SER CE80 CH 20

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**Integrated Impact Number:** 316      **SRP Section Number:** 5.4.6

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to include staff guidance regarding the operability of RCIC steam line isolation valves.

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NUREG-0933 Generic Issue 87, "Failure of HPCI Steam Line Without Isolation," describes a safety concern regarding a postulated break in a HPCI steam supply line and the uncertainty associated with the ability of HPCI steam supply line isolation lines to operate under those conditions. An unisolated HPCI steam line break would be a LOCA outside containment and could adversely affect other systems required to provide core cooling. GI-87 recognized that this situation could also occur in systems other than HPCI.

GI-87 was resolved and staff positions related to this issue are described in GL 89-10. GL 89-10 addresses a more general concern with the operation of safety-related MOVs under design-basis conditions. GL 89-10, Supplement 3 contains certain positions specific to ensuring the reliable operation of RCIC steam supply isolation valves in the event of a steam line break. In the ABWR DFSE, the staff indicated that, since closure of the RCIC isolation valves was assumed in the steam line break analysis, this functional requirement should be incorporated into the RCIC ITAAC. The staff also indicated that an applicant referencing the ABWR design should verify test data showing that the steam isolation valves will isolate under actual operating conditions.

Consideration should be given to modifying Review Procedures to include a review of the ability of RCIC steam line isolation valves to operate under design-basis conditions consistent with the positions identified above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

15351/NUREG 0933; 22889/FINAL SER ABWR CH 5; 22909/NRC GENERIC LETTER 89-10 Sup 3

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**Integrated Impact Number:** 317      **SRP Section Number:** 5.4.6

#### **Suggested Changes to the SRP Section:**

Add acceptance criteria and review procedures to address station blackout requirements.

As specified in the plant's station blackout coping analysis, the capability of the RCIC system to provide sufficient core cooling, independent of preferred and on-site emergency ac power, needs to be reviewed. Regulatory Position C.3 of RG 1.155 provides review guidance for this evaluation.

Consider adding 10 CFR 50.63 and RG 1.155 as acceptance criteria and developing applicable review procedures for the RCIC system.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

23322/CODE OF FED. REGS 10CFR50; 23323/RG 1.155; 24366/FINAL SER ABWR CH 5

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**Integrated Impact Number:** 318      **SRP Section Number:** 6.2.1.2

#### **Suggested Changes to the SRP Section:**

Modify Acceptance Criteria to make construction permit stage criteria applicable to a standard design certification review.

Acceptance Criteria II.B.5 of SRP Section 6.2.1.2 states that, at the construction permit stage, a factor of 1.4 should be applied to the peak differential pressure calculated for the subcompartment, structure and the enclosed components, for use in the design of the structure and the component supports. For Design Certification in the ABWR FSER, the staff evaluated the subcompartment analysis against this construction permit stage acceptance criteria.

Consider modifying Acceptance Criteria to make construction permit stage criteria applicable to a standard design certification review.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22851/FINAL SER ABWR CH 6

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**Integrated Impact Number:** 319      **SRP Section Number:** 6.2.1.5

#### **Suggested Changes to the SRP Section:**

Revise the Areas of Review, Acceptance Criteria, and Review Procedures to address the use of realistic, or best-estimate, evaluation models in accordance with 10 CFR 50.46 and Regulatory Guide 1.157.

At present, SRP section 6.2.1.5 Acceptance Criteria specifies 10 CFR 50.46 and paragraph I.D.2 of 10 CFR 50 Appendix K as the sources of requirements for evaluation of containment pressure used to evaluate ECCS performance capability. 10 CFR 50.46 and Appendix K were revised on September 16, 1988 to permit the use of an acceptable evaluation model in lieu of Appendix K. Regulatory Guide 1.157 was issued to describe models, correlations, data, model evaluation procedures, and methods acceptable to the NRC staff for meeting the requirements for a realistic or best-estimate calculation of ECCS performance during a loss-of-coolant accident and for estimating the uncertainty in the calculations. Position C.3.12.1 of the Regulatory Guide provides guidance for calculation of containment pressures when using the realistic evaluation

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model option. The Area of Review section currently contains a review interface to SRP section 6.3 that addresses mass and energy release data derived in accordance with 10 CFR 50.46 and Appendix K that is used in the minimum containment pressure analysis. This Area of Review (review interface) should also reflect the changes made to 10 CFR 50.46 that permit the use of an acceptable evaluation model in lieu of Appendix K.

Consideration should be given to revising the Areas of Review, Acceptance Criteria, and Review Procedures to accommodate use of realistic, or best-estimate evaluation models as allowed by 10 CFR 50.46 and as described in Regulatory Guide 1.157.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

9255/RG 1.157; 9354/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 320      **SRP Section Number:** 6.4

#### **Suggested Changes to the SRP Section:**

Revise specific criteria and Review Procedures to cite additional guidance (ASME AG-1) supporting review of control room habitability system compliance with TMI Action Plan Item III.D.3.4 and GDC 19. Issue 83, "Control Room Habitability," identifies significant discrepancies with respect to licensing bases found during a survey of existing plant control rooms. The EPRI Evolutionary Plant FSER describes staff positions related to evolutionary reactor resolution of GSI 83. The staff indicated that in addition to verifying conformance with current SRP Section 6.4 Acceptance Criteria and guidance, that they would also verify conformance with the following HVAC-related standards: ANSI/ANS 59.2, ASME Code AG-1, and ASTM D3803. The staff indicated that these verifications will demonstrate that the control room will adequately protect the control room operators and will remain habitable in accordance with TMI Task Action Plan Item III.D.3.4 and GDC 19 which are currently identified as Acceptance Criteria for SRP Section 6.4. The EPRI FSER Issue 83 discussion was not repeated in full in the ABWR and CE System 80+ FSERs. Only portions reiterated in these SERs are considered candidates for Type I changes under the SRP-UDP.

In the ABWR FSER, in conjunction with its evaluation of the applicant's resolution of Issue 83, the staff stated that its review criteria for Issue 83 is to verify that the control room is designed to provide adequate protection to the operating personnel during and following an accident and that the design must meet the guidance given in the SRP Sections 6.4, 6.5.1, 9.4.1, and 15.6.5.5 (which include many citations of Regulatory Guide 1.52 for relevant criteria and guidance). The staff also stated that the design must be in accordance with the requirements of GDC 2, 4, and 19 of 10 CFR Part 50, Appendix A, ASME AG-1, "Code on Nuclear Air and Gas Treatment" (1991) and the ASME AG-1a-92 Addenda, ASME 509, and ASME N510.



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Consideration should be given to revising specific criteria and Review Procedures to cite ASME AG-1-1991 and the AG-1a-92 Addenda identified above for review of control room habitability system compliance with TMI Action Plan Item III.D.3.4 and GDC 19.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

15325/NUREG 0933; 22341/FINAL SER CE80 CH 20; 22631/FINAL SER EPRI CH 1;  
25946/FINAL SER ABWR CH 20

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**Integrated Impact Number:** 321      **SRP Section Number:** 6.4

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria, Review Procedures, and Evaluation Findings to reflect the design certification/combined operating license (DC/COL) licensing process. SRP Section 6.4 currently addresses site-specific toxic substance hazards in the design of habitability features, but does not identify portions of the overall review which may only be adequately performed at the COL stage.

In the ABWR DFSE and CE System 80+ DSE, the applicants asserted, and the staff concurred, that due to the site-specific nature of toxic substance hazards, COL applicants referencing certified standard designs must identify the site-specific toxic substance hazards and demonstrate that control room operators are adequately protected against the effects of toxic releases, in accordance with guidance and criteria currently identified in SRP Section 6.4. Where necessary, the COL applicant must also provide specific detectors to permit automatic control room isolation.

Consideration should be given to revising Acceptance Criteria, Review Procedures, and Evaluation Findings to reflect the DC/COL licensing process in reviews of control room habitability features related to toxic substance detection and protection, based upon the above information.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22715/DRAFT FINAL SER ABWR CH 6; 22716/DRAFT SER CE80 CH 6

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**Integrated Impact Number:** 323      **SRP Section Number:** 9.4.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to cite additional guidance (ASME AG-1) supporting review of

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control room area ventilation system compliance with GDC 19. Issue 83, "Control Room Habitability," identifies significant discrepancies with respect to licensing bases found during a survey of existing plant control rooms. The EPRI Evolutionary Plant FSER describes staff positions related to evolutionary reactor resolution of GSI 83. The staff indicated that in addition to verifying conformance with current SRP Acceptance Criteria and guidance, that they would also verify conformance with the following HVAC-related standards: ANSI/ANS 59.2, ASME Code AG-1, and ASTM D3803. The staff indicated that these verifications will demonstrate that the control room will adequately protect the control room operators and will remain habitable in accordance with GDC 19 which is currently identified as Acceptance Criteria for SRP Section 9.4.1. The EPRI FSER Issue 83 discussion was not repeated in full in the ABWR and CE System 80+ FSERs. Only portions reiterated in these SERs are considered candidates for Type I changes under the SRP-UDP.

In the ABWR FSER, in conjunction with its evaluation of the applicant's resolution of Issue 83, the staff stated that its review criteria for Issue 83 is to verify that the control room is designed to provide adequate protection to the operating personnel during and following an accident and that the design must meet the guidance given in the SRP Sections 6.4, 6.5.1, 9.4.1, and 15.6.5.5 (which include many citations of Regulatory Guide 1.52 for relevant criteria and guidance). The staff also stated that the design must be in accordance with the requirements of GDC 2, 4, and 19 of 10 CFR Part 50, Appendix A, ASME AG-1, "Code on Nuclear Air and Gas Treatment" (1991) and the ASME AG-1a-92 Addenda, ASME 509, and ASME N510.

Consideration should be given to revising Review Procedures to cite ASME AG-1-1991 and the AG-1a-92 Addenda identified above for review of control room area ventilation system compliance with GDC 19.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

15329/NUREG 0933; 22344/FINAL SER CE80 CH 20; 22632/FINAL SER EPRI CH 1;  
25947/FINAL SER ABWR CH 20

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**Integrated Impact Number:** 324      **SRP Section Number:** 9.4.1

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria, Review Procedures, and Evaluation Findings to reflect the design certification/combined operating license (DC/COL) licensing process. SRP Section 9.4.1 currently addresses site-specific toxic substance hazards in the design of CRAVS, but does not identify portions of the overall review which may only be adequately performed at the COL stage.

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In the ABWR DFSE and CE System 80+ DSE, the applicants asserted, and the staff concurred, that due to the site-specific nature of toxic substance hazards, COL applicants referencing certified standard designs must identify the site-specific toxic substance hazards and demonstrate that control room operators are adequately protected against the effects of toxic releases, in accordance with guidance and criteria currently identified in SRP Sections 6.4 and 9.4.1. Where necessary, the COL applicant must also provide specific detectors to permit automatic control room isolation.

Consideration should be given to revising Acceptance Criteria, Review Procedures, and Evaluation Findings to reflect the DC/COL licensing process in reviews of CRAVS features related to toxic substance detection and protection, based upon the above information.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22719/DRAFT FINAL SER ABWR CH 9; 22720/DRAFT SER CE80 CH 9

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**Integrated Impact Number:** 325      **SRP Section Number:** 9.4.1

#### **Suggested Changes to the SRP Section:**

Add Acceptance Criteria and Review Procedures to address station blackout (SBO) considerations.

The capability of the Control Room Area Ventilation System (CRAVS) to provide suitable environmental conditions and/or the capability of control room area SBO-required equipment to function reliably under the expected environmental conditions needs to be reviewed for a station blackout event and covering the specified duration. Regulatory Position C.3 of RG 1.155 provides review guidance for this evaluation.

Consider adding 10 CFR 50.63 and RG 1.155 as Acceptance Criteria and developing applicable Review Procedures for the Control Room Area Ventilation System with respect to the review described above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23324/RG 1.155; 23325/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 326      **SRP Section Number:** 3.4.1

#### **Suggested Changes to the SRP Section:**

SRP Section 3.4.1 cites Regulatory Guide 1.59 in Subsection II, Acceptance Criteria. Regulatory Guide 1.59 cites ANSI N170-1976 in Appendix A. The current version is ANS 2.8-1992.

Consideration should be given to performing a detailed side by side comparison between ANSI N170-1976 and ANS 2.8-1992 to allow SRP reviewers to use the more current version of the standard. IPD 7.0 form number 3.4.1-1 has been prepared to address the need to revise Regulatory Guide 1.59.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

No comparison needed. Per 11/94 conversation with Civil Engineering and Geosciences Branch staff, Part 100 is under revision and has been issued for public comment. A regulatory guide is being written. Because of related ongoing work, do not perform comparison at this time.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23327/C&S: ANSI N170

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**Integrated Impact Number:** 327      **SRP Section Number:** 5.4.1.1

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria to address the revision to GDC 4 as it relates to the application of "leak-before-break" (LBB) methods when determining the highest anticipated overspeed used in the design of the RCP flywheels.

GDC 4 has been revised to allow for application of LBB methods in plant design. In the CE80+ FSER, the staff documented its review of the RCP flywheel design. The RCP flywheel is designed to withstand the highest anticipated overspeed predicted for the largest break size remaining after the application of LBB. The staff documented in the FSER review that this approach is consistent with the application of LBB. Current SRP Acceptance Criteria, step II.4.b, states the highest anticipated overspeed used in the design of the RCP flywheel should include consideration of the maximum rotational speed of the flywheel if a break occurs in the reactor coolant piping in either the suction or discharge side of the pump.

Consideration should be given to revising the Acceptance Criteria to be consistent with the

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revision to GDC 4 and the allowed application of LBB methods when determining the flywheels highest anticipated overspeed due to a break in the reactor coolant piping.

In addition, consider revising Regulatory Guide 1.14, Reactor Coolant Pump Flywheel Integrity, to include leak before break methods in determining flywheel speed after a LOCA. This action will be tracked by IPD-7.0 Form number 5.4.1.1-2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21747/CODE OF FED. REGS 10CFR50; 24431/FINAL SER CE80 CH 5

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**Integrated Impact Number: 328      SRP Section Number: 5.4.1.1**

#### **Suggested Changes to the SRP Section:**

ASTM E-208-1969 is cited in SRP section 5.4.1.1 as a guidance document under Acceptance Criteria step II.1.a pertaining to the determination of the nil-ductility transition temperature of the flywheel material. The current version of this standard is ASTM E-208-1991, "Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels." ASTM E-208 (no date specified) is referenced in the CE80+ DSER in a summary on materials selection.

PNL is currently performing a detailed side-by-side comparison between cited and current versions of this standard. Consider revising the reference to ASTM E-208 to specify the current 1991 version pending completion and review of the side-by-side comparison.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22857/C&S: ASTM E208; 23223/DRAFT SER CE80 CH 5

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**Integrated Impact Number: 329      SRP Section Number: 5.4.1.1**

#### **Suggested Changes to the SRP Section:**

ASTM A-370-1972 is cited in SRP section 5.4.1.1 as a guidance document under Acceptance Criteria step II.1.b pertaining to the determination of the Charpy V-notch energy of the flywheel material. The current version of this standard is ASTM A-370-1992, "Standard Test Methods and Definitions for Mechanical Testing of Steel Products." ASTM A-370 (no date specified) is referenced in the CE80+ DSER in a summary on materials selection.

PNL has performed a detailed side-by-side comparison between cited and current revisions of

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this standard. Consideration should be given to updating the citation of standard ASTM A-370 pending review and approval of the associated standard comparison.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22858/C&S: ASTM A370; 23224/DRAFT SER CE80 CH 5

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**Integrated Impact Number:** 330      **SRP Section Number:** 5.4.1.1

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria and Review Procedures to address the changes in the terminology used to describe the fracture toughness properties.

Acceptance Criteria for fracture toughness step II.2 specifies that fracture toughness criteria are derived from ASME Code section III, Appendix G. Appendix G has been revised since 1981; the most current version incorporated by reference in 10 CFR 50.55a is the 1989 edition dated July 1, 1989. The Code has been changed so that the critical stress intensity factor  $K(Ic)$  currently described in SRP Section 5.4.1.1, Revision 1 plus a standard deviation is now designated as the reference stress intensity factor  $K(Ir)$ . In addition, Review Procedures step III.2 also uses the terminology critical stress intensity factor which should be revised to be consistent with the new designations.

Consideration should be given to revising the Acceptance Criteria and the Review Procedures so that the terminology used to describe fracture toughness properties are consistent with the new designations.

In addition consider revising Regulatory Guide 1.14, Reactor Coolant Pump Flywheel Integrity, to include this reference stress intensity factor  $K(Ir)$  to be consistent with the ASME Code. This action will be tracked by IPD-7.0 Form number 5.4.1.1-2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22859/C&S: ASME III

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**Integrated Impact Number:** 331      **SRP Section Number:** 4.5.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for review of the adequacy of austenitic stainless materials in BWRs to resist intergranular stress corrosion cracking (IGSCC).

Generic Issue A-42 addressed the issue of pipe cracking in the heat-affected zones of welds in primary system piping in BWRs. This issue was resolved with the initial issuance of NUREG-0313. NRC Generic Letter 88-01 and its supplement provide staff positions for minimizing the probability of cracking in BWR austenitic stainless steel and associated welds. The letter provides positions relating to acceptable base material properties, weld material ferrite content, and fabrication practices. GL 88-01 also transmitted Revision 2 to NUREG-0313, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping."

In the ABWR FSER, the staff reviewed the use of austenitic stainless steel and verified implementation of a previous staff request that the applicant use Rev. 2 of NUREG-0313 versus the originally proposed use of Rev. 1. In the EPRI Evolutionary Plant FSER, the staff recommended that licensees and applicants follow Rev. 2 of NUREG-0313 to prevent IGSCC in stainless steel.

Consideration should be given to revising Review Procedures to identify current guidance for review of austenitic stainless steel in BWRs to resist IGSCC based upon Generic Letter 88-01, and NUREG-0313, Rev. 2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11071/NUREG 0933; 23352/FINAL SER EPRI CH 5; 23353/FINAL SER ABWR CH 5;  
23363/NRC GENERIC LETTER 88-01; 23412/NUREG 0313

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**Integrated Impact Number:** 332      **SRP Section Number:** 4.5.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address staff positions on grinding that are more restrictive than RG 1.37.

The SRP currently cites RG 1.37 which includes controls of surface preparation by manual grinding. In the EPRI Evolutionary Plant FSER, the staff reviewed a list of grinding controls applicable to BWRs that are more restrictive than RG 1.37. The staff found the controls to be

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adequate and noted they would be required for PWRs as well as BWRs.

In the CE System 80+ FSER, the staff discussed the above EPRI controls but did not explicitly take the position that they were applicable to the design (a PWR). The staff accepted the applicant's proposed grinding controls for austenitic stainless steel and noted the applicant's intent to avoid fabrication processes which would severely cold work the surface of austenitic stainless steel components.

Consideration should be given to revising Review Procedures for manual grinding of austenitic stainless materials used in control rod drive mechanisms based upon the above information. In addition, consideration should be given to revision of RG 1.37 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-5.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1081/RG 1.37; 23470/FINAL SER EPRI CH 1; 25387/FINAL SER CE80 CH 5;  
25591/FINAL SER CE80 CH 6

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**Integrated Impact Number: 333      SRP Section Number: 6.1.1**

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for review of the adequacy of BWR austenitic stainless steel piping to resist cracking mechanisms such as intergranular stress corrosion cracking (IGSCC).

GSI A-42, pertaining to the high incidence of cracks discovered in heat-affected zones of welds in primary piping in BWRs, was resolved via issuance of NUREG-0313. The original issuance of NUREG-0313 superseded Branch Technical Position (BTP) MTEB 5-7 as indicated in SRP Section 5.2.3. BTP MTEB 5-7 is currently cited in SRP Section 6.1.1 as a guideline for review of austenitic stainless steel ESF materials and fabrication practices.

NRC Generic Letter 88-01 provides staff positions for minimizing the probability of cracking in BWR austenitic stainless steel piping and associated welds. The letter provides positions relating to acceptable base material properties, weld material ferrite content, and fabrication practices. Generic Letter 88-01 also transmitted NUREG-0313, Rev. 2. Staff positions regarding evolutionary BWR conformance with NUREG-0313, Rev. 2, are taken in the EPRI Evolutionary Plant FSER and the ABWR FSER.

Consideration should be given to revising Review Procedures to identify current guidance for review of the adequacy of BWR austenitic stainless steel piping to resist IGSCC based upon Generic Letter 88-01, its supplement, and NUREG-0313, Rev. 2.



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#### **Potential Impacts/Documents supporting the Suggested Changes:**

11083/NUREG 0933; 22160/NRC GENERIC LETTER 88-01 SUP 1; 22162/FINAL SER ABWR CH 5; 22184/FINAL SER EPRI CH 1; 22251/NUREG 0313; 23364/NRC GENERIC LETTER 88-01

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**Integrated Impact Number:** 334      **SRP Section Number:** 6.1.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address staff positions that are more restrictive than RG 1.31.

SRP Section 6.1.1 cites RG 1.31 as criteria for control of delta ferrite content in stainless steel weld metal. Review Procedures for review of methods for controlling the amount of delta ferrite in stainless steel weld deposits also cite RG 1.31. RG 1.31 specifies ferrite content limits between 5 and 20% for weld metal.

Generic Letter 88-01 transmitted NUREG-0313, Rev. 2 and provides positions related to welding which are applicable to BWRs based upon the recommendations of NUREG-0313, Rev. 2.

In the CE System 80+ FSER, in conjunction with reviews of ESF materials, the staff compared the ferrite content limits for austenitic stainless steel castings and weld metal against industry guidelines and NUREG-0313, Rev. 2. NUREG-0313, Rev. 2 identifies a lower ferrite content limit of 7.5% which is more restrictive than the lower limit specified in RG 1.31.

In the CE System 80+ FSER, the staff also accepted the applicant's commitment to limit the maximum ferrite content for austenitic stainless steel weld metal to 15% and concluded that the lower limits specified will provide reasonable assurance that components of these materials will maintain adequate fracture toughness for their 60 year life.

Consider revising Review Procedures for austenitic stainless steel to address staff positions outlined in NUREG-0313, Rev. 2, the CE System 80+ FSER, and industry guidelines. In addition, consideration should be given to revision of RG 1.31 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-7.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22306/FINAL SER CE80 CH 6; 23364/NRC GENERIC LETTER 88-01; 23365/RG 1.31

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**Integrated Impact Number:** 335      **SRP Section Number:** 6.1.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for austenitic stainless steel to address staff positions more restrictive than RG 1.44.

RG 1.44 is cited in SRP 6.1.1 for control of the use of sensitized stainless steel.

NRC Generic Letter 88-01 regarded implementation of new staff positions covering technical areas related to intergranular stress corrosion cracking in BWR austenitic stainless steel piping. GL 88-01 also transmitted Revision 2 to NUREG-0313, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." The resolution to New Generic Issue 119.4, "BWR Piping Materials," notes that updating of RG 1.44 to reflect the staff's findings in NUREG- 0313, Rev. 2 is required.

Consider revising Review Procedures for austenitic stainless steel to address staff positions more restrictive than RG 1.44 as outlined in NUREG 0313, Rev. 2. In addition, consideration should be given to a revision of RG 1.44 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-6. This integrated impact is also referenced in IPD 7.0 Future Form Number 6.1.1-1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

8872/RG 1.44; 15763/NUREG 0933; 23364/NRC GENERIC LETTER 88-01

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**Integrated Impact Number:** 336      **SRP Section Number:** 6.1.1

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria for review of post-LOCA hydrogen gas evolution to include 10 CFR 50.44.

SRP Section 6.1.1 currently provides Review Procedures for hydrogen generation due to ESF fluid interaction with materials present in containment. GDC 4 and GDC 41 are currently identified as Acceptance Criteria for this aspect of the review. USI A-48, pertaining to post accident hydrogen control and mitigation for current containment designs, was resolved, except for large PWR containments covered by Generic Issue 121, via amendment of 10 CFR 50.44. 10 CFR 50.44 requires that a means be included for control of hydrogen gas that may be generated following a LOCA by various mechanisms including the corrosion of metals.

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Consideration should be given to revising Acceptance Criteria to include 10 CFR 50.44 for review of the acceptability of the post LOCA hydrogen generation properties of ESF fluids and materials present in containment.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4752/NUREG 0933; 22187/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 337      SRP Section Number: 6.1.1**

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of the acceptability of Nickel-Chromium-Iron alloys (e.g., Inconel) as ESF materials.

The EPRI Evolutionary Plant FSER provides a staff position that alloys such as Inconel 600 are susceptible to primary water stress corrosion cracking and should generally not be used in applications where stress corrosion cracking is a concern. The staff also identified alternative alloys considered acceptable with respect to stress corrosion cracking in conjunction with these positions.

In the CE80+ FSER, the staff notes that the applicant plans to use Inconel 690 in lieu of 600 as an ESF pressure-retaining material. The staff views the Inconel 690 alloy as the preferred nickel base alloy in the primary and secondary coolant loops because of its improved corrosion resistance compared to Inconel 600.

Consideration should be given to developing Review Procedures for review of the acceptability of Nickel-Chromium-Iron alloys as ESF materials based upon the staff positions discussed above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22176/FINAL SER CE80 CH 6; 22182/FINAL SER EPRI CH 1

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**Integrated Impact Number: 338      SRP Section Number: 6.1.1**

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of corrosion allowances proposed for ESF materials.

In the EPRI Evolutionary Plant FSER, the staff noted that experience has shown previously

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accepted standard corrosion allowances to be inadequate and that the designer should control corrosion mechanisms in the design of primary and secondary piping systems. The staff also stated that the general corrosion allowance should comply with the allowance specified in Section III of the ASME Code and ANSI/ASME B.31.1, Power Piping.

In the CE 80+ DSER, the staff requested that the applicant include a corrosion allowance for ESF materials for a 60 year design life and provide a technical basis for the allowance. The requested information was provided, reviewed, detailed, and found acceptable in the CE 80+ FSER.

Consideration should be given to developing Review Procedures for review of corrosion allowances proposed for ESF materials based upon the above staff positions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22180/FINAL SER CE80 CH 6; 22183/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 339      **SRP Section Number:** 6.1.1

#### **Suggested Changes to the SRP Section:**

ASTM A-262, pertaining to detecting susceptibility to intergranular attack in stainless steel, is cited as a guidance document in conjunction with RG 1.44 in SRP Section 6.1.1 Review Procedure III.A. RG 1.37 is also cited in SRP Section 6.1.1 for cleaning and cleanliness control. ASTM A-262-68 is endorsed by RG 1.37 and ASTM A-262-70 (Practice A or E) is endorsed by RG 1.44, with exceptions. The current version of this standard is ASTM A262-1993. PNL has performed a detailed side-by-side comparison between the cited and the latest versions of this standard.

Consider revising the reference to ASTM A-262 to specify the 1993 version pending NRC review and acceptance of the side-by-side comparison.

Consideration should also be given to revising RGs 1.44 and 1.37 as future work items. The future work recommendations will be tracked with IPD 7.0 Form numbers 4.5.1-1 (RG 1.44), 4.5.1-4 (RG 1.37) and 5.2.3-6 (RG 1.44).

#### **Potential Impacts/Documents supporting the Suggested Changes:**

8869/RG 1.37; 8872/RG 1.44; 22249/C&S: ASTM A262

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**Integrated Impact Number:** 340      **SRP Section Number:** 6.1.1

#### **Suggested Changes to the SRP Section:**

AWS Std D1.1, pertaining to welding of ferritic steels in structural applications, is cited as acceptance criteria for moisture control on low hydrogen welding materials in SRP Section 6.1.1. Since the SRP citation is not date specific, the version of this standard in effect at the time of issuance of SRP Section 6.1.1 (July 1981) would be applied as acceptance criteria for reviews of ESF materials. The latest version of this standard is AWS Std D1.1, Thirteenth Edition published in 1994. PNL has performed a detailed side-by-side comparison between cited and latest versions of this standard.

Consider revising the reference to AWS Std D1.1 to specify the Thirteenth Edition pending NRC review and acceptance of the side-by-side comparison.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22250/C&S: AWS D1.1

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**Integrated Impact Number:** 341      **SRP Section Number:** 6.1.1

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of the compatibility of ESF materials exposed to fluids (e.g., reactor coolant) with the water chemistry specifications provided for those fluids.

The current Areas of Review for SRP Section 6.1.1 include reviews of the compatibility of ESF materials and fluids to which the materials are exposed.

NRC Generic Letter 88-01 provides a staff position that hydrogen water chemistry (HWC) control and stringent controls on conductivity inhibit the initiation and growth of IGSCC in BWRs.

The ABWR FSER provides staff positions that the designer should follow EPRI Reports NP-5283-SR-A, September 1987 for HWC installations and NP-4947-SR, October 1988, for water chemistry limits and establishment of a successful water chemistry control program.

The EPRI Evolutionary Plant FSER Chapter 1 provides a staff position that BWR plant systems water chemistry design basis in accordance with EPRI NP-4947-SR as supplemented by Table 1.2 of the FSER is acceptable. Positions are also provided that PWR plant systems water chemistry design basis in accordance with EPRI Report NP-7077, Rev. 2, November 1990,

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(primary water) and EPRI Report NP-6239, Rev. 2, (secondary water) as supplemented by Table 1.3 of the FSER are acceptable.

In the CE System 80+ FSER, the staff accepted the applicant's primary and secondary water chemistry commitments based upon a determination that the applicant meets or exceeds the guidelines of the EPRI Utility Requirements Document.

Consideration should be given to developing Review Procedures for review of the compatibility of ESF materials exposed to fluids with the water chemistry specifications provided for those fluids.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22161/FINAL SER ABWR CH 6; 22185/FINAL SER EPRI CH 1; 22186/FINAL SER EPRI CH 1; 22251/NUREG 0313; 23364/NRC GENERIC LETTER 88-01; 24507/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 342      **SRP Section Number:** 6.1.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for review of the composition and compatibility of ESF fluids.

The current Areas of Review and Review Procedures for SRP Section 6.1.1 include reviews of the composition and compatibility of ESF materials and ESF fluids.

The ABWR FSER provides a staff position that the designer should follow EPRI Report NP-3589-SR-LD, April 1985 for the water used in emergency core cooling systems and spray systems. This position appears not applicable where hydrogen water chemistry in accordance with the latest applicable EPRI topical reports (which have been accepted by the staff) is used, based on the staff's acceptance of the applicant's commitments to use hydrogen water chemistry in accordance with later EPRI reports without following EPRI Report NP-3589.

The EPRI Evolutionary Plant FSER Chapter 1 provides staff positions that PWR plant systems water chemistry design basis in accordance with EPRI Report NP-7077, Rev. 2, November 1990, (primary water) and EPRI Report NP-6239, Rev. 2, (secondary water) as supplemented by Table 1.3 of the FSER are acceptable.

In the CE System 80+ FSER, the staff accepted the applicant's primary and secondary water chemistry commitments based upon a determination that the applicant meets or exceeds the guidelines of the EPRI Utility Requirements Document.

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Consideration should be given to revising Review Procedures for review of the composition and compatibility of ESF fluids based upon information provided in the above FSERs.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22161/FINAL SER ABWR CH 6; 22186/FINAL SER EPRI CH 1; 24507/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 346      **SRP Section Number:** 6.1.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address staff positions on grinding that are more restrictive than RG 1.37.

The SRP currently cites RG 1.37 which includes controls of surface preparation by manual grinding. In the EPRI Evolutionary Plant FSER, the staff reviewed a list of grinding controls applicable to BWRs that are more restrictive than RG 1.37. The staff found the controls to be adequate and noted they would be required for PWRs as well as BWRs.

In the CE System 80+ FSER, the staff discussed the above EPRI controls but did not explicitly take the position that they were applicable to the design (a PWR). The staff accepted the applicant's proposed grinding controls for austenitic stainless steel and noted the applicant's intent to avoid fabrication processes which would severely cold work the surface of austenitic stainless steel components (also see INTEGRATED IMPACT #807 for SRP Section 5.2.3).

Consideration should be given to revising Review Procedures for manual grinding of austenitic stainless reactor coolant pressure boundary materials based upon the above information. In addition, consideration should be given to revision of RG 1.37 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-5. This integrated impact is also referenced in IPD 7.0 Future Regulatory Action Needs form 6.1.1-1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

8869/RG 1.37; 22175/FINAL SER CE80 CH 6; 23474/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 348      **SRP Section Number:** 9.2.1

#### **Suggested Changes to the SRP Section:**

Evaluate documentation associated with the resolution of Generic Issue 153 for potential

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changes in the SRP. These documents include NUREG-1461, "Regulatory Analysis for the Resolution of GI-153, Loss of Essential Service Water in LWRs," and all its associated references such as Information Notice 92-49, "Recent Loss or Severe Degradation of Service Water Systems."

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23517/NUREG 0933

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**Integrated Impact Number:** 349      **SRP Section Number:** 9.4.5

#### **Suggested Changes to the SRP Section:**

Add acceptance criteria and review procedures to address station blackout (SBO) considerations.

10 CFR 50.63(a)(2) requires that support systems must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability of the engineered safety feature ventilation system (ESFVS) to maintain suitable environmental conditions and/or the capability of SBO-required equipment located in areas served by the ESFVS to function reliably under the expected environmental conditions needs to be reviewed for a SBO event. Regulatory Position C.3 of RG 1.155 provides regulatory guidance for this review.

Consider adding 10 CFR 50.63 and RG 1.155 as acceptance criteria and associated review procedures for the ESF ventilation system with respect to the review described above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

10520/RG 1.155; 23503/CODE OF FED. REGS 10CFR50; 23504/RG 1.155

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**Integrated Impact Number:** 350      **SRP Section Number:** 3.9.4

#### **Suggested Changes to the SRP Section:**

Revise references to the listed standards to reflect the latest version of the standards. The following standards are referenced in Acceptance Criteria step 2.b.(2) of the specific criteria for SRP Section 3.9.4. These standards are referenced in the SRP as acceptable standards for Quality Group D piping systems.

- (1) ANSI B16.5, "Steel Pipe Flanges and Flanged Fittings,"
- (2) ANSI B16.9, "Wrought Steel Butt Welding Fittings,"



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- (3) ANSI B16.11, "Steel Fittings Steel Welding and Threaded,"
- (4) ANSI B16.25, "Butt Welding Ends - Pipe, Valves, Flanges, and Fittings,"
- (5) ANSI B31.1, "Power Piping,"
- (6) ANSI B16.34, "Steel Valves with Flanged and Butt Welding Ends,"

Since the citation of these standards are limited to Quality Group D piping systems, no code comparison was performed on the basis of the limited potential safety significance of any differences between the cited and latest versions of the standards. The latest versions of the standards are listed below. Under the SRP-UDP, however, standards citations existing in the SRP are not updated (nor are new citations added to address update of standards citations in other regulatory documents cited in the SRP) without evidence of NRC review and acceptance of updated standards and NRC authorization to update such citations. Pending NRC review and authorization, consideration should be given to revising SRP Section 3.9.4 to reference the current version of the standards.

- (1) ANSI/ASME B16.5-1988, "Pipe Flanges and Flanged Fittings,"
- (2) ANSI/ASME B16.9-1993, "Factory-Made Wrought Steel Buttwelding Fittings,"
- (3) ANSI/ASME B16.11-1991, "Forged Fittings, Socket - Welding and Threaded,"
- (4) ANSI/ASME B16.25-1992, "Buttwelding Ends,"
- (5) ANSI/ASME B31.1-1992, "Power Piping,"
- (6) ANSI/ASME B16.34-1988, "Valves - Flanged and Buttwelding End"

#### **INSPECTION PROGRAM BRANCH COMMENT:**

ANSI B31.1 - No comparison needed. This standard applies to non-safety related applications for new plants.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23568/C&S: ANSI B16.5; 23569/C&S: ANSI B16.9; 23570/C&S: ANSI B16.11;  
23571/C&S: ANSI B16.25; 23572/C&S: ANSI B31.1; 23573/C&S: ANSI B16.34

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**Integrated Impact Number: 354      SRP Section Number: 8.1**

#### **Suggested Changes to the SRP Section:**

Add references to RG 1.155 to the appropriate sections of Table 8-1 of SRP Section 8.1.

10 CFR 50.63, " Loss of All Alternating Current Power," was issued in June 1988 and addresses the subject of station blackout. The NRC staff subsequently published RG 1.155 to provide guidance for complying with 10 CFR 50.63. The RG establishes target reliability levels for

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electric power systems, and guidance for procedures for restoring emergency power. Topics such as standby emergency power source target reliabilities and analyses to determine an appropriate specified duration of time that a plant should be able to cope with a station blackout are covered in the RG. SRP Chapter 8 does not include references to RG 1.155.

Consider adding references to RG 1.155 to Sections 8.2, 8.3.1, and 8.3.2 of Table 8-1 of SRP Section 8.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18809/RG 1.155

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**Integrated Impact Number: 357      SRP Section Number: 8.1**

#### **Suggested Changes to the SRP Section:**

Delete RG 1.108 from Table 8-1 of SRP Section 8.

RG 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems At Nuclear Power Plants," was withdrawn August 5, 1993. The guidance that was contained in RG 1.108 was updated and incorporated into Revision 3 of RG 1.9, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants."

Consider deleting references to RG 1.108 from Table 8-1 of SRP Section 8.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18829/RG 1.108

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**Integrated Impact Number: 359      SRP Section Number: 8.1**

#### **Suggested Changes to the SRP Section:**

Add references to RG 1.153 to the appropriate sections of Table 8-1 of SRP Section 8.

The staff published RG 1.153, "Criteria for Power, Instrumentation, and Control," in December 1985. The RG endorses, with some modification and supplements, IEEE Std 603-1980 as a method acceptable to the NRC staff for complying with Commission's regulations with regard to the design, reliability, qualification, and testability of the power (including electric, pneumatic, and hydraulic power), instrumentation, and control portions of safety systems. The RG

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identifies Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 2, 4, 10, 12, 13, 15, 17, 18, 20, 21, 22, 23, 24, 25, 29, 34, 37, and 54 as being applicable to the power, instrumentation, and control portions of safety systems. The RG is not referenced in SRP Chapter 8.

Consider adding references to RG 1.153 to Sections 8.2, 8.3.1, and 8.3.2 of Table 8-1 of SRP Section 8.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18832/RG 1.153

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<b>Integrated Impact Number:</b> 360	<b>SRP Section Number:</b> 8.1
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#### **Suggested Changes to the SRP Section:**

Revise the title of RG 1.9 in Table 8-1 of SRP Section 8 to be consistent with the latest revision of the Regulatory Guide.

The staff issued Revision 3 of RG 1.9, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants," in July 1993. The RG incorporates guidance that was previously included in RG 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems At Nuclear Power Plants," which was subsequently withdrawn August 5, 1993. In addition, the RG endorses IEEE 387-1984, which is the current version of IEEE standard.

Consider revising the title of RG 1.9 in Table 8-1 of SRP Section 8 to agree with Revision 3 of the RG.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18841/RG 1.9

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<b>Integrated Impact Number:</b> 362	<b>SRP Section Number:</b> 8.1
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#### **Suggested Changes to the SRP Section:**

Add references to 10 CFR 50.63 to the appropriate sections of Table 8-1 of SRP Section 8.

NRC published 10 CFR 50.63, "Loss of all alternating current power," in June 1988. The station blackout rule, 10 CFR 50.63, requires each light-water- cooled nuclear power plant licensed to

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operate to be able to withstand for a specified duration, and recover, from a station blackout as defined in 10 CFR 50.2. The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Paragraph (c),(2) states that the alternate ac power source(s) will constitute acceptable capability to withstand station blackout provided an analysis is performed which demonstrates that the plant has this capability from the onset of the station blackout until the alternate ac source(s) and required shutdown equipment are started and lined up to operate. SRP Chapter 8 does not include 10 CFR 50.63 as acceptance criteria.

Consider adding references to 10 CFR 50.63 to Sections 8.2, 8.3.1, and 8.3.2 of Table 8-1 of SRP Section 8.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18798/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 363      **SRP Section Number:** 8.1

#### **Suggested Changes to the SRP Section:**

Delete reference to RG 1.81 in Table 8-1 with respect to SRP Section 8.2.

RG 1.81 provides guidance related to the requirements found in 10 CFR 50, Appendix A. Criterion 5, "General Design Criteria for Nuclear Power Plants - Sharing of Structures, Systems, and Components." The RG is referenced in Table 8-1 of SRP Section 8.1, as being applicable to SRP Sections 8.2, 8.3.1, and 8.3.2, but is not referenced or discussed as guidance or acceptance criteria in SRP Section 8.2. RG 1.81 does not address offsite power systems.

Consider deleting the reference to RG 1.81 in Table 8-1 with respect to SRP Section 8.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18815/RG 1.81

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**Integrated Impact Number:** 364      **SRP Section Number:** 8.1

#### **Suggested Changes to the SRP Section:**

Delete reference to RG 1.47 and BTP ICSB-21 in Table 8-1 with respect to SRP Section 8.2.

RG 1.47 describes an acceptable method of complying with the requirements of IEEE Std. 279-1971 and Appendix B to 10 CFR Part 50 with regard to indicating the inoperable status of a portion of the protection system, systems actuated or controlled by the protection system, and auxiliary or supporting systems that must be operable for the protection system and the systems it actuates to perform their safety-related functions. Branch Technical Position ICSB-21 provides guidance for application of RG 1.47. The RG and BTP are referenced in Table 8-1 of SRP Section 8.1 as being applicable to SRP Section 8.2 but is not referenced or discussed as guidance or acceptance criteria on SRP Section 8.2. The RG and BTP do not address offsite power systems.

Consider deleting the reference to RG 1.47 and BTP ICSB-21 in Table 8-1 with respect to SRP Section 8.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18811/RG 1.47

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**Integrated Impact Number:** 365      **SRP Section Number:** 8.2

#### **Suggested Changes to the SRP Section:**

Consider specifying the current version of IEEE 308.

Revision 2 of RG 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," endorses IEEE 308-1974 rather than the current version which is IEEE 308-1991. RG 1.32 and IEEE 308-1974 are referenced in Acceptance Criteria 1 and 2.

A detailed comparison between the 1974 and 1991 versions would be needed to support endorsement of the current version. PNL has performed a comparison between the 1974 and 1980 versions. IPD 7.0 form number 8.1-4 has been initiated in connection with SRP Section 8.1 to revise RG 1.32 to endorse the current version.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

8607/RG 1.32

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**Integrated Impact Number:** 366      **SRP Section Number:** 8.2

#### **Suggested Changes to the SRP Section:**

Add references to 10 CFR 50.63, RG 1.155, and a SECY 90-016 position on alternate ac power source to the appropriate sections of SRP Section 8.2.

10 CFR 50.63, "Loss of All Alternating Current Power," was issued in June 1988 and addresses the subject of station blackout. The NRC staff subsequently published RG 1.155 to provide guidance for complying with 10 CFR 50.63. The RG establishes target reliability levels for electric power systems, and guidance for procedures for restoring emergency power. Topics such as analyses to determine an appropriate specified duration of time that a plant should be able to cope with a station blackout are covered in the RG. In SECY 90-016, the staff recommended imposition for evolutionary ALWRs of an alternate ac source of diverse design capable of powering at least one complete set of normal shutdown loads.

Consider adding references to 10 CFR 50.63 and RG 1.155 and the SECY 90-016 position to acceptance criteria and review procedures.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

8655/RG 1.155; 9049/CODE OF FED. REGS 10CFR50; 23662/SECY 90-016; 26018/FINAL SER CE80 CH 8; 26019/FINAL SER ABWR CH 8

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**Integrated Impact Number:** 367      **SRP Section Number:** 8.2

#### **Suggested Changes to the SRP Section:**

Incorporate relevant portions of BTP ICSB-11 into review procedures.

BTP ICSB 11, "Stability of Offsite Power Systems," provides guidance for determining whether a power grid meets stability requirements for providing preferred power.

Consider incorporating the staff position stated in BTP ICSB 11 (PSB) into SRP Section 8.2, Review Procedure III.1.(f) which addresses grid stability analyses.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

19494/BTP ICSB 11 (PSB)

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**Integrated Impact Number:** 368      **SRP Section Number:** 8.2

#### **Suggested Changes to the SRP Section:**

Add review procedures for plant grounding and lightning protection.

In the FSER for the System 80+ design, the staff stated that EPRI Evolutionary Plant Utility Requirements Document guidelines regarding plant grounding and lightning protection need to be addressed in the design. In response to ACRS and staff concerns regarding the ABWR design, the FSER reflects that the applicant committed to design criteria for grounding, surge, and lightning protection that meets the relevant EPRI plant grounding and surge protection guidelines. The staff concluded that the EPRI guidelines, in combination with adequate alternatives and justification for any proposed deviations, constitute an acceptable method for addressing grounding, surge protection, and lightning protection in a manner that satisfies the requirements of GDC 17.

Consider adding review procedures for plant grounding and lightning protection based upon acceptable EPRI guidelines.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23656/FINAL SER CE80 CH 8; 23658/FINAL SER ABWR CH 8

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**Integrated Impact Number:** 369      **SRP Section Number:** 8.2

#### **Suggested Changes to the SRP Section:**

Add review procedure on alternate source of power for nonsafety loads.

In SECY-91-078, the staff discussed its position that an evolutionary ALWR design should include an alternate power source to the nonsafety loads unless the design can demonstrate that the design margins in the evolutionary ALWR will result in transients for a loss of nonsafety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs. An additional source of power (to loads such as reactor coolant pumps, reactor recirculation pumps, main feedwater pumps, condensate pumps, and circulating water pumps) would significantly reduce the number of plant trips that involve a loss of power to the nonsafety loads and require that the plant be shut down under natural circulation. These

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events continue to be identified as more severe than the turbine-trip-only event in standard plant safety analysis reports.

Consider adding a review procedure on alternate source of power for nonsafety loads.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23659/FINAL SER EPRI CH 11; 23697/SECY 91-078; 26016/FINAL SER ABWR CH 8;  
26017/FINAL SER CE80 CH 8

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**Integrated Impact Number:** 370      **SRP Section Number:** 8.2

#### **Suggested Changes to the SRP Section:**

Add review procedure on connection of offsite power sources to the safety buses.

In SECY-91-078, the staff discussed its position that for evolutionary ALWR designs, at least one offsite circuit to each redundant safety division should be supplied directly from one of the offsite power sources with no intervening nonsafety buses, in such a manner that the offsite source can power the safety buses upon a failure of any nonsafety bus. The staff noted that IEEE 765-1983, "IEEE Standard for Preferred Power Supply for nuclear Power Generating Stations," states that the direct connection of the two offsite circuits to each redundant safety bus may further improve availability. In the ABWR Advance SER, the staff discussed this issue with respect to powering safety and nonsafety loads from common transformer windings.

Consider adding a review procedure on connection of offsite power sources to the safety buses. This item also needs to be included as a review procedure in SRP Section 8.3.1 on onsite ac power systems.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23657/ADVANCE SER ABWR CH 8; 23660/FINAL SER EPRI CH 11; 23698/SECY 91-078;  
26016/FINAL SER ABWR CH 8; 26017/FINAL SER CE80 CH 8

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**Integrated Impact Number:** 372      **SRP Section Number:** 6.5.2

#### **Suggested Changes to the SRP Section:**

Revise the Review Procedures to incorporate a discussion and reference related to methods for evaluating the fission product removal capability of containment spray systems.



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The effectiveness of containment spray systems in providing an atmospheric cleanup function is determined in part by the fission product removal coefficients. These coefficients are calculated from various spray system parameters and are utilized as inputs to accident dose calculations. The staff has developed a model that performs quantitative uncertainty analysis of spray system performance. The results of the analysis are used to develop fission product removal coefficients for containment spray systems. This model is described in NUREG/CR 5966 and differs from the simplified approach described in Section III.4.c of SRP 6.5.2.

The staff used the model from NUREG/CR 5966 to perform a comparative analysis of the spray model used by ABB-CE for the ABB-CE System 80+ containment spray design. The staff found the ABB-CE model to be acceptable based on this comparison as documented in Chapter 15, Appendix A, Section 15.A.10 of the ABB-CE System 80+ FSER.

Consider revising the Review Procedures to include NUREG/CR 5966, "A Simplified Model of Aerosol Removal by Containment Sprays", June 1993, as an acceptable method of evaluating the fission product removal capability of the containment spray system.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23696/FINAL SER CE80 CH 15; 24304/NUREG CR-5966

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**Integrated Impact Number: 373      SRP Section Number: 8.1**

#### **Suggested Changes to the SRP Section:**

Add a footnote to Table 8-1 of SRP Section 8.1 that states NRC policy regarding codes and standards.

This Draft Final SER, NUREG-1469, Section 8, provides the NRC staff evaluation of the GE ABWR Safety Analysis Report (SAR) as it relates to the Electric Power System. Section 8.3.1.1 of NUREG-1469 provides guidance regarding the use of IEEE Standards, or versions of IEEE Standards, that have not been reviewed and evaluated by the Staff. This policy was subsequently stated in SECY-93-087 and approved by the Commission on July 21, 1993. The approved policy statement is as follows:

"... staff will review both evolutionary and passive plant design applications using the newest codes and standards that have been endorsed by the NRC. Unapproved revisions to codes and standards will be reviewed on a case-by-case basis."

Additional guidance in Draft SER, NUREG-1469 states that differences between referenced

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### **Integrated Impacts**

standards and endorsed standards would have to be identified, justified, and approved for use.

Table 8-1 of SRP Section 8.1 references several IEEE standards but does not provide guidance on the use of standards not endorsed by the NRC.

Consider adding a footnote to Table 8-1 of SRP Section 8.1 that states the NRC policy concerning the use of industry codes and standards.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23580/DRAFT FINAL SER ABWR CH 8

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**Integrated Impact Number:** 374      **SRP Section Number:** 6.2.2

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to reflect additional guidance for the sizing of ECCS strainers as presented in the ABWR ASER.

SRP 6.2.2 specific Acceptance Criterion 6 indicates that the design of the sumps and the protective screen assemblies is a critical element in assuring long-term recirculation cooling capability. Therefore, evaluation of potential debris generation and associated effects including debris screen blockage under postulated post-LOCA conditions is necessary.

The last paragraph of Review Procedures indicates that the staff reviews plan and elevation drawings of the protective screen assemblies, showing the relative positions and orientations of the trash bars or grating and the stages of screening, to determine that the potential for debris clogging the screening is minimized. Further, the paragraph states that a discussion of the adequacy of the surface area of screening with respect to assuring a low velocity of approach of the water to minimize the potential for debris in the water being sucked against the screening should be presented in the applicant's SAR.

In 1985, the NRC issued Regulatory Guide 1.82, Revision 1, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," which provides guidance on the sizing criteria for ECCS strainers. Recent events at operating reactors involving the clogging of ECCS strainers has led the staff to conclude that the guidance within RG 1.82, Revision 1, may not be conservative enough to eliminate this concern. To address this issue for operating reactors, the NRC issued several Notices and Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers." Resolution of this issue for operating reactors is continuing. In the ABWR ASER and in the CE80+ FSER, the staff states that an acceptable resolution to this issue is to size the strainers in accordance with RG 1.82, Revision 1 but provide a factor of 3

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sizing margin to account for uncertainty in the synergetic effects of strainer clogging from insulation, corrosion products, and other debris.

Consideration should be given to revising Review Procedures to assure adequate sizing of ECCS strainers in evolutionary plants. IPD 7.0 Form 6.2.2-1 has been initiated to track the staff's consideration of the need for revision of Regulatory Guide 1.82.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

9803/RG 1.82; 23764/NRC BULLETIN 93-002; 23766/ADVANCE SER ABWR CH 6;  
23943/FINAL SER CE80 CH 20

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**Integrated Impact Number:** 375      **SRP Section Number:** 6.2.4

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to reflect additional guidance for the review of technical specifications presented in Generic Letter 83-02.

SRP Section 6.2.4 Review Procedures indicate that at the operating license stage of review, the reviewer determines that the content and intent of proposed technical specifications pertaining to operability and leak testing of containment isolation equipment is in agreement with requirements developed by the staff. Generic Letter 83-02, "NUREG-0737 Technical Specifications," addresses NUREG-0737 items which require technical specifications. Enclosure 1 to this generic letter contains information which may be relevant to this review.

Consideration should be given to revising Review Procedures to reflect additional guidance for the review of technical specifications presented in Generic Letter 83-02, especially Items 5 (containment purge valves) and 6 (radiation signal on purge valves) in Enclosure 1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

3941/NRC GENERIC LETTER 83-02

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**Integrated Impact Number:** 376      **SRP Section Number:** 6.2.4

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and References to cite TMI Action Plan Item II.E.4.4 and 10 CFR 50.34(f)(2)(xv).

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SRP Section 6.2.4, Acceptance Criteria II.n, cites Branch Technical Position CSB 6-4 as a source of guidance on containment purging during normal operation. 10 CFR 50.34(f)(2)(xv) is related to TMI Action Plan Item II.E.4.4, "Purging," which also addresses the issue of containment venting. This regulation directs applicants to provide information regarding the provision of a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. The information provided should demonstrate high assurance that the purge system will reliably isolate under accident conditions. NUREG-0933, in its discussion of TMI Action Plan Item II.E.4.4, identifies that generic correspondence was issued to licensees regarding various aspects of this issue. The CE80+ FSER and ABWR FSER contain detailed descriptions of the review performed for TMI Action Plan Item II.E.4.4 and 10 CFR 50.34(f)(2)(xv), respectively. Further analysis is required to determine what changes may be appropriate to SRP Section 6.2.4 Review Procedures and BTP CSB 6-4 to reflect these sources of information.

Consideration should be given to revising Acceptance Criteria and References to cite TMI Action Plan Item II.E.4.4 and 10 CFR 50.34(f)(2)(xv). Consideration should also be given to performing further analysis of the documents identified above to determine if changes to existing Review Procedures and BTP CSB 6-4 are appropriate.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

19844/CODE OF FED. REGS 10CFR50; 22980/ADVANCE SER ABWR CH 20;  
22981/DRAFT SER CE80 CH 20

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**Integrated Impact Number:** 377      **SRP Section Number:** 6.2.4

#### **Suggested Changes to the SRP Section:**

As required by 10 CFR 50.63(a)(2), the containment isolation system must be capable of maintaining containment integrity independent of preferred and onsite emergency AC power in the event of a station blackout for the involved containment isolation valves. Regulatory Guide 1.155, Position C.3.2.7, provides guidance on meeting 10 CFR 50.63 with respect to the containment isolation system being capable of maintaining appropriate containment integrity.

Consider adding 10 CFR 50.63 and Regulatory Guide 1.155 as Acceptance Criteria and developing applicable Review Procedures for the containment isolation system.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4087/RG 1.155; 23792/CODE OF FED. REGS 10CFR50; 23793/RG 1.155

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**Integrated Impact Number:** 381      **SRP Section Number:** 9.5.4

#### **Suggested Changes to the SRP Section:**

Replace citation of ANSI N195.

ANSI N195 is cited in SRP Section 9.5.4 paragraph II.4.f and IV.3. This standard is used as guidance related to the design of the diesel engine fuel oil storage and transfer system. Consideration should be given to citing the current version of this code (ANSI/ANS 59.51-1989). A detailed code comparison is currently being performed by PNL to evaluate the difference between the cited code and current version.

Consider replacing citation to ANSI N195 with ANSI/ANS 59.51 as related to functional, performance, and design requirements for the emergency diesel engine fuel oil storage and transfer system pending completion of the detailed standard comparison.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22698/C&S: ANSI N195

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**Integrated Impact Number:** 382      **SRP Section Number:** 5.2.2

#### **Suggested Changes to the SRP Section:**

ASME Boiler and Pressure Vessel Code (BPVC) Section III, Article NB-7611 "Spring-Loaded Safety Valves" was referenced in SRP 5.2.2 section II, Acceptance Criteria. Although no code year or addenda were cited, there has not been an article 7611 with this title/subject since 1974. Beginning with the 1977 edition of the ASME (BPVC), the article noted above was altered slightly and the number/title was changed to NB-7511.1 "Spring-Loaded Valves." The ASME BPVC is referenced in 10 CFR 50.55a. No code comparison was performed between the version cited in the SRP and the current version of the code on the basis that the NRC staff is actively involved in the development of the code and has a high degree of familiarity with the code and its evolution. The version of the code acceptable to the NRC for any particular application is established (incorporated by reference) in 10CFR50.55a. Staff positions regarding specific code cases are included in Regulatory Guides 1.84, 1.85, and 1.147 which are routinely updated. Consideration should be given to revising SRP section 5.2.2 such that references to the ASME BPVC are not specific to a particular version.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

23913/C&S: ASME III

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**Integrated Impact Number: 384      SRP Section Number: 6.2.3**

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to address secondary containment isolation valve (CIV) failures in reviews of secondary containment functional capabilities to maintain negative pressure following a LOCA.

SRP Section 6.2.3 Acceptance Criteria and Review Procedures for secondary containment accident response analyses currently consider the most severe single active failure in the emergency power system, in the primary containment heat removal system, in the core cooling systems, and in the secondary containment depressurization and filtration system.

In the ABWR DFSE, the staff requested information regarding an assumption that all lines that do not receive an isolation signal are open and the worst-case secondary CIV fails to close in analysis of the ability of the Standby Gas Treatment System (SGTS) to draw a negative pressure on the secondary containment volume following a LOCA. A COL action item was established to perform an SGTS draw-down analysis demonstrating capability to maintain design negative pressure following a LOCA which includes assumptions that all lines that do not receive an isolation signal are open and the postulated worst single failure of a secondary containment CIV to close. This issue is identified as acceptably resolved in the FSER based upon the staff's acceptance of the COL action item and related draw-down analysis commitments contained in the SSAR.

Consideration should be given to revising Acceptance Criteria and Review Procedures to address these assumptions in reviews of secondary containment functional capabilities to maintain negative pressure following a LOCA.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22862/FINAL SER ABWR CH 6

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**Integrated Impact Number: 385      SRP Section Number: 9.5.2**

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of the adequacy of communications systems and plant

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design features to support effective communications. SRP Section 9.5.2 includes reviews of capabilities of communication systems to provide effective communications during normal plant operations, transients, fires, accidents, and losses of offsite power.

In the EPRI Evolutionary Plant FSER, the staff reviewed proposed requirements addressing use of wireless communication systems and adverse operating experience with communications systems. The proposed wireless systems and digital computer-based instrumentation and control systems for ALWRs increased the staff's review emphasis on consideration of electromagnetic and radio frequency interference (EMI/RFI). The staff found the final EPRI requirements acceptable to address issues related to in-plant coverage, testing/inspection to demonstrate effectiveness and lack of interference, security-related regulatory requirements, problems in high noise areas, communications with personnel wearing protective equipment, channel allocation problems, independence between separate communications systems, and interference among communications systems and electronic/electrical equipment and instrumentation.

The staff also found that the EPRI-proposed general requirements for communications systems are applicable to security communications systems and that security communications should be explicitly considered in the designer's analysis and definition of specific communication needs.

In the CE System 80+ FSER, the staff applied the acceptable EPRI requirements to the review of the applicant's proposed communications systems and plant design features for protection of electronic equipment from EMI/RFI. The staff also considered EMI/RFI issues in the review of the ABWR.

Consideration should be given to developing Review Procedures for review of the adequacy of communications systems and plant design features to support effective communications, based upon the EPRI requirements accepted by the staff.

SRP Section 13.6 includes reviews of communication systems for security and cites 10 CFR 73.55 as Acceptance Criteria for security communications systems. Consideration should also be given to adding Areas of Review (Review Interfaces) with SRP Section 13.6 to support the overall review described above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22924/FINAL SER CE80 CH 9; 22925/FINAL SER CE80 CH 9; 22926/FINAL SER EPRI CH 10; 22928/FINAL SER EPRI CH 10; 22929/FINAL SER EPRI CH 10; 22931/FINAL SER ABWR CH 7

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**Integrated Impact Number: 386      SRP Section Number: 9.5.2**

#### **Suggested Changes to the SRP Section:**

Add Areas of Review and Review Procedures for review of the availability of communications systems during remote shutdown operations. SRP Section 9.5.2 includes reviews of capabilities of communication systems to provide effective communications during normal plant operations, transients, fires, accidents, and losses of offsite power. The availability of communications systems during remote shutdown operations is not explicitly discussed in the SRP.

In the ABWR FSER, the staff identified a concern regarding communication systems that are to be used when the plant is under control of the remote shutdown station, and required the applicant to demonstrate availability of the communications systems, assuming a main control room fire.

Consideration should be given to adding Areas of Review and Review Procedures for review of the availability of communications systems during remote shutdown operations.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22931/FINAL SER ABWR CH 7

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**Integrated Impact Number: 387      SRP Section Number: 2.4.2**

#### **Suggested Changes to the SRP Section:**

ANSI N170 is referenced in connection with Regulatory Guide 1.59 related to determination of design basis flooding at power reactor sites. The current version of this standard is ANSI/ANS-2.8-1992. Consideration should be given to conducting a detailed side-by-side comparison of cited and current versions to allow SRP reviewers to use the most recent standard.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

No comparison needed. Per 11/94 conversation with Civil Engineering and Geosciences Branch staff, Part 100 is under revision and has been issued for public comment. A regulatory guide is being written. Because of related ongoing work, do not perform comparison at this time.

This impact will not be processed further, pending revisions of Part 100 and issuance of related RGs.



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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22834/C&S: ANS 2.8; 22835/C&S: ANSI N170

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**Integrated Impact Number: 388      SRP Section Number: 2.4.2**

#### **Suggested Changes to the SRP Section:**

SRP Section 2.4.2 reviews flood-producing phenomena, including local intense precipitation.

10CFR52 contains requirements for site parameter envelopes that are to be included in applications for design certifications and manufacturing licenses for standard plant designs. Applications which reference standard plant designs approved under 10CFR52 are required to address the conformance of site specific parameters with the site parameter envelope for the approved certified standard design. In SERs documenting staff review of evolutionary plant applications for design certification, the staff addressed the requirements related to the site parameter envelope in Section 2.6.

Consideration should be given to developing a new SRP section (see IPD-7.0 Form 2.3.1-1) for review of the site parameter envelope associated with standard plant applications, as a candidate for future work. Consideration should also be given to revising existing SRP sections, including SRP 2.4.2, for review of site-specific parameters to reflect the site parameter-related requirements of 10 CFR 52, for applications referencing a standard plant design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23909/ADVANCE SER ABWR CH 2; 23910/ADVANCE SER CE CH 2; 23922/CODE OF FED. REGS 10CFR52; 23923/CODE OF FED. REGS 10CFR52

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**Integrated Impact Number: 389      SRP Section Number: 2.4.3**

#### **Suggested Changes to the SRP Section:**

SRP Section 2.4.3 reviews probable maximum flooding from streams and rivers.

10CFR52 contains requirements for site parameter envelopes that are to be included in applications for design certifications and manufacturing licenses for standard plant designs. Applications which reference standard plant designs approved under 10CFR52 are required to address the conformance of site specific parameters with the site parameter envelope for the approved certified standard design. In SERs documenting staff review of evolutionary plant applications for design certification, the staff addressed the requirements related to the site

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parameter envelope in Section 2.6.

Consideration should be given to developing a new SRP section (see IPD-7.0 Form 2.3.1-1) for review of the site parameter envelope associated with standard plant applications, as a candidate for future work. Consideration should also be given to revising existing SRP sections, including SRP 2.4.3, for review of site-specific parameters to reflect the site parameter-related requirements of 10 CFR 52, for applications referencing a standard plant design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20358/NRC POLICY STATEMENT 52 FR 34884; 23932/CODE OF FED. REGS 10CFR52;  
23933/CODE OF FED. REGS 10CFR52; 23934/ADVANCE SER ABWR CH 2;  
23935/ADVANCE SER CE CH 2

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**Integrated Impact Number:** 390      **SRP Section Number:** 2.4.3

#### **Suggested Changes to the SRP Section:**

ANSI N170 is referenced in connection with Regulatory Guide 1.59 related to determination of design basis floods at power reactor sites. The current version of this standard is ANSI/ANS-2.8-1992. Consideration should be given to conducting a detailed side-by-side comparison of cited and current versions to allow SRP reviewers to use the most recent standard.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

No comparison needed. Per 11/94 conversation with Civil Engineering and Geosciences Branch staff, Part 100 is under revision and has been issued for public comment. A regulatory guide is being written. Because of related ongoing work, do not perform comparison at this time.

This impact will not be processed further, pending revision of RG 1.59.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22836/C&S: ANS 2.8; 22837/C&S: ANSI N170

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**Integrated Impact Number:** 391      **SRP Section Number:** 2.4.5

#### **Suggested Changes to the SRP Section:**

SRP Section 2.4.5 reviews the determination of the design basis surge and seiche flooding, which are site specific parameters.

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10CFR52 contains requirements for site parameter envelopes that are to be included in applications for design certifications or manufacturing licenses for standard plant designs. Applications which reference standard plant designs approved under 10CFR52 are required to address the conformance of site specific parameters for the subject facility with the applicable site parameter envelope.

In the CE80+ DSER and the ABWR DFSER, the NRC staff addressed the portions of Part 52 which concern a site parameter envelope in Section 2.6.

Consideration should be given to developing a new SRP section (see IPD-7.0 Form 2.3.1-1) for review of the site parameter envelope associated with standard plant applications, as a candidate for future work. Consideration should also be given to revising existing SRP sections, including SRP 2.4.5, for review of site-specific parameters to reflect the site parameter-related requirements of 10 CFR 52, for applications referencing a standard plant design.

This Impact has been revised to be processed further.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22877/DRAFT SER CE80 CH 2; 22878/DRAFT FINAL SER ABWR CH 2; 23928/CODE OF FED. REGS 10CFR52; 23929/CODE OF FED. REGS 10CFR52; 23930/ADVANCE SER ABWR CH 2; 23931/ADVANCE SER CE CH 2

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**Integrated Impact Number:** 393      **SRP Section Number:** 2.4.6

#### **Suggested Changes to the SRP Section:**

SRP Section 2.4.6 reviews the determination of the design basis tsunami.

10CFR52 contains requirements for site parameter envelopes that are to be included in applications for design certifications or manufacturing licenses for standard plant designs. Applications which reference standard plant designs approved under 10CFR52 are required to address the conformance of site specific parameters for the subject facility with the applicable site parameter envelope.

In the CE80+ DSER and the ABWR DFSER, the NRC staff addressed the portions of Part 52 which concern a site parameter envelope in Section 2.6.

Consideration should be given to developing a new SRP section (see IPD-7.0 Form 2.3.1-1) for review of the site parameter envelope associated with standard plant applications, as a candidate for future work. Consideration should also be given to revising existing SRP sections, including

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SRP 2.4.6, for review of site-specific parameters to reflect the site parameter-related requirements of 10 CFR 52, for applications referencing a standard plant design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22873/CODE OF FED. REGS 10CFR52; 22874/DRAFT SER CE80 CH 2; 22875/DRAFT FINAL SER ABWR CH 2; 23936/CODE OF FED. REGS 10CFR52; 23937/CODE OF FED. REGS 10CFR52; 23938/ADVANCE SER ABWR CH 2; 23939/ADVANCE SER CE CH 2

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**Integrated Impact Number:** 395      **SRP Section Number:** 2.5.4

#### **Suggested Changes to the SRP Section:**

Section 2.5.4 reviews stability of subsurface materials, including site specific soil properties.

10CFR52 contains requirements for site parameter envelopes that are to be included in applications for design certifications and manufacturing licenses for standard plant designs. Applications which reference standard plant designs approved under 10CFR52 are required to address the conformance of site specific parameters with the site parameter envelope for the approved certified standard design. In SERs documenting staff review of evolutionary plant applications for design certification, the staff addressed the requirements related to the site parameter envelope in Section 2.6.

Consideration should be given to developing a new SRP section (see IPD-7.0 Form 2.3.1-1) for review of the site parameter envelope associated with standard plant applications, as a candidate for future work. Consideration should also be given to revising existing SRP sections, including SRP 2.5.4, for review of site specific parameters to reflect the site parameter-related requirements of 10 CFR 52.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

19709/CODE OF FED. REGS 10CFR52; 19718/NRC POLICY STATEMENT 52 FR 34884; 23911/ADVANCE SER CE CH 2; 23912/ADVANCE SER ABWR CH 2; 23940/CODE OF FED. REGS 10CFR52; 23941/CODE OF FED. REGS 10CFR52

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**Integrated Impact Number:** 397      **SRP Section Number:** 9.1.1

#### **Suggested Changes to the SRP Section:**

ANS 57.1 (no version specified) and ANS 57.3 (proposed) are cited as Specific Criteria related to prevention of criticality and to radiological aspects of the design of a new fuel storage facility.

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The current versions are ANSI/ANS-57.1- 1992 and ANSI/ANS-57.3-1983. Consideration should be given to conducting detailed side-by-side comparisons to allow SRP reviewers to use the current version of these standards.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

ANS 57.1 - No comparison needed. Per 11/94 conversation with Plant Systems Branch staff, this standard is used as background only and the new version can be cited in the SRP without the performance of a side by side comparison. It is noted that RG 1.13 on spent fuel pools does not cite ANS 57.1, and the review procedures of SRP Sections 9.1.1 through 9.1.5 are silent with respect to ANS 57.1.

ANS 57.3 - No comparison needed. The latest version is 1983. Per PNL's 10-6-93 letter, the "proposed" version cited in SRP 9.1.1 and the 1983 version are the same.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22841/C&S: ANS 57.1; 22842/C&S: ANS 57.3

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**Integrated Impact Number:** 398      **SRP Section Number:** 9.1.2

#### **Suggested Changes to the SRP Section:**

ANS 57.2/ANSI N210-1976 is cited as a guidance document related to prevention of criticality and provides design objectives for spent fuel storage facilities. The current version is ANSI/ANS-57.2-1983. PNL is conducting a detailed side-by-side comparison between the current and cited versions of the standard. Pending completion and review of the side-by-side comparison, consideration should be given to citing the current version of the standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22960/C&S: ANS 57.2

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**Integrated Impact Number:** 399      **SRP Section Number:** 9.1.2

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for review of the design of fuel storage racks provided for storage of recently discharged fuel. Section III.1 of SRP 9.1.2 states that the staff reviews high density storage on a case-by-case basis.

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In the EPRI FSER, the staff recommended that FDA/DC applicants submit a design that uses low-density storage racks in the spent fuel pool for, as a minimum, the most recently discharged fuel.

Consideration should be given to revising Review Procedures for review of the design of fuel storage racks provided for storage of recently discharged fuel to reflect the above staff position.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22959/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 400      **SRP Section Number:** 3.6.1

#### **Suggested Changes to the SRP Section:**

Modify acceptance criteria to reflect an alternate approach to compliance with GDC-4.

Acceptability of the plant design for protection against postulated pipe ruptures outside containment is based upon GDC-4. GDC-4 requires that structures, systems, and components important to safety be appropriately protected against dynamic effects, including the effects of pipe whipping and discharging fluids that may result from equipment failures. GDC-4 was modified in 1987 allowing dynamic effects associated with postulated pipe ruptures to be excluded from the plant design basis when analyses, referred to as "leak-before-break," demonstrate that the probability of fluid system piping rupture is extremely low. SRP 3.6.1 does not currently provide for application of such leak-before-break (LBB) criteria.

In SECY 93-087, the staff recommended that the Commission approve the application of the LBB approach to both evolutionary and passive ALWRs seeking design certification under 10 CFR Part 52. Such application would be limited to instances in which appropriate bounding limits are established using preliminary analysis results during the design certification phase and verified during the COL phase by performing the appropriate inspections, tests, analysis and acceptance criteria (ITAAC). In the staff requirements memorandum (SRM) for SECY 93-087, the Commission approved the staff's position on LBB for the ALWRs. The ABWR and CE80+ ASERs reflect application of this approach; however the staff elicited a commitment from GE to eliminate discussion of LBB in the ABWR SSAR.

Consider revising the Acceptance Criteria and modifying the Review Procedures to acknowledge an alternate approach to meeting the requirements of GDC-4.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

4489/CODE OF FED. REGS 10CFR50; 21682/NUREG 1061 VOL 3; 21722/CODE OF FED. REGS 10CFR50; 23752/ADVANCE SER ABWR CH 3; 23753/ADVANCE SER CE CH 3; 23754/SECY 93-087; 23755/ADVANCE SER ABWR CH 3; 23794/NRC POLICY STATEMENT 54 FR 18649

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**Integrated Impact Number:** 401      **SRP Section Number:** 3.6.1

#### **Suggested Changes to the SRP Section:**

This impact will not be processed further. This issue will be tracked using IPD-7.0 form # 3.6.2-1.

Rulemaking in 1987 (52 FR 41288) modified GDC-4 in that dynamic effects of pipe ruptures may be excluded from a plant's design basis provided it is demonstrated that the probability of pipe rupture is extremely low. In conjunction with this rulemaking, the NRC developed a new SRP Section 3.6.3 (issued for comment at 52 FR 32626), "Leak-Before-Break Evaluation Procedures." To date, SRP 3.6.3 has not been issued for use.

Developing and issuing SRP Section 3.6.3 should be considered a candidate for future work.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4489/CODE OF FED. REGS 10CFR50; 21682/NUREG 1061 VOL 3; 21722/CODE OF FED. REGS 10CFR50; 23752/ADVANCE SER ABWR CH 3; 23753/ADVANCE SER CE CH 3

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**Integrated Impact Number:** 402      **SRP Section Number:** 3.9.6

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to reflect amendments of 10 CFR 50.55a related to inservice testing of pumps and valves.

SRP Section 3.9.6 currently cites 10 CFR 50.55a(g) as Acceptance Criteria as it relates to including pumps and valves whose function is required for safety in the inservice inspection program to verify operational readiness by periodic testing.

The regulations under 10 CFR 50.55a were amended (57 FR 34666) effective September 8, 1992) to add paragraph (f) which now provides detailed ASME Section XI inservice testing requirements for pumps and valves. The amendment separated requirements for in service

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testing of pumps and valves (formerly located in paragraph (g)) from requirements for inservice inspection and Section XI requirements applicable to other components which remain covered under paragraph (g). In addition, 10 CFR 50.55a(b)(2) was amended (by adding paragraph (vii)) to reference certain requirements for containment isolation valve (CIV) testing that appear in earlier versions of ASME Section XI, but which do not appear in the currently referenced versions of the code. Paragraph (viii) was also added to 50.55a(b)(2) to clarify acceptable editions and addenda of the ASME Operations and Maintenance Manual referenced in recent editions of Section XI.

In the ABWR and CE80+ ASERs, the staff applied the amended requirements of 10 CFR 50.55a identified above to reviews of the inservice testing of safety-related pumps and valves, including reviews of inservice testing of containment isolation valves.

Consideration should be given to revising Acceptance Criteria to reflect amendments of 10 CFR 50.55a related to in service testing of pumps and valves.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22117/CODE OF FED. REGS 10CFR50; 23496/ADVANCE SER ABWR CH 3;  
23522/ADVANCE SER CE CH 3; 23550/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 403      **SRP Section Number:** 3.9.6

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of safety-related motor-operated valve (MOV) testing provisions.

SRP Section 3.9.6 currently provides reviews of inservice test (IST) program compliance with provisions of Section XI of the ASME Code. Section XI inservice testing is recognized as not sufficient to provide assurance of operability of MOVs under design-basis conditions.

As a part of the resolution of TMI Action Plan Item II.E.6.1, "In Situ Testing of Valves," Generic Letter 89-10 extended the scope of the MOV design-basis testing and switch setting program outlined for certain safety-related MOVs in Bulletin 85-03 and its Supplement to include all safety-related MOVs, as well as all position-changeable MOVs. The extended program includes periodic in- situ testing under maximum practical conditions, qualification testing of prototype MOVs, and inspection/maintenance of MOVs to provide assurance that they will function when subjected to the design-basis conditions (including design-basis degraded voltage conditions) that are to be considered during both normal operation and abnormal events within the design basis of the plant. For position-changeable MOVs, determination of maximum



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differential pressure or flow conditions should include consideration of MOV ability to recover from mispositioning. Several Supplements to Generic Letter 89-10 have been issued, providing further guidance for safety-related MOV testing programs. The Generic Letter 89-10 program also resolves Issue 87, "Failure of High Pressure Coolant Injection Steamline Without Isolation."

As indicated in Integrated Impact Number 407, the staff applied Generic Letter 89-10 guidance to review of MOV test programs for evolutionary plants.

Consideration should be given to developing Review Procedures for review of safety-related MOV testing provisions for current plants, consistent with Generic Letter 89-10 and associated supplemental guidance.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11874/NRC BULLETIN 85-03; 11877/NRC BULLETIN 85-03 Sup 1; 11917/NRC GENERIC LETTER 89-10; 11920/NRC GENERIC LETTER 89-10 Sup 3; 11922/NRC GENERIC LETTER 89-10 Sup 1; 21609/NRC GENERIC LETTER 89-10 SUP 4; 22431/NRC GENERIC LETTER 89-10 SUP 5; 23765/NRC GENERIC LETTER 89-10 SUP 6; 23781/NUREG 0933; 23783/NUREG 0933

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**Integrated Impact Number:** 404      **SRP Section Number:** 3.9.6

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of evolutionary plant testability, design features, and test program elements necessary to verify operability of safety-related check valves under design-basis conditions.

SRP Section 3.9.6 currently provides reviews of inservice test (IST) program compliance with provisions of Section XI of the ASME Code. Section XI inservice testing is recognized as not sufficient to provide assurance of operability of components under design-basis conditions.

SECY 90-016 proposed the four positions applicable to evolutionary plant safety-related pumps and valves, including that check valve testing should incorporate advanced, nonintrusive techniques to address degradation and performance. The Commission approved these positions as supplemented by the staff's response to the ACRS comments to resolve check valve testing issues, to indicate how requirements are to be applied, and to consider alternatives.

In the EPRI Evolutionary Plant FSER, the ABWR ASER, and the CE80+ ASER, the staff applied these positions to review of proposed safety-related valve testing commitments. The staff also elicited further clarifying details from the applicants with respect to DC/COL applicant

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responsibilities related to IST of valves.

Consideration should be given to developing Review Procedures for review of evolutionary plant testability design features and test program elements necessary to verify safety-related check valve operability under design-basis conditions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11870/NRC BULLETIN 83-03; 22200/FINAL SER EPRI CH 8; 23055/SECY 93-087; 23495/ADVANCE SER ABWR CH 3; 23514/ADVANCE SER ABWR CH 20; 23519/ADVANCE SER CE CH 20; 23521/ADVANCE SER CE CH 3; 23547/FINAL SER EPRI CH 1; 23549/FINAL SER EPRI CH 5

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**Integrated Impact Number:** 405      **SRP Section Number:** 3.9.6

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of pressure isolation valve (PIV) testing provisions.

SRP Section 3.9.6 currently provides reviews of inservice test (IST) program compliance with provisions of Section XI of the ASME Code.

The adequacy of isolation valves between low pressure systems and the reactor coolant system (pressure isolation valves) was considered under TMI Action Plan item II.E.6.1, "In Situ Testing of Valves," Task Action Plan item B-63, "Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary," Generic Issue 96, "RHR Suction Valve Testing," Generic Issue 105, "Interfacing Systems LOCA at LWRs," and Generic Letters 80-14 and 87-06. Per the discussions for GSI B-63 and Issue 96 in NUREG-0933, the resolution of GSI B-63 involved leakage testing and/or monitoring of PIVs in accordance with a staff position originally proposed for, but apparently not included in Revision 2 to SRP Section 3.9.6.

The staff issued Generic Letter 80-14 to address PIV concerns (specifically Event V isolating check valve configurations) for ECCS systems. Generic Letter 80-14 identifies several acceptable methods to assure component integrity including continuous low pressure side pressure monitoring for each valve, periodic leakage testing on each valve under plant conditions and frequencies specified in the letter, and periodic nondestructive examinations (e.g., ultrasonic or radiographic) of each valve. Generic Letter 87-06 defines pressure isolation valves and requests information on PIVs and testing of PIVs at licensed plants.

In SECY 90-016, the staff recommended that evolutionary ALWR systems connected to the reactor coolant system (RCS) that are not designed to withstand full RCS pressure should

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provide the capability for leak testing of PIVs. The Commission subsequently approved the staff's position. In SERs for evolutionary plants, the staff applied this position and also determined that ABWR and CE System 80 + applications included acceptable commitments to test PIVs.

Consideration should be given to developing Review Procedures for review of pressure isolation valve (PIV) testing, consistent with the above positions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

15464/NUREG 0933; 18045/NUREG 0933; 21593/NRC GENERIC LETTER 80-14; 23041/SECY 90-016; 23505/ADVANCE SER ABWR CH 3; 23512/ADVANCE SER ABWR CH 20; 23514/ADVANCE SER ABWR CH 20; 23518/ADVANCE SER CE CH 20; 23519/ADVANCE SER CE CH 20; 23530/ADVANCE SER CE CH 3; 23780/NRC GENERIC LETTER 87-06; 23781/NUREG 0933

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**Integrated Impact Number:** 406      **SRP Section Number:** 3.9.6

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures to address staff guidance for inservice testing of certain specific pumps and valves, including: check valves, power operated valves, containment isolation valves, and pumps tested on minimum flow lines.

TMI item II.E.6.1 considered issues related to in situ testing of valves, including check valves. In Bulletin 83-03, the staff discussed limitations of the effectiveness of Code requirements for check valve testing.

In response to deficiencies identified during reviews of proposed IST programs, the staff issued Generic Letter 89-04, providing detailed guidance for compliance with 10 CFR 50.55a requirements for inservice testing. The letter included guidance on testing of: check valves, power operated valves, rapid-acting valves, BWR control rod scram valves, pumps using a minimum-flow return line, and containment isolation valves.

Supplement 1 to Generic Letter 89-04 notified addressees that the staff is issuing NUREG-1482, "Guidelines for Inservice Testing Programs at Nuclear Power Plants." NUREG-1482 contains recommendations that addressees may follow in developing and implementing inservice testing programs. NUREG-1482 through the staff's endorsement in supplement 1 to Generic Letter 89-04, provides the requisite approval for updating an IST program to the requirements of OM-6 and OM-10 (and OM-1 through reference in OM-10) provided the use of OM-6 and OM-10 is documented in the IST program. The NUREG, through supplement 1 to Generic Letter 89-04,

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also gives approval to implement selected portions of OM-6 and OM-10 as discussed in NUREG-1482.

Generic Issue 70 considered the adequacy of PWR pressurizer power-operated relief and block valves. As part of the resolution of Generic Issue 70, the staff concluded in Generic Letter 90-06 that PORVs and valves in PORV control air systems (classified as non-safety-related at several licensed PWRs), and block valves should be included within the scope of the inservice testing program. Plant conditions and test frequencies for valve stroke tests were also recommended, as was inclusion of block valves in the MOV test program discussed in Generic Letter 89-10 (see Integrated Impact #403).

Consideration should be given to developing Review Procedures to address staff guidance for inservice testing of pumps and valves, as outlined in Generic Letters 83-03, 89-04, 89-10 and 90-06.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

7487/NRC GENERIC LETTER 89-04; 11870/NRC BULLETIN 83-03; 11926/NRC GENERIC LETTER 90-06; 15464/NUREG 0933; 23529/ADVANCE SER CE CH 3; 23533/ADVANCE SER CE CH 3; 23781/NUREG 0933; 23782/NUREG 0933; 25630/NRC GENERIC LETTER 89-04 SUP 1

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**Integrated Impact Number:** 407      **SRP Section Number:** 3.9.6

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of evolutionary plant testability design features and test program elements necessary to verify operability of safety-related pumps and valves under maximum flow and design-basis differential pressure conditions.

SRP Section 3.9.6 currently provides reviews of inservice test (IST) program compliance with provisions of Section XI of the ASME Code. Section XI inservice testing is recognized as not sufficient to provide assurance of operability of components under design-basis conditions.

SECY 90-016 proposed the four positions applicable to evolutionary plant safety-related pumps and valves, including: designs should incorporate provisions for full flow testing (maximum design flow) and provisions to test MOVs under design-basis differential pressure (dP), and, a program should determine the frequency necessary for disassembly and inspection to detect unacceptable degradation that cannot be otherwise detected. The Commission approved these positions as supplemented by the staff's response to the ACRS comments to emphasize requirements of Generic Letter 89-10 on motor-operated valve (MOV) testing, to resolve check

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valve testing issues, to indicate how requirements are to be applied, and to consider alternatives. (Integrated Impact #403 addresses application of Generic Letter 89-10 for current plants.)

In SECY 93-087, the staff clarified positions as follows: full flow testing should be conducted at maximum design flow with analysis to extrapolate to design pressure if not practicable to conduct pump tests at design flow and pressure; and, a qualification test (under design-basis dP) should be conducted prior to installation and in-situ valve tests should be conducted under the maximum practicable dP and flow when not practicable to achieve design-basis dP.

In the EPRI Evolutionary Plant FSER and the ABWR and CE80+ ASERs, the staff applied these positions to review of proposed safety-related pump and valve testing commitments and elicited further clarifying details with respect to DC/ COL applicant responsibilities related to IST of pumps and valves.

Consideration should be given to developing Review Procedures for review of evolutionary plant testability design features and test program elements necessary to verify pump and valve operability under maximum flow and design-basis differential pressure conditions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11874/NRC BULLETIN 85-03; 11917/NRC GENERIC LETTER 89-10; 11922/NRC GENERIC LETTER 89-10 Sup 1; 22200/FINAL SER EPRI CH 8; 22202/FINAL SER EPRI CH 1; 23055/SECY 93-087; 23495/ADVANCE SER ABWR CH 3; 23500/ADVANCE SER ABWR CH 3; 23511/ADVANCE SER ABWR CH 20; 23514/ADVANCE SER ABWR CH 20; 23519/ADVANCE SER CE CH 20; 23521/ADVANCE SER CE CH 3; 23548/FINAL SER EPRI CH 1; 23549/FINAL SER EPRI CH 5; 23765/NRC GENERIC LETTER 89-10 SUP 6

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**Integrated Impact Number:** 409      **SRP Section Number:** 3.9.6

#### **Suggested Changes to the SRP Section:**

Incorporate staff guidance supplementing the provisions of the ASME Code, Section XI, into Acceptance Criteria for review of inservice testing of pumps and valves.

SRP Section 3.9.6 currently provides reviews of inservice test (IST) program compliance with provisions of Section XI of the ASME Code but does not include/reference guidance which specifically supplements Code provisions.

Reg. Guide 1.147 identifies several Section XI ASME Code Cases related to Code inservice testing requirements which are acceptable to the staff. Reg. Guide 1.147 is maintained current

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by frequent updates as new code cases are issued and reviewed.

Consideration should be given to incorporating Reg. Guide 1.147 into Acceptance Criteria for inservice testing of pumps and valves.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23551/RG 1.147

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**Integrated Impact Number:** 410      **SRP Section Number:** 2.5.2

#### **Suggested Changes to the SRP Section:**

Currently, 10 CFR Part 100 requires that the magnitude of the OBE be at least one-half the magnitude of the SSE. Industry experience has been that such an OBE controls the design of some safety systems and produces unnecessary and inconsistent margins for the SSE loading. The requirement that the OBE be at least one-half the SSE was written into the regulation when the staff did not have much experience with the resistance of plants to ground motion from seismic events.

The staff agreed that the OBE should not control the design of safety-related systems, and is involved in a rulemaking process to amend 10 CFR Part 100, Appendix A. Changes to Appendix A would eliminate the OBE from design consideration when the OBE is established at less than or equal to one-third of the SSE. In this case the OBE would function as an inspection level earthquake below which the effect on the health and safety of the public would be insignificant and above which the licensee would be required to shut down the plant and inspect for damages.

EPRI, with the concurrence of all other evolutionary plant vendors, requested that the regulation be changed to reduce the magnitude of the OBE relative to the SSE. In SECY-90-016 (Item IV.A.) the staff recommended that the Commission approve the review approach to consider requests to decouple the OBE from the SSE on a design-specific basis for evolutionary designs. This was approved by the Commission in its SRM of June 26, 1990.

In the CE80+ FSER the staff accepted the elimination of OBE from design considerations. The staff also concluded that when the OBE was eliminated from design analysis it would still be necessary to define the OBE as an inspection level earthquake with a maximum vibratory ground acceleration equal to one-third of the SSE or less.

Consider revising SRP Section 2.5.2 to incorporate information regarding elimination of the OBE from the design of systems, structures, and components.

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(NOTE: This INTEGRATED IMPACT has been changed based on review of the Final CE80+ SER.)

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22060/DRAFT SER CE80 CH 2; 23045/SECY 93-087; 23920/FINAL SER EPRI CH 1

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**Integrated Impact Number: 411      SRP Section Number: 2.5.2**

#### **Suggested Changes to the SRP Section:**

Delete reference to RG 1.132 from Acceptance Criteria.

RG 1.132 describes site investigations required to evaluate the site for geotechnical parameters needed for engineering analysis and design. It provides guidance and recommendations for developing site-specific investigation programs for conducting subsurface investigations including the spacing and depth of borings, and sampling. Such guidance pertains to technical rationale for acceptance criteria covered in SRP Sections 2.5.1 and 2.5.4. The investigations provide information relevant to structural geology, stratigraphy, lithology, and geologic history of the site, which is covered in 2.5.1. They also provide information relevant to local foundation and groundwater conditions as well as geotechnical parameters needed for engineering analysis and design, which is covered in 2.5.4.

The information obtained from the site investigations is needed for site response analysis which is relevant to vibratory ground motion covered in SRP Section 2.5.2. However, although information obtained from the investigations relates to 2.5.2, it should not be considered as technical rationale for acceptance criteria for 2.5.2. It is not consistent with the scope of the review of 2.5.2. The seismologist does not review the drilling and sampling program details involved in the site investigations. The mechanics of how the dynamic parameters of the subsurface strata are obtained should not be the responsibility of the staff seismologist reviewing 2.5.2. It should be the responsibility of the staff geologist reviewing 2.5.1 and the geotechnical engineer reviewing 2.5.4.

Consider deleting RG 1.132 as acceptance criteria from SRP Section 2.5.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

19708/RG 1.132

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**Integrated Impact Number:** 413      **SRP Section Number:** 2.5.2

#### **Suggested Changes to the SRP Section:**

Item 5 on page 2.5.2-8 of SRP Section 2.5.2 addresses probabilistic estimates of seismic hazards. In the EPRI Evolutionary Plant FSER, the staff discussed its comparison of ABWR hazard estimates derived by LLNL using the historical earthquake method and by EPRI with the results obtained using the ABWR bounding seismic hazard curve. The use of the LLNL hazard curves resulted in the prediction of much higher core damage frequencies than did the use of the EPRI hazard curve. The staff also concluded that the LLNL hazard results suggest the EPRI bounding curve for rock sites is not conservative. Additional details are presented in SECY 93-087, Item II.C.

Consider adding staff conclusions on the use of LLNL and EPRI seismic hazard curves to Item 5 on page 2.5.2-8 of SRP Section 2.5.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23062/SECY 93-087; 23919/FINAL SER EPRI CH 1; 25736/FINAL SER CE80 CH 2;  
25737/NRC NOTICE 94-32; 25738/FINAL SER ABWR CH 2

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**Integrated Impact Number:** 415      **SRP Section Number:** 15.2.1 - 15.2.5

#### **Suggested Changes to the SRP Section:**

Modify Acceptance Criteria and Review Procedures to include General Design Criterion (GDC) 17, "Electric Power Systems" and staff guidance for the assumption of the loss of offsite power (LOOP), in addition to the limiting single-failure event, for the analysis of all transients and accidents.

SRP Section 15.2.1-15.2.5 reviews the sequence of events, analytical model and input thereto, and the predicted consequences of a number of transients which are expected to occur with moderate frequency and result in unplanned decreases in heat removal by the secondary system.

In the CE 80+ DSER, the staff stated that, in accordance with the requirements of GDC 17, LOOP should not be considered as a single-failure event, but should be assumed in the analysis for each event without changing the event category. In addition, the staff stated that the applicant is required to discuss each of the transient and accident analyses to justify that the analyses meet the GDC 17 requirements.

Consider modifying Acceptance Criteria to include GDC 17, and revising Review Procedures to



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incorporate staff guidance for the assumption of the LOOP, in addition to the limiting single-failure event, for the analysis of all transients and accidents. Alternatively, address this issue generically as proposed for SRP Section 15.0.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23026/CODE OF FED. REGS 10CFR50; 23027/ADVANCE SER CE CH 15

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**Integrated Impact Number:** 416      **SRP Section Number:** 15.2.1 - 15.2.5

#### **Suggested Changes to the SRP Section:**

ANSI N18.2-74 and ANS Trial Use Standard N212-74 are cited in a parenthetical phrase in the Acceptance Criteria to define the term "moderate frequency" events. Furthermore, the standards are listed in Section VI as References 8 and 9. The subject standards have been superseded by ANS 51.1 and ANS 52.1, respectively; the current versions, dated 1983, were reaffirmed in 1988. Consideration should be given to a comparison of the current and cited versions to allow SRP reviewers to use the more current versions of the standards.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

ANS N212, ANSI N18.2

No comparison needed. ANS N212 and ANSI N18.2 are used for definition only.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23035/C&S: ANSI N18.2-74; 23036/C&S: ANS N212-74

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**Integrated Impact Number:** 419      **SRP Section Number:** 15.6.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address staff positions related to evaluations of reactor coolant pump (RCP) operation during loss-of-coolant accidents (LOCAs).

Review Procedure III.6.c requires the reviewer to evaluate the assumption made regarding RCP trip to assure that it is consistent and conservatively modeled with respect to the final pump trip criteria which results from resolution of TMI Action Plan Item II.K.3.5.

Generic Letters 83-10A through 83-10F document the review of vendor and NRC analyses to determine whether RCP trips are necessary during a LOCA. The NRC concluded that there is a

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wide range of transients and LOCAs where it is beneficial for the operators to maintain forced circulation cooling and mixing through operation of the RCPs. However, for certain small-break LOCAs, continued operation of the RCPs or delayed RCP trips could lead to core damage. Generic Letters 83-10A through 83-10F encouraged PWR owners group studies of this concern.

Generic Letters 85-12, 86-05, and 86-06 provide guidance concerning implementation of the RCP trip criteria and contain a safety evaluation for each of the three PWR owners groups. The NRC concluded that the need for RCP trip following a transient or accident should be determined by each licensee on a case-by-case basis to ensure that whatever decision is made regarding pump operation it will result in safe, reliable operation of reactors and will not adversely affect the ability of the plant to comply with rules and regulations.

Consider revising Review Procedures to address staff positions related to RCP operation following a transient or accident as described in NRC Generic Letters 85-12, 86-05, and 86-06.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23704/NRC GENERIC LETTER 83-10A; 23705/NRC GENERIC LETTER 83-10B;  
23706/NRC GENERIC LETTER 83-10C; 23707/NRC GENERIC LETTER 83-10D;  
23708/NRC GENERIC LETTER 83-10E; 23709/NRC GENERIC LETTER 83-10F;  
23710/NRC GENERIC LETTER 85-12; 23711/NRC GENERIC LETTER 86-05; 23712/NRC  
GENERIC LETTER 86-06

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**Integrated Impact Number:** 420      **SRP Section Number:** 9.2.5

#### **Suggested Changes to the SRP Section:**

Add Review Procedures for confirmation that an ongoing program of surveillance and control is implemented and maintained to reduce fouling problems.

Reports of serious fouling events caused by mud, silt, corrosion products, or aquatic bivalve organisms in open-cycle service water systems led the NRC to establish Generic Issue 51, "Improving the Reliability of Open-Cycle Service Water Systems." To resolve this issue, the NRC initiated a research program to compare alternative surveillance and control programs to minimize the effects of fouling on plant safety. Initially the program was restricted to a study of biofouling, but in 1987 the program was expanded to also address fouling by mud, silt, and corrosion products. The research program was completed in 1989 and the NRC issued Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," directing licensees and applicants to implement a surveillance and control program. Generic Letter 89-13 Supplement 1 provided responses to utility questions concerning Generic Letter 89-13.

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The EPRI Evolutionary Plant FSER, the CE80+ FSER, and the ABWR FSER describe provisions to address GI 51 and indicate the staff's expectation of compliance with Generic Letter 89-13. The EPRI Evolutionary FSER describes the staff's application of Generic Letter 89-13 guidance to the service water system and the ultimate heat sink.

Consider adding Review Procedures to confirm that appropriate provisions and surveillance programs will be implemented to detect and control fouling of the ultimate heat sink, including the service water intake structure.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22935/FINAL SER EPRI CH 1; 22938/FINAL SER ABWR CH 20; 22942/FINAL SER CE80 CH 20; 22943/NUREG 0933; 22945/NRC GENERIC LETTER 89-13 Sup 1; 22946/NRC GENERIC LETTER 89-13

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**Integrated Impact Number:** 421      **SRP Section Number:** 9.2.5

#### **Suggested Changes to the SRP Section:**

Revise Branch Technical Position (BTP) ASB 9-2 guidance for decay heat analyses to allow use of ANS 5.1, "Decay Heat Power in Light Water Reactors" (October 1979).

SRP Section 9.2.5 states that the design of decay heat removal systems should be evaluated against BTP ASB 9-2, which requires a 20-percent uncertainty factor to be included for the first 1000 seconds following shutdown and 10-percent between 1000 seconds and 10 million seconds. In the EPRI FSER, the staff stated that ANS 5.1 (October 1979) is a conservative predictor of decay heat generation and concluded that ANS 5.1 can be used in lieu of BTP ASB 9-2 for decay heat generation rates in the design of decay heat removal systems.

Consider revising BTP ASB 9-2 to allow use of ANS 5.1 (October 1979) in lieu of the values currently defined by the BTP.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22933/FINAL SER EPRI CH 5

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**Integrated Impact Number:** 423      **SRP Section Number:** 9.2.5

#### **Suggested Changes to the SRP Section:**

1. GDC 45, cited in the SRP, requires cooling systems be designed to allow periodic

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inspections. Regulatory Guide 1.127 describes a basis acceptable to the staff for developing an inservice and surveillance program for dams, slopes, canals, and other water-control structures.

Consider adding a reference to Regulatory Guide 1.127 to SRP Section 9.2.5.

2. The EPRI FSER documents a staff position that when the ultimate heat sink (UHS) is formed by a dam or a system of dikes or levees, the UHS facility be seismic Category I and, where applicable, be designed in accordance with "Federal Guidelines for Dam Safety."

Consider adding this requirement to SRP Section 9.2.5.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1647/RG 1.127; 23583/FINAL SER EPRI CH 8

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**Integrated Impact Number:** 424      **SRP Section Number:** 10.4.1

#### **Suggested Changes to the SRP Section:**

Add areas of review discussion on seismic requirements for the main condenser for those BWRs which take credit for main steam line and condenser fission product holdup and plateout, in lieu of a main steam isolation valve leakage control system (MSIVLCS).

In the EPRI FSER and ABWR FSER, the NRC staff has taken the position that proposals to eliminate the MSIVLCS by taking credit for fission product plateout and holdup in the main steam lines and the condenser would be acceptable, subject to complying with additional requirements. In the EPRI FSER, the ABWR FSER, and in SECY 93-087 (Item II.E), the staff described positions regarding design approaches to provide assurance that the condenser would maintain its integrity during and following an SSE. The approach consists of seismically analyzing the condenser anchorages to demonstrate that the condenser is capable of sustaining the SSE loading condition without failure.

Consideration should be given to revising SRP 10.4.1 to indicate that the condenser may play a role in mitigating the consequences of an accident for a BWR without a MSIVLCS, and therefore may have to be seismically analyzed as per SRP Sections 3.2 and 3.9.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23065/SECY 93-087; 23915/FINAL SER EPRI CH 1; 23916/DRAFT FINAL SER ABWR CH 10; 23917/SECY 93-087

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**Integrated Impact Number:** 427      **SRP Section Number:** 9.3.4

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to address measures to protect low pressure or holdup tanks, which can contain primary coolant, against vacuum conditions that could result in tank damage and the potential release of radioactive material.

Generic Letter 80-21 transmitted NRC Bulletin 80-05 to licensees of all operating power reactor facilities. The licensees were requested to take corrective actions to protect low pressure or holdup tanks against potential vacuum conditions. NRC Bulletin 80-05 discusses plant events that could result in vacuum conditions and collapse of the CVCS Holdup Tanks (HUTs). The Bulletin and Generic Letter required a review of the design of all such systems to assure that measures are in place to protect against vacuum conditions that could damage tanks with the potential for release of radioactive material or detrimental effects with regard to overall safety of plant operations.

Consideration should be given to adding a Review Procedure to address measures to protect low pressure or holdup tanks that can contain primary coolant against vacuum conditions that could result in tank damage and potential release of radioactive material.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4397/NRC BULLETIN 80-05; 22494/NRC GENERIC LETTER 80-21

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**Integrated Impact Number:** 428      **SRP Section Number:** 9.3.4

#### **Suggested Changes to the SRP Section:**

Generic Issue 105 addressed interfacing systems loss-of-coolant accidents (ISLOCA). The CVCS is an interfacing system subject to review for ISLOCA concerns. In Memorandum for F. Gillespie from W. Minners, "Proposed Resolution of Generic Issue 105, 'Interfacing Systems LOCA in LWRs,'" April 2, 1993, RES proposed a new SRP section as part of the resolution of the issue. Positions relative to future plant design for ISLOCA are discussed in SECY 90-16 and SECY 93-087.

Consideration should be given to developing a new SRP section on interfacing system LOCA as a future work item. This future work recommendation will be tracked with IPD-7.0 Form # 6.3-2.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

21821/FINAL SER EPRI CH 5; 23334/NUREG 0933; 23335/NRC MEMORANDUM 04/02/93, from W. Minners to F. Gillespie; 23336/NRC MEMORANDUM 06/03/93, from E. Beckjord to J. Taylor; 23338/SECY 90-016

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**Integrated Impact Number:** 429      **SRP Section Number:** 9.3.4

#### **Suggested Changes to the SRP Section:**

Add Acceptance Criteria and Review Procedures to address the capability of the CVCS systems to provide reactor coolant pump seal injection and to support core cooling following a station blackout event.

10 CFR 50.63, the Station Blackout (SBO) rule, requires that each light-water-cooled nuclear power plant be able to withstand and recover from an SBO of a specified duration. Regulatory Guide 1.155 describes a means acceptable to the NRC staff for implementing the requirements of 10 CFR 50.63. Regulatory Guide 1.155 position C.3 contains staff positions related to systems and components required for decay heat removal and the ability to maintain adequate reactor coolant system inventory. Draft regulatory guide DG-1008 proposes guidance on the scope and content of actions to ensure that a plant has the capability to provide adequate seal injection flow to the RCP seals. However, the draft regulatory is proposed guidance that has not been finalized.

A review of the SBO event and the applicant's coping analysis will be the subject of an integrated impact for SRP Section 8.4. A review of the CVCS conformance with Regulatory Guide 1.155 positions, as applicable, would be coordinated with the review proposed for SRP Section 8.4.

Consideration should be given to adding 10 CFR 50.63 and Regulatory Guide 1.155 as Acceptance Criteria and developing applicable Review Procedures to address station blackout requirements in regard to the CVCS.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23204/RG 1.155; 23515/CODE OF FED. REGS 10CFR50; 25122/DRAFT RG DG-1008

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**Integrated Impact Number:** 430      **SRP Section Number:** 9.3.4

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to address proper design of the miniflow systems required to ensure CVCS pump protection.

NRC Bulletins 88-04, 86-03, 80-18 and NRC Generic Letter 89-04 address various miniflow design concerns. NRC Bulletin 88-04 addressed a concern that safety related centrifugal pumps could interact during parallel pump operation under miniflow conditions and result in dead headed operation of the weaker pump. NRC Bulletin 86-03 addressed a design deficiency that created a single-failure vulnerability in the minimum flow recirculation line of pumps that could cause a failure of more than one pump. NRC Bulletin 80-18 addressed corrective actions for maintaining adequate miniflow to the Centrifugal Charging Pumps (CCPs), designed for ECCS purposes, during all conditions including when the CCPs are automatically started, and where the miniflow isolation valves are automatically isolated upon safety injection initiation. NRC Generic Letter 89-04 contains staff positions on flow instrumentation for the miniflow lines. The CE-80+ FSER addressed the staff's concerns regarding the adequacy of miniflow systems for safety-related pumps, including those concerns addressed in NRC Bulletin 88-04. Those aspects of the miniflow issues concerning circumstances related primarily to the ECCS function of the CVCS should be reviewed in section 6.3, and those aspects of the miniflow issues not concerning ECCS functions for CVCS should be reviewed in this section.

Consideration should be given to adding a Review Procedure to address applicable miniflow issues for the CVCS pumps.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4212/NRC BULLETIN 80-18; 23340/NRC BULLETIN 86-03; 23341/NRC BULLETIN 88-04; 23342/NRC GENERIC LETTER 89-04; 25616/FINAL SER CE80 CH 3

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**Integrated Impact Number:** 431      **SRP Section Number:** 9.3.4

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to verify the provision of a diverse seal injection system where needed to address station blackout.

The potential for reactor coolant pump (RCP) seal failures due to a loss of cooling is being considered under Generic Issue (GI) 23. The preliminary results of the staff's studies for GI 23 indicate that the pump seal leak rates could be substantially higher than those assumed for the

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resolution of GSI A-44, "Station Blackout." Resolution of GI 23, which also incorporates seal failure issues from GI 65 and GI 130, currently has a high priority. Treatment of RCP seal issues for current plants is contingent on final resolution of GI 23 and disposition of draft Regulatory Guide 1008.

In the EPRI Evolutionary Plant FSER, the staff states that unless the final agency resolution of GI 23 dictates other actions for evolutionary plant designs, they will expect the FDA/DC applicant to submit a design that provides for independent RCP seal cooling during station blackout or to provide adequate testing of the proposed seal design to demonstrate integrity following extended loss of seal injection and cooling. In the CE 80+ FSER, the staff stated that ABB-CE was required to submit adequate test data to demonstrate seal integrity during a station blackout for an extended period, or provide a diverse seal injection system, which should be independent of the CVCS and associated support systems to the extent practicable. Subsequently, ABB-CE added an additional system, the dedicated seal injection system (DSIS), to provide reactor coolant pump seal injection. The DSIS is considered to be part of the CVCS.

Consider modifying Review Procedures as necessary to provide a review of the diverse RCP seal cooling function of the CVCS, where applicable, as part of the proposed resolution of GI 23.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23347/FINAL SER EPRI CH 1; 23515/CODE OF FED. REGS 10CFR50; 23516/RG 1.155; 23840/NUREG 0933; 25604/FINAL SER CE80 CH 9; 25605/FINAL SER CE80 CH 20

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**Integrated Impact Number:** 432      **SRP Section Number:** 10.2.3

#### **Suggested Changes to the SRP Section:**

ASTM E-208 is referenced as guidance in Acceptance Criteria related to mechanical testing of the Nil-ductility temperature. The current version of this standard is ASTM E-208, 1991.

PNL is conducting a side-by-side comparison between cited and current versions of ASTM E-208. Pending completion and review of this comparison, consideration should be given to citation of ASTM E-208, 1991.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23725/C&S: ASTM E208

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**Integrated Impact Number:** 433      **SRP Section Number:** 10.2.3

#### **Suggested Changes to the SRP Section:**

ASTM A-370 is referenced as guidance in Acceptance Criteria related to mechanical testing of steel products. The current version of this standard is ASME SA-370, 1992.

PNL is conducting a side-by-side comparison between ASME SA-370, 1992 and the cited version of ASTM A-370. Pending completion and review of this comparison, consideration should be given to citation of ASME SA-370, 1992 in place of ASTM A-370.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23726/C&S: ASTM A370

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**Integrated Impact Number:** 434      **SRP Section Number:** 10.2.3

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to incorporate preservice inspection of solid or welded rotor designs to meet a level of assurance equivalent to that provided for shrunk-on rotors by Acceptance Criterion II.3.

Industry has proposed the elimination of the use of shrunk-on rotor disks to solve the fatigue and stress-corrosion-cracking problems of the rotor disk assemblies. EPRI, ABB-CE and GE have incorporated this solution by requiring a one-piece rotor using either an integral forging or welded design. Because SRP Section 10.2.3 gives specific guidelines for preservice inspection primarily for shrunk-on designs, and a one-piece disk design will be used in the advanced reactors, SRP Section 10.2.3 does not fully apply. In the EPRI Evolutionary Plant FSER and the ABWR ASER, the staff concluded that the use of a one-piece rotor may require that additional emphasis be placed on post-machining inspection.

Consideration should be given to revising Review Procedures to address preservice inspection of solid or welded rotor designs.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23727/FINAL SER EPRI CH 13; 23731/ADVANCE SER ABWR CH 10; 23732/ADVANCE SER CE CH 10

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**Integrated Impact Number:** 436      **SRP Section Number:** 4.5.2

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for austenitic stainless steel to address staff positions more restrictive than RG 1.44.

RG 1.44 is cited in SRP 4.5.2 for control of the use of sensitized stainless steel.

NRC Generic Letter 88-01 regarded implementation of new staff positions covering technical areas related to intergranular stress corrosion cracking in BWR austenitic stainless steel piping. GL 88-01 also transmitted Revision 2 to NUREG-0313, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." The resolution to New Generic Issue 119.4, "BWR Piping Materials," notes that updating of RG 1.44 to reflect the staff's findings in NUREG-0313, Rev. 2 is recommended.

Consider revising Review Procedures for austenitic stainless steel to address staff positions more restrictive than RG 1.44 as outlined in NUREG 0313, Rev. 2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1842/RG 1.44; 11072/NUREG 0933; 23362/NRC GENERIC LETTER 88-01; 23413/NUREG 0313

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**Integrated Impact Number:** 437      **SRP Section Number:** 4.5.2

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address staff positions that are more restrictive than RG 1.31.

SRP Section 4.5.2 cites RG 1.31 as criteria for control of delta ferrite content in stainless steel weld metal. RG 1.31 specifies ferrite content limits between 5 and 20% for weld metal.

Generic Letter 88-01 transmitted NUREG-0313, Rev. 2 and provides positions related to welding which are applicable to current generation BWRs based upon the recommendations of NUREG-0313, Rev. 2.

In the CE System 80+ FSER, in conjunction with reviews of control rod drive structural and ESF materials, the staff compared the ferrite content limits for austenitic stainless steel castings and weld metal against industry guidelines and NUREG-0313, Rev. 2. NUREG-0313, Rev. 2 identifies a lower ferrite content limit of 7.5% which is more restrictive than the lower limit

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specified in RG 1.31.

Consider revising Review Procedures for austenitic stainless steel to address staff positions outlined in NUREG-0313, Rev. 2 and industry guidelines.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1836/RG 1.31; 23284/FINAL SER CE80 CH 4; 23362/NRC GENERIC LETTER 88-01; 23413/NUREG 0313

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**Integrated Impact Number:** 438      **SRP Section Number:** 4.5.2

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of the acceptability of Nickel-Chromium-Iron alloys (e.g., Inconel) for reactor internal and core support materials.

The EPRI Evolutionary Plant FSER provides a staff position that alloys such as Inconel 600 are susceptible to primary water stress corrosion cracking and should generally not be used in applications where stress corrosion cracking is a concern. The staff also identified alternative alloys considered acceptable with respect to stress corrosion cracking in conjunction with these positions.

In the CE80+ FSER, the staff notes that the applicant plans to use Inconel 690 in lieu of 600 to fabricate the flow skirt. The staff views the Inconel 690 alloy as the preferred nickel base alloy in the primary coolant loops because of its improved corrosion resistance compared to Inconel 600.

Consideration should be given to developing Review Procedures for review of the acceptability of Nickel-Chromium-Iron alloys as reactor internal and core support materials based upon the staff positions discussed above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23283/FINAL SER CE80 CH 4; 23420/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 440      **SRP Section Number:** 4.5.2

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for review of the adequacy of reactor coolant chemistry measures to

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minimize intergranular stress corrosion cracking (IGSCC) in BWR austenitic stainless materials.

Generic Issue A-42 addressed the issue of pipe cracking in the heat-affected zones of welds in primary system piping in BWRs. This issue was resolved with the initial issuance of NUREG-0313. NRC Generic Letter 88-01 and its supplement provide staff positions for minimizing the probability of cracking in BWR austenitic stainless steel and associated welds. The letter includes a position addressing acceptable reactor coolant chemistry programs for BWRs. GL 88-01 also transmitted Revision 2 to NUREG-0313, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping."

In the ABWR FSER, the staff reviewed the use of austenitic stainless steel and verified implementation of a previous staff request that the applicant use Rev. 2 of NUREG-0313 versus the originally proposed use of Rev. 1. The staff also reviewed and accepted the applicant's proposed hydrogen water chemistry program. In the EPRI Evolutionary Plant FSER, the staff recommended that licensees and applicants follow Rev. 2 of NUREG-0313 to prevent IGSCC in stainless steel.

Consideration should be given to revising Review Procedures for review of the adequacy of reactor coolant chemistry measures to minimize IGSCC in BWR austenitic stainless materials.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11072/NUREG 0933; 23287/FINAL SER EPRI CH 1; 23295/FINAL SER ABWR CH 4; 23350/FINAL SER EPRI CH 5; 23351/FINAL SER ABWR CH 5; 23362/NRC GENERIC LETTER 88-01; 23413/NUREG 0313

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**Integrated Impact Number:** 443      **SRP Section Number:** 4.5.2

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to include RG 1.37 and to address staff positions on grinding that are more restrictive than RG 1.37.

Other materials-related SRP Sections (4.5.1, 5.2.3, 5.3.1, 5.4.2.1, 6.1.1 and 10.3.6) currently cite RG 1.37 which includes controls of surface preparation by manual grinding. In the EPRI Evolutionary Plant FSER, the staff reviewed a list of grinding controls applicable to BWRs that are more restrictive than RG 1.37. The staff found the controls to be adequate and noted they would be required for PWRs as well as BWRs.

In the CE System 80+ FSER, the staff discussed the above EPRI controls but did not explicitly take the position that they were applicable to the design (a PWR). The staff accepted the

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applicant's proposed grinding controls for austenitic stainless steel and noted the applicant's intent to avoid fabrication processes which would severely cold work the surface of austenitic stainless steel components.

Consideration should be given to revising Review Procedures to cite RG 1.37 and to include a review of controls for manual grinding of austenitic stainless reactor internal and core support materials based upon the above staff position.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1839/RG 1.37; 23471/FINAL SER EPRI CH 1; 25388/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 445      **SRP Section Number:** 5.3.2

#### **Suggested Changes to the SRP Section:**

Modify Acceptance Criteria and Review Procedures to reflect staff positions related to radiation embrittlement of reactor vessel materials.

Generic Letter 88-11 transmitted Rev. 2 to Reg. Guide 1.99 stating that the revised Reg. Guide would be used for review of pressure temperature limits. The staff applied Reg. Guide 1.99, Rev. 2 in their review of pressure-temperature limits for the ABWR and CE80+ evolutionary reactors.

In the CE80+ FSER, the staff concluded that the radiation induced shifts in reference temperatures for the reactor vessel materials can be predicted with reasonable accuracy and conservatism using the methodology of RG 1.99, Revision 2.

Consider modifying Acceptance Criteria and Review Procedures to reflect staff positions related to radiation embrittlement of reactor vessel materials as outlined in Reg. Guide 1.99, Rev. 2, the CE80+ FSER.

PNL COMMENT: The Part A description above has been revised to reflect the staff's positions from the CE 80+ FSER. The staff conclusions in Section 5.3.2 of the FSER supersede issues associated with the use of Regulatory Guide 1.99 identified in the DSER that were part of the original basis for this integrated impact (See PI 23466). The staff concludes in the FSER that the use of Regulatory Guide 1.99, Revision 2, is acceptable for predicting the radiation induced effects on the fracture toughness properties of the reactor vessel materials. Therefore, the issues identified in the DSER will not be incorporated in SRP Section 5.3.2.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

13482/NRC GENERIC LETTER 88-11; 13537/RG 1.99; 21616/NRC GENERIC LETTER 92-01; 23466/DRAFT SER CE80 CH 5; 23467/FINAL SER CE80 CH 5; 23468/FINAL SER ABWR CH 5

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**Integrated Impact Number:** 446      **SRP Section Number:** 5.3.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria, Review Procedures and Branch Technical Position (BTP) MTEB 5-2 to incorporate 10 CFR 50.60 and NUREG-0744.

At present, the SRP Acceptance Criteria include Appendices G and H of 10 CFR 50 with respect to reactor vessel fracture toughness and surveillance program. The regulatory requirement to comply with Appendices G and H is contained in 10 CFR 50.60 which is not currently identified in SRP 5.3.2.

Appendix G to 10 CFR Part 50 requires that the Charpy upper shelf energy throughout the life of the vessel be no less than 50 ft-lb unless it is demonstrated that lower values will provide margins of safety against failure equivalent to those provided by Appendix G of the ASME code. BTP MTEB 5-2 (Attached to SRP 5.3.2) provides guidance for making conservative estimates and assumptions for older plants that may be used to show compliance with these requirements. USI A-11 was initiated to address the staff's concern that some vessels were projected to have beltline materials with Charpy upper shelf energy less than 50 ft-lb. NUREG-0744, developed as resolution of USI A-11, provides a method for evaluating reactor vessel materials when their Charpy upper shelf energy is predicted to fall below 50 ft-lb.

Consider revising Acceptance Criteria to include 10 CFR 50.60 in connection with 10 CFR 50 Appendices G and H. Also consider revising Review Procedures and attached BTP MTEB 5-2 to reference the methodology in NUREG-0744, Rev. 1 for evaluation of vessel materials Charpy upper shelf energy less than 50 ft-lb.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1545/NUREG 0933; 6991/CODE OF FED. REGS 10CFR50; 13449/CODE OF FED. REGS 10CFR50; 13476/NRC GENERIC LETTER 82-26

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**Integrated Impact Number:** 447      **SRP Section Number:** 5.3.2

#### **Suggested Changes to the SRP Section:**

ASTM E-185-73, "Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," is cited in Branch Technical Position (BTP) MTEB 5-2, "Fracture Toughness Requirements," for surveillance program requirements. BTP MTEB 5-2 is cited as acceptance criteria for fracture toughness requirements in SRP Section 5.3.2.

10 CFR 50, Appendix H, endorses ASTM E-185 1982. The latest version of this standard is ASTM E-185-94. PNL is not currently performing a detailed side-by-side comparison between cited and latest versions of this standard. This comparison was planned, but was placed on hold pending industry developments.

Consider revising the BTP MTEB 5-2 reference to ASTM E-185 to specify the 1982 version.

PNL Comment: Part A was revised on 6-6-95 to reflect that the latest version of the standard is dated 1994, a side-by-side comparison is not currently being performed and has been placed on hold, and that the 1982 version is endorsed by 10 CFR. This INTEGRATED IMPACT will be processed to update to the 1982 version based on the 10 CFR endorsement and a separate placeholder integrated impact will address the 1994 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23574/C&S: ASTM E185-73

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**Integrated Impact Number:** 449      **SRP Section Number:** 11.2

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria, Review Procedures and Evaluation Findings to replace citations of superseded sections in 10 CFR Part 20.

Acceptance Criterion II.1 cites Section 20.106 of 10 CFR 20 as it relates to radioactivity in effluent to unrestricted areas. Section 20.106, in turn, cites Appendix B (to Sections 20.1-20.602). On May 21, 1991, the NRC issued a revision to these requirements for protection against ionizing radiation. The revised part (Sections 20.1001-20.2401) became effective for implementation on June 20, 1991 and mandatory on January 1, 1994.

The staff, in the CE 80+ and the ABWR ASERs, reviewed the liquid effluent concentrations of radionuclides to the liquid effluent concentration limits for the respective radionuclides given in

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the revised 10 CFR 20, Appendix B (to Sections 20.1001-20.2402), Table 2, Column 2. The staff independently verified that the annual average radionuclide concentrations in liquid effluent to unrestricted areas complied with 10 CFR Part 20, Section 20.1302.

Consider revising the Acceptance Criteria, Review Procedures, and Evaluation Findings to replace citations of superseded sections in 10 CFR Part 20.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4059/CODE OF FED. REGS 10CFR20; 23584/ADVANCE SER CE CH 11;  
23585/ADVANCE SER ABWR CH 11; 23737/CODE OF FED. REGS 10CFR20

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**Integrated Impact Number:** 451      **SRP Section Number:** 11.2

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures to address tank failure prevention.

SRP Section 11.2 reviews the license applicant's design for special features provided to control leakage from system components. GDC 61, cited as Acceptance Criteria in the SRP, requires that radioactive waste and other systems which may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions. Generic Letter 80-21 forwarded IEB 80-05 and was directed primarily to holdup tanks for reactor coolant but did allude to other low pressure tanks important to safety or containing radioactive materials. IE Bulletin 80-05 addresses the release of radioactive material or other adverse effects as a result of vacuum conditions causing tank buckling.

The staff, in the ABB-CE SYSTEM 80+ and the ABWR ASERs, reviewed and found the design features adequately address the concern identified in Bulletin 80-05 and are, therefore, acceptable. The staff also found these features in compliance with GDC 61.

Consider developing Review Procedures to address tank failure concerns identified in IEB 80-05 and Generic Letter 80-21.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21916/NRC BULLETIN 80-05; 22600/NRC GENERIC LETTER 80-21; 23587/ADVANCE SER ABWR CH 11; 23588/ADVANCE SER CE CH 11

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**Integrated Impact Number:** 453      **SRP Section Number:** 11.4

#### **Suggested Changes to the SRP Section:**

Add GDC 61 and 10 CFR 61 as Acceptance Criteria and develop appropriate Review Procedures to address the requirements of 10 CFR 61. In addition, revise Branch Technical Position (BTP) ETSB 11-3 to address staff positions on waste form.

Acceptance Criteria, specific criteria II.2 & II.3 require that all wet wastes be solidified or dewatered in accordance with a process control program that meets the requirements of Branch Technical Position (BTP) ETSB 11-3. Generic Issue C-17 was established because there were no current criteria for acceptability of solidification agents. Issue C-17 was resolved based upon publication of 10 CFR Part 61 in December 1982, including Section 61.56 which addresses waste characteristics. In addition a BTP on waste form has been developed under TMI Action Plan Item IV.C.1.

GDC 61 requires that radioactive waste, and other systems which may contain radioactivity, be designed to assure adequate safety under normal and postulated accident conditions. The scope of SRP 11.4 includes the review of expected and design volumes of waste to be processed and handled, the wet and dry types of waste to be processed, the activity and expected radionuclide distribution contained in the waste, equipment design capacities, and the principal parameters employed in the design of the solid waste system. The Area of Review, Acceptance Criteria (specific criteria) II.7 and II.8, and Review Procedures III.4 & III.6 through III.8 all provide for reviews consistent with GDC 61.

The ABWR and ABB-CE System 80+ ASERs and the EPRI Evolutionary Plant FSER all indicate that the staff reviewed the solid waste management system against the requirements of 10 CFR 61.

Consider adding GDC 61 and 10 CFR 61 as Acceptance Criteria and developing appropriate Review Procedures to address the requirements of 10 CFR 61. In addition, consider revising Branch Technical Position (BTP) ETSB 11-3 to address staff positions on waste form.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

5087/CODE OF FED. REGS 10CFR50; 8739/NRC GENERIC LETTER 84-12;  
18311/NUREG 0933; 23263/FINAL SER EPRI CH 12; 23264/NUREG 0933;  
23265/ADVANCE SER ABWR CH 11; 23618/ADVANCE SER CE CH 11; 23619/NRC  
MEMORANDUM 02/14/83 from T.C. Johnson to R.E. Browning

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### **Integrated Impacts**

**Integrated Impact Number:** 454      **SRP Section Number:** 11.4

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria, Review Procedures and Evaluation Findings to replace citations of superseded sections in 10 CFR Part 20.

Acceptance Criterion II.A cites section 20.106 of 10 CFR 20 as it relates to radioactivity in effluent to unrestricted areas. On May 21, 1991 the NRC issued a revision to those requirements for protection against ionizing radiation. The revised part (Sections 20.1001-20.2401) became affective for implementation on June 20, 1991 and mandatory on January 1, 1994.

The staff, in the CE 80+ ASER, reviewed the capability of the solid waste management system in conjunction with liquid and gaseous waste management systems, to maintain liquid and gaseous effluents arising from the system operation below the limits in 10 CFR Part 20, Section 20.1302.

Consider revising the Acceptance Criteria, Review Procedures, and Evaluation Findings to replace citations of superseded sections in 10 CFR Part 20.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

5055/CODE OF FED. REGS 10CFR20; 23620/ADVANCE SER CE CH 11; 23739/CODE OF FED. REGS 10CFR20

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**Integrated Impact Number:** 459      **SRP Section Number:** 11.3

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to address the content of the Offsite Dose Calculation Manual (ODCM) as it relates to control of radioactive gaseous effluents.

Review Procedure III.7 evaluates the applicable portions of the technical specifications to determine that their content and intent are in agreement with the requirements developed as a result of the staff's review.

In order to keep releases of radioactive materials to unrestricted areas as low as is reasonably achievable, 10 CFR 50.36a requires each license to include radiological effluent technical specifications (RETS) on effluents from nuclear power reactors.

The NRC staff issued Generic Letter 89-01 to relocate procedural details of the current RETS to

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the ODCM and Process Control Program (PCP). Generic Letter 89-01 states that programmatic controls can be implemented in the Administrative Controls section of the Technical Specifications (TS) to satisfy existing regulatory requirements for RETS. At the same time, the procedural details of the current TS on radioactive effluents and radiological environmental monitoring can be relocated to the Offsite Dose Calculation Manual. PNL has proposed to evaluate the ODCM as part of SRP Section 11.5, Process and Effluent Monitoring Instrumentation and Sampling System, and the PCP as part of SRP Section 11.4, Solid Waste Management System.

Consider modifying Review Procedures to address the content of the Offsite Dose Calculation Manual as it relates to control of radioactive gaseous effluents.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

2822/CODE OF FED. REGS 10CFR50; 8508/NRC GENERIC LETTER 89-01; 21246/NRC  
GENERIC LETTER 89-01

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**Integrated Impact Number:** 460      **SRP Section Number:** 11.3

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to address the requirements of Branch Technical Position (BTP) ETSB 11-5.

GDC 61, cited in Acceptance Criterion II.A.5 of SRP Section 11.3, requires radioactive waste systems to be designed to assure adequate safety under normal and accident conditions. Reg. Guide 1.70, Table 15-4, Summary Tabulation for Specific Design Basis Accidents, includes a requirement to address a Waste Gas System Failure in SAR section 15.7.1. However, SRP 15.7.1, "Waste Gas System Failure" has been deleted.

Review Procedure III.2.a. and III.3 of SRP 11.3 address the system's capacity to process waste with the single failure of a non-redundant component and provisions to stop continuous leakage paths after an explosion. Neither of these review procedures address the radiological consequences of such a single failure or leakage or provide criteria for evaluating the acceptability of these consequences.

The EPRI Evolutionary FSER, ABWR ASER, and CE System 80+ ASER include evaluations of the gaseous waste system against the requirements of BTP ETSB 11-5. Although BTP ETSB 11-5 is attached to SRP Section 11.3 and addresses postulated radioactive releases due to a waste gas system leak or failure, it is not specifically cited in SRP Section 11.3.

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Consider revising Acceptance Criteria and Review Procedures to evaluate postulated radioactive releases due to a waste gas system leak or failure as identified in Branch Technical Position ETSB 11-5.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

2837/RG 1.70; 21206/CODE OF FED. REGS 10CFR50; 22769/FINAL SER EPRI CH 12; 22770/DRAFT FINAL SER ABWR CH 11; 22771/DRAFT SER CE80 CH 11

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**Integrated Impact Number:** 461      **SRP Section Number:** 11.3

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria, Review Procedures, and Evaluation Findings to replace citations of superseded sections in 10 CFR Part 20.

Acceptance Criterion II.A.1 cites section 20.106 of 10 CFR 20 as it relates to radioactivity in effluent to unrestricted areas. Section 20.106, in turn, cites Appendix B (to Sections 20.1-20.602). On May 21, 1991, the NRC issued a revision to these requirements for protection against ionizing radiation. The revised part (Sections 20.1001-20.2401) became effective for implementation on June 20, 1991 and mandatory on January 1, 1994.

The staff, in the CE 80+ and the ABWR ASERs, reviewed the gaseous effluent concentrations of radionuclides to the gaseous effluent concentration limits listed in the revised 10 CFR 20, Appendix B (to Sections 20.1001-20.2402), Table 2, Column 1. The staff concluded that the designs complied with 10 CFR Part 20, Section 20.1302.

Consider revising the Acceptance Criteria, Review Procedures, and Evaluation Findings to replace citations of superseded sections in 10 CFR Part 20.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

2816/CODE OF FED. REGS 10CFR20; 2817/CODE OF FED. REGS 10CFR20; 21229/CODE OF FED. REGS 10CFR20; 23605/ADVANCE SER CE CH 11; 23606/ADVANCE SER ABWR CH 11; 23738/CODE OF FED. REGS 10CFR20

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**Integrated Impact Number:** 463      **SRP Section Number:** 5.4.2.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to include RG 1.31 as a reference for adequacy of ferrite content

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limits specified for austenitic stainless weld metal.

Section 5.2.3 uses RG 1.31 as criteria for control of delta ferrite content in stainless steel weld metal. RG 1.31 specifies ferrite content limits between 5 and 20% for weld metal.

Staff evaluation in Section 5.4.2 of CE System 80+ FSER refers to Section 5.2.3 of this FSER, and notes that the ferrite content limits for austenitic stainless steel given in the CESSAR DC were broader than industry guidelines. The staff indicated that the CE's modified upper ferrite limit provided in Amendment L would provide reasonable assurance that components would maintain adequate fracture toughness for the 60 year life of the plant. Section 5.2.3 of the CE System 80+ FSER cites RG 1.31 as providing appropriate controls for ferrite in stainless steel weld metal.

Consider revising Review Procedures for austenitic stainless steel to include RG 1.31 as a reference. In addition, consideration should be given to revision of RG 1.31 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-7.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

3051/RG 1.31; 23369/FINAL SER CE80 CH 5; 25167/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 464      **SRP Section Number:** 5.4.2.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to review the selection of appropriate bolting material per RG 1.65.

In the EPRI Evolutionary FSER, the staff concludes that the implementation of the requirements in Section 5.3.3 of Chapter 1 of the Evolutionary Requirements Document will ensure that metallic fasteners will perform in service as designed. In the CE System 80+ FSER, the staff reviewed materials and inspections for bolts in the RCPB to ensure they were in accordance with NRC guidance (RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs").

The NUREG 0933 discussion of Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants," indicates that the staff recommended that a new SRP Section be developed to incorporate existing guidance and industry recommendations. Generic Letter 91-17 provides additional information on the NRC's resolution of GSI 29. The bases for resolution are further documented in NUREG-1339. In NUREG-1339, the staff noted that a new SRP Section entitled, for example, "Safety-Related Bolting," would expand the limited coverage now included in the SRP and provide a systematic method for implementation of the staff position regarding the basis for resolution of GSI 29.

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NRC Bulletin 82-02 requests actions on incidents of severe degradation of threaded fasteners in closures in the reactor coolant pressure boundary.

Consider revising Review Procedures to review the selection of appropriate bolting material per RG 1.65.

Consider future work to develop a new SRP Section to provide guidance for staff reviewers regarding bolting. The future work should include definition of appropriate review interfaces to other SRP Sections. This future work will be tracked by IPD-7.0 Form Number 5.2.3-1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

3004/NRC BULLETIN 82-02; 23559/FINAL SER CE80 CH 5; 23560/NUREG 0933; 23561/NRC GENERIC LETTER 91-17; 23562/FINAL SER EPRI CH 1; 23563/NUREG 1339; 25590/RG 1.65

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**Integrated Impact Number:** 465      **SRP Section Number:** 5.4.2.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for review of steam generator materials cleaning and cleanliness controls.

The SRP currently cites RG 1.37 with regard to cleaning and cleanliness controls. RG 1.37 cites ANSI N45.2.1 in describing the staff's position with regard to cleaning and cleanliness controls. In the CE 80+ FSER, the staff indicated that ANSI/ASME NQA-2 supersedes ANSI N45.2.1, cited in RG 1.37, and indicated that they had reviewed ANSI/ASME NQA-2-1983 and found it acceptable.

Consideration should be given to revising Review Procedures for steam generator materials cleaning and cleanliness controls to cite ANSI/ASME NQA-2-1983, in addition to RG 1.37 as review guidance based upon the above staff positions. In addition, consideration should be given to revision of RG 1.37 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-3.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

Per 11/94 conversation with Quality Assurance and Maintenance Branch staff, N45.2.1 requirements are being incorporated into NQA-1 and NQA-2. RG 1.28, Revision 3 endorsed NQA-1. NRC has a program to revise the endorsement based on the results of an evaluation of the graded QA program. Also NQA is going through a review of both standards.

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Per 11-10-94 telecon with Office of Research staff, two draft regulatory guides were prepared to endorse NQA-1 and NQA-2 through their 1993 addenda. Both regulatory guides were put on hold due to NRC/NEI work on the graded QA program. In the interim, NQA-1 and NQA-2 were consolidated into a new NQA-1.

This Integrated Impact will not be processed further, pending action by the staff regarding NQA-1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

3056/RG 1.37; 23435/C&S: ASME NQA-2; 23436/C&S: ANSI N45.2.1; 23437/FINAL SER CE80 CH 4

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**Integrated Impact Number:** 467      **SRP Section Number:** 5.4.2.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures and Branch Technical Position (BTP) EMCB 5-3 to incorporate staff positions that address the resolution of Unresolved Safety Issues (USIs) A-3, A-4, and A-5.

To address secondary water chemistry and the control of impurities in secondary side water, this SRP section includes BTP EMCB 5-3, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators." In NUREG 0933 USIs A-3, A-4 and A-5 were initiated for steam generator tube integrity concerns in Westinghouse, CE and B&W plants, respectively. GL 85-02 was issued as part of this USI program and describes an integrated program for assuring steam generator tube integrity and mitigating the consequences of a steam generator tube rupture. GL 85-02 also transmitted NUREG-0844 which includes recommendations for secondary water chemistry programs to minimize steam generator tube degradation. Specifically, NUREG-0844 states that plant programs should incorporate the secondary water chemistry guidelines in EPRI report "PWR Secondary Water Chemistry Guidelines," October 1982 (EPRI NP-2704) including measures taken to minimize steam generator corrosion, including materials selection, chemistry limits, control methods, and condenser inservice inspections. Revision 2 of the "PWR Secondary Water Chemistry Guidelines" was issued in December 1988 as EPRI NP-6239.

In the EPRI Evolutionary Plant FSER, the staff found the requirements of EPRI report "PWR Secondary Water Chemistry Guidelines," Revision 2, December 1988 (EPRI NP-6239), its subsequent revisions, and as supplemented by the guidelines in Table 1.4 in the FSER, to be an adequate design basis for PWR secondary water chemistry. Table 1.4 provides guidelines for secondary makeup water chemistry that are not included in EPRI NP-6239. In the CE System 80+ FSER, the applicant committed to comply with EPRI NP-6239 and the supplemental EPRI-URD guidelines for secondary makeup water (Table 1.4). The staff noted and accepted

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resolution of several discrepancies between the proposed secondary water chemistry and the EPRI guidelines. The CE System 80+ FSER also noted that several recommendations from GL 85-02 were being followed to ensure SG tube integrity.

Consideration should be given to revising the Review Procedures and BTP EMCB 5-3 to incorporate staff positions that address the resolution of Unresolved Safety Issues (USI) A-3, A-4, and A-5, as represented by the above staff positions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

19773/NRC GENERIC LETTER 85-02; 23565/NUREG 0844; 23566/FINAL SER CE80 CH 5; 23567/FINAL SER EPRI CH 1; 25545/FINAL SER CE80 CH 20; 25546/FINAL SER CE80 CH 5; 25547/FINAL SER CE80 CH 5; 25548/EPRI NP-6239

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**Integrated Impact Number:** 468      **SRP Section Number:** 5.4.2.1

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of corrosion allowances proposed for steam generator materials.

In the EPRI Evolutionary Plant FSER, the staff noted that experience has shown previously accepted standard corrosion allowances to be inadequate and that the designer should control corrosion mechanisms in the design of primary and secondary piping systems. The staff also stated that the general corrosion allowance should comply with the allowance specified in Section III of the ASME Code and ANSI/ASME B.31.1, Power Piping. In the CE 80+ DSER, the staff requested that the applicant include a corrosion allowance for steam generator materials for a 60 year design life and provide a technical basis for the allowance. The requested information was provided, reviewed and found acceptable in the CE 80+ FSER.

Consideration should be given to developing Review Procedures for review of corrosion allowances proposed for steam generator materials based upon the above staff positions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23477/FINAL SER CE80 CH 5; 23479/FINAL SER CE80 CH 5; 23481/FINAL SER EPRI CH 1

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### **Integrated Impacts**

**Integrated Impact Number:** 470      **SRP Section Number:** 5.4.2.1

#### **Suggested Changes to the SRP Section:**

Develop review procedures for review of the acceptability of Nickel-Chromium-Iron alloys (e.g., Inconel) as steam generator materials.

The EPRI Evolutionary Plant FSER provides a staff position that alloys such as Inconel 600 are susceptible to primary water stress corrosion cracking and should generally not be used in applications where stress corrosion cracking is a concern. The staff specifically discouraged the use of Alloy 600 in steam generators. The staff also identified alternative alloys considered acceptable with respect to stress corrosion cracking in conjunction with these positions.

In the CE80+ FSER, the staff notes that the applicant plans to use Inconel 690 in lieu of Inconel 600 for RCPB components. In Section 5.2.3 of the CE80+ FSER, the staff stated it views the Inconel 690 alloy as the preferred nickel base alloy in the primary and secondary coolant loops because of its improved corrosion resistance compared to Inconel 600.

Bulletin 89-01 and its two supplements indicate that Inconel 600 tube plugs may fail due to primary water stress corrosion cracking.

Consideration should be given to developing Review Procedures for review of the acceptability of Nickel-Chromium-Iron alloys as steam generator materials based upon the staff positions discussed above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

2990/NRC BULLETIN 89-01; 19791/NRC BULLETIN 89-01 Sup 2; 19792/NRC BULLETIN 89-01 Sup 1; 23423/FINAL SER EPRI CH 1; 23424/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 471      **SRP Section Number:** 5.4.2.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to include RG 1.44 as a reference for the use of sensitized stainless steel. Provisions should also be made for staff positions more restrictive than RG 1.44.

SRP Section 5.4.2.1 does not cover the use of austenitic stainless steels. However, in the CE 80+ FSER, the staff reviewed the use of austenitic stainless steel in its SRP Section 5.4.2.1 review. The staff noted controls imposed in Sections 4.5.1, 4.5.2, and 5.2.3 for welding, fabrication, and water chemistry and carbon content of austenitic stainless steel were adequate to

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mitigate the occurrence of IGSCC. In section 5.2.3 of the CE 80+ FSER, the staff cites RG 1.44 as providing appropriate guidance for avoiding sensitized stainless steel.

RG 1.44 is cited in SRP 4.5.1, 4.5.2 and 5.2.3 for control of the use of sensitized stainless steel.

NRC Generic Letter 88-01 regarded implementation of new staff positions covering technical areas related to intergranular stress corrosion cracking in BWR austenitic stainless steel piping. GL 88-01 also transmitted Revision 2 to NUREG-0313, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." The resolution to New Generic Issue 119.4, "BWR Piping Materials," notes that updating of RG 1.44 to reflect the staff's findings in NUREG-0313, Rev. 2 is recommended.

Consider revising Review Procedures for austenitic stainless steel to include RG 1.44 as a reference and to address staff positions more restrictive than RG 1.44 as outlined in NUREG 0313, Rev. 2. In addition, consideration should be given to a revision of RG 1.44 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-6.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

3057/NRC GENERIC LETTER 88-01; 3060/RG 1.44; 23356/FINAL SER EPRI CH 5; 23376/NUREG 0933; 23382/NUREG 0933; 23406/FINAL SER CE80 CH 5; 23415/NUREG 0313; 25425/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 473      **SRP Section Number:** 10.3.6

#### **Suggested Changes to the SRP Section:**

Add Review Procedures covering the review of the use of corrosion/erosion-resistant materials in the steam and feedwater systems.

Pipe wall thinning and pipe wall rupture caused by erosion/corrosion in feedwater system piping and other power conversion systems are addressed in Generic Letter 89-08. NUREG-1344, which is an enclosure to Generic Letter 89-08, provides insight into the erosion/corrosion phenomena and recommendations for establishing controls to minimize its effects. The staff review of licensees' responses to Bulletin 87-01 indicates that the pipe wall thinning problem is widespread for single-phase and two-phase high-energy carbon steel systems.

In the EPRI Evolutionary Plant FSER, the staff concluded requirements including selective use of erosion/corrosion-resistant materials would ensure that materials in feedwater, steam, and condensate systems will perform as designed.

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In the CE 80+ DSER, the staff requested that the applicant propose specific materials and describe the methodologies for identifying corrosion/erosion susceptible locations and for selecting resistant materials. These open items (10.3-5 and 10.3-7) were closed in the CE 80+FSER based on selection of erosion/corrosion resistant materials and establishment of inspection programs that relied on EPRI NP-3944, "Erosion/corrosion in Nuclear Plant Steam Piping: Causes and Inspection Guidelines," for guidance. It was noted that EPRI NP-3944 has been accepted by the staff.

In the ABWR FSER, the staff noted that GE had committed to a monitoring program for erosion-corrosion that followed the guidance of NUREG-1344, "Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants."

Consideration should be given to adding Review Procedures to review use of corrosion/erosion-resistant materials in steam and feedwater systems based on the above staff positions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

9187/NRC GENERIC LETTER 89-08; 23269/FINAL SER CE80 CH 10; 23274/FINAL SER EPRI CH 1; 23564/FINAL SER CE80 CH 10; 24548/FINAL SER ABWR CH 3; 24549/NUREG 0933; 25474/NUREG 1344

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**Integrated Impact Number:** 474      **SRP Section Number:** 5.3.3

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to discuss Section 50.61 of 10 CFR 50 and PWR reactor vessel acceptability under pressurized thermal shock (PTS) conditions.

Acceptance Criteria II.2 identifies SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," and SRP Section 5.3.1, "Reactor Vessel Materials" as the source of criteria for the materials used in the reactor vessel, and the regulations that they satisfy. Subsection II.2 also discusses review of fracture toughness properties and radiation effects on the reactor vessel material nil-ductility transition reference temperatures.

Unresolved Safety Issue A-49 was initiated to assess the significance of simultaneous occurrence of a severe overcooling transient and a repressurization of a PWR primary system. The NRC resolved this issue for operating reactors with the publication of 10 CFR 50.61 which establishes a screening criterion for the reference temperatures for nil-ductility transition.

PNL has recommended that a review of pressurized thermal shock, as described in Generic

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Letter 92-1, be added to SRP Section 5.3.2.

Consider revising Acceptance Criteria to discuss section 50.61 of 10 CFR 50 and PWR reactor vessel acceptability under pressurized thermal shock conditions.

PNL COMMENT: Part A was revised on 6-5-95 to indicate that review of pressurized thermal shock will be added to SRP Section 5.3.2 instead of SRP Section 5.3.1 as previously indicated.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

2655/CODE OF FED. REGS 10CFR50; 6068/NUREG 0933; 21617/NRC GENERIC LETTER 92-01

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**Integrated Impact Number:** 475      **SRP Section Number:** 16.0

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to add citations for NUREGs representing improved Standard Technical Specifications (STS).

SRP 16.0 Acceptance Criteria cite older versions of vendor specific Standard Technical Specifications. The Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors (58 FR 39132) which became effective on July 22, 1993, identified the following as NRC-approved Standard Technical Specifications:

NUREG-1430, Babcock and Wilcox Plants;  
NUREG-1431, Westinghouse Plants;  
NUREG-1432, Combustion Engineering Plants;  
NUREG-1433, General Electric Plants, BWR/4; and  
NUREG-1434, General Electric Plants, BWR/6.

In the ABWR FSER and CE 80+ FSER, proposed technical specifications were compared to these NUREGs, as appropriate but standard technical specifications applicable for these designs were developed to address differences with respect to predecessor designs and unique features associated with these designs. The staff accepted new standard technical specifications applicable to these designs.

Consider revising Acceptance Criteria to add citations for the NUREG documents resulting from the Technical Specification Improvement Program.

TECHNICAL SPECIFICATIONS BRANCH REVIEW COMMENTS PROVIDED IN AN

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12/27/94 NOTE FROM C. GRIMES TO G. SUH:

The SRP could include references to the improved STS.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23640/ADVANCE SER ABWR CH 16; 23641/ADVANCE SER CE CH 16; 23642/NRC POLICY STATEMENT 58 FR 39132; 23667/NUREG 1430; 23668/NUREG 1431; 23669/NUREG 1432; 23670/NUREG 1433; 23671/NUREG 1434; 25949/FINAL SER CE80 CH 16

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**Integrated Impact Number:** 478      **SRP Section Number:** 5.3.3

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to incorporate 10 CFR 50.60.

At present, the SRP Acceptance Criteria include Appendices G and H of 10 CFR 50 with respect to reactor vessel fracture toughness and surveillance program. The regulatory requirement to comply with Appendices G and H is contained in 10 CFR 50.60 which is not currently identified in SRP 5.3.3.

Consider revising Acceptance Criteria to include 10 CFR 50.60 in connection with 10 CFR 50 Appendices G and H.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

2691/CODE OF FED. REGS 10CFR50; 2692/CODE OF FED. REGS 10CFR50; 23646/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 482      **SRP Section Number:** 10.3.6

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures for review of steam and feedwater component and system cleaning and cleanliness controls.

The SRP currently cites RG 1.37 and ANSI N45.2.1-1973 with regard to cleanliness and cleanliness controls. RG 1.37 also cites ANSI N45.2.1 in describing the staff's position with regard to cleanliness and cleanliness controls. In the CE 80+ FSER, the staff indicated that ANSI/ASME NQA-2 supersedes ANSI N45.2.1, cited in RG 1.37, and indicated that they had

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reviewed ANSI/ASME NQA-2-1983 and found it acceptable.

Consideration should be given to revising Acceptance Criteria and Review Procedures for steam and feedwater component and system cleaning and cleanliness controls to cite ANSI/ASME NQA-2-1983, in addition to RG 1.37. In addition, consideration should be given to revision of RG 1.37 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-3.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

Per 11/94 conversation with Quality Assurance and Maintenance Branch staff, N45.2.1 requirements are being incorporated into NQA-1 and NQA-2. RG 1.28, Revision 3 endorsed NQA-1. NRC has a program to revise the endorsement based on the results of an evaluation of the graded QA program. Also NQA is going through a review of both standards.

Per 11-10-94 telecon with Office of Research staff, two draft regulatory guides were prepared to endorse NQA-1 and NQA-2 through their 1993 addenda. Both regulatory guides were put on hold due to NRC/NEI work on the graded QA program. In the interim, NQA-1 and NQA-2 were consolidated into a new NQA-1.

This Integrated Impact will not be processed further, pending action by the staff regarding NQA-1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

9469/RG 1.37; 23268/FINAL SER CE80 CH 4; 23281/FINAL SER CE80 CH 17;  
23440/C&S: ANSI N45.2.1

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**Integrated Impact Number:** 483      **SRP Section Number:** 10.3.6

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of corrosion allowances proposed for steam and feedwater system materials.

In the EPRI Evolutionary Plant FSER, the staff noted that experience has shown previously accepted standard corrosion allowances to be inadequate and that the designer should control corrosion mechanisms in the design of primary and secondary piping systems. The staff also stated that the general corrosion allowance should comply with the allowance specified in Section III of the ASME Code and ANSI/ASME B.31.1, Power Piping. In the CE 80+ DSER, the staff requested that the applicant include a corrosion allowance for steam and feedwater system materials for the design life and provide a technical basis for the allowance. The

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requested information was provided, reviewed and found acceptable in the CE 80+ FSER.

Consideration should be given to developing Review Procedures for review of corrosion allowances proposed for steam and feedwater system materials based upon the above staff positions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23478/FINAL SER CE80 CH 10; 23480/DRAFT SER CE80 CH 10; 23482/FINAL SER EPRI CH 1; 25475/FINAL SER ABWR CH 5

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**Integrated Impact Number:** 485      **SRP Section Number:** 10.3.6

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address staff positions on prevention of stress corrosion cracking in austenitic stainless steel components including controls on grinding that are more restrictive than RG 1.37.

The SRP currently cites RG 1.37 which includes controls of surface preparation by manual grinding. In the EPRI Evolutionary Plant FSER, the staff reviewed a list of grinding controls applicable to BWRs which are more restrictive than RG 1.37. The staff found the controls to be adequate and noted they would be required for PWRs as well as BWRs.

In the CE System 80+ FSER, the staff discussed the above EPRI controls but did not explicitly take the position that they were applicable to the design (a PWR). The staff accepted the applicant's proposed grinding controls for austenitic stainless steel and noted the applicant's intent to avoid fabrication processes which would severely cold work the surface of austenitic stainless steel components.

Consideration should be given to revising Review Procedures for austenitic stainless steel steam and feedwater system materials. In addition, consideration should be given to revision of RG 1.37 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-5.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

9469/RG 1.37; 23377/NRC GENERIC LETTER 88-01; 23416/NUREG 0313; 23475/FINAL SER EPRI CH 1; 24557/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 487      **SRP Section Number:** 10.3.6

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for review of the adequacy of austenitic stainless materials to resist intergranular stress corrosion cracking (IGSCC).

Generic Issue A-42 addressed the issue of pipe cracking in the heat-affected zones of welds in primary system piping in BWRs. This issue was resolved with the initial issuance of NUREG-0313. NRC Generic Letter 88-01 and its supplement provide staff positions for minimizing the probability of cracking in BWR austenitic stainless steel and associated welds. The letter provides positions relating to acceptable base material properties, weld material ferrite content, and fabrication practices. GL 88-01 also transmitted Revision 2 to NUREG-0313, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping."

In the CE 80+ FSER, the staff noted that extraction steam piping, heater drain piping downstream of the drain control valves, and other piping exposed to wet steam or flashing liquid flow will be chromium-molybdenum alloy steel, stainless steel, or equivalent. In the ABWR FSER, the staff approved the use of NUREG-0313 for the control of IGSCC. In the EPRI Evolutionary Plant FSER, the staff recommended that licensees and applicants follow Rev. 2 of NUREG-0313 to prevent IGSCC in stainless steel.

Consideration should be given to revising Review Procedures to identify current guidance for review of austenitic stainless steel to resist IGSCC based upon Generic Letter 88-01, and NUREG-0313, Rev. 2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11085/NUREG 0933; 23270/FINAL SER CE80 CH 10; 23358/FINAL SER ABWR CH 5; 23359/FINAL SER EPRI CH 5; 23377/NRC GENERIC LETTER 88-01; 23416/NUREG 0313

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**Integrated Impact Number:** 488      **SRP Section Number:** 10.3.6

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to include RG 1.44 as a reference for the use of sensitized stainless steel. Provisions should also be made for staff positions more restrictive than RG 1.44.

SRP Section 10.3.6 does not cover the use of austenitic stainless steels. However, In the CE 80+ FSER, the staff noted that extraction steam piping, heater drain piping downstream of the drain



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control valves, and other piping exposed to wet steam or flashing liquid flow will be chromium-molybdenum alloy steel, stainless steel, or equivalent.

Similarly, SRP Section 5.4.2.1 does not cover the use of austenitic stainless steels. However, in the CE 80+ FSER, the staff reviewed the use of austenitic stainless steel in its SRP Section 5.4.2.1 review. The staff noted controls imposed in Sections 4.5.1, 4.5.2, and 5.2.3 for welding, fabrication, and water chemistry and carbon content of austenitic stainless steel were adequate to mitigate the occurrence of IGSCC.

RG 1.44 is cited in SRP 4.5.1, 4.5.2 and 5.2.3 for control of the use of sensitized stainless steel.

NRC Generic Letter 88-01 regarded implementation of new staff positions covering technical areas related to intergranular stress corrosion cracking in BWR austenitic stainless steel piping. GL 88-01 also transmitted Revision 2 to NUREG-0313, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." The resolution to New Generic Issue 119.4, "BWR Piping Materials," notes that updating of RG 1.44 to reflect the staff's findings in NUREG-0313, Rev. 2 is required.

Consider revising Review Procedures for austenitic stainless steel to include RG 1.44 as a reference and to address staff positions more restrictive than RG 1.44 as outlined in NUREG 0313, Rev. 2. In addition, consideration should be given to a revision of RG 1.44 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-6.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

9463/RG 1.44; 15764/NUREG 0933; 23270/FINAL SER CE80 CH 10; 23377/NRC GENERIC LETTER 88-01; 23577/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 489      **SRP Section Number:** 10.3.6

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to include RG 1.31 as a reference for adequacy of ferrite content limits specified for austenitic stainless weld metal. Provisions should also be made for staff positions that are more restrictive than RG 1.31.

SRP Section 10.3.6 does not cover the use of austenitic stainless steels. However, in the System 80+ FSER the staff noted that extraction steam piping, heater drain piping downstream of the drain control valves, and other piping exposed to wet steam or flashing liquid flow will be chromium-molybdenum alloy steel, stainless steel, or equivalent.

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SRP Section 5.2.3 cites RG 1.31 as criteria for control of delta ferrite content in stainless steel weld metal. Review Procedures for review of methods for controlling the amount of delta ferrite in stainless steel weld deposits also cite RG 1.31. RG 1.31 specifies ferrite content limits between 5 and 20% for weld metal.

Generic Letter 88-01 transmitted NUREG-0313, Rev. 2 and provides positions related to welding which are applicable to BWRs based upon the recommendations of NUREG-0313, Rev. 2.

In the CE System 80+ FSER, in conjunction with reviews of control rod drive structural and ESF materials, the staff compared the ferrite content limits for austenitic stainless steel castings and weld metal against industry guidelines and NUREG-0313, Rev. 2. NUREG-0313, Rev. 2 identifies a lower ferrite content limit of 7.5% which is more restrictive than the lower limit specified in RG 1.31.

In the CE System 80+ FSER, the staff also accepted the applicant's commitment to limit the maximum ferrite content for austenitic stainless steel weld metal to 15% and concluded that the lower limits specified will provide reasonable assurance that components of these materials will maintain adequate fracture toughness for their design life.

Consider revising Review Procedures for austenitic stainless steel to include RG 1.31 as a reference and to address staff positions outlined in NUREG-0313, Rev. 2 and industry guidelines. In addition, consideration should be given to revision of RG 1.31 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-7.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23270/FINAL SER CE80 CH 10; 23377/NRC GENERIC LETTER 88-01; 23576/RG 1.31; 23579/FINAL SER CE80 CH 5; 24559/NUREG 0313; 24560/FINAL SER CE80 CH 4; 24561/FINAL SER CE80 CH 6

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**Integrated Impact Number:** 490      **SRP Section Number:** 5.2.4

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to provide guidance for application of ASME Section XI, Subsection IWH for examination of piping susceptible to wall thinning due to erosion/corrosion.

In SRP Section 5.2.4, General Design Criterion (GDC) 32, "Inspection of Reactor Coolant Pressure Boundary," is met, in relevant part, through compliance with Section XI of the ASME Code, "Rules for Inservice Inspection of Nuclear Power Plant Components," as outlined in 10

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CFR Part 50, Section 50.55a.

In the CE 80+ FSER, the staff indicates that ASME Class 1, 2, and 3 carbon and low-alloy-steel piping items that are susceptible to wall thinning due to the single phase (water) erosion/corrosion phenomenon will be subject to examination in accordance with Subsection IWH of ASME Code Section XI.

In the CE 80+ FSER, the staff also noted that Subsection IWH had not been incorporated into the Code at the time of their review. The SER states that the COL applicant will review Subsection IWH after NRC approval and ensure that the System 80+ design programs comply accordingly. Upon approval, further analysis of necessary changes to the Acceptance Criteria may be required.

Consider modifying Review Procedures to provide guidance for application of ASME Section XI, Subsection IWH for examination of piping susceptible to wall thinning due to erosion/corrosion.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14116/CODE OF FED. REGS 10CFR50; 14125/CODE OF FED. REGS 10CFR50;  
14204/NRC GENERIC LETTER 89-08; 23775/NUREG 0933; 23776/NRC BULLETIN 87-01;  
23777/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 492      **SRP Section Number:** 5.2.4

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures and Acceptance Criteria to provide guidance for application of ASME Code Section XI, Appendices VII and VIII for qualification of personnel and performance demonstration of ultrasonic examination systems.

In SRP Section 5.2.4 General Design Criterion (GDC) 32, "Inspection of Reactor Coolant Pressure Boundary," is met, in relevant part, through compliance with Section XI of the ASME Code, "Rules for Inservice Inspection of Nuclear Power Plant Components," as outlined in 10 CFR Part 50, Section 50.55a.

The ASME has published in ASME Code Section XI, Division 1, Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." In both the ABWR FSER and the CE 80+ FSER, the staff indicates that the NRC has found both of these appendices to be appropriate. In the ABWR FSER, the staff requested that the applicant include provisions that

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ultrasonic testing be performed in accordance with Appendices VII and VIII. In the CE 80+ FSER, the staff indicated that the applicant should indicate that the Section XI requirements are to be augmented with the requirements in Appendices VII and VIII.

Upon final approval of these appendices, further analysis of potentially necessary changes to the Acceptance Criteria may be required.

Consider modifying Review Procedures to review application of ASME Section XI, Appendices VII and VIII for qualification of personnel and performance demonstration of ultrasonic examination systems.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14116/CODE OF FED. REGS 10CFR50; 14125/CODE OF FED. REGS 10CFR50;  
23773/FINAL SER ABWR CH 5; 23774/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 493      **SRP Section Number:** 5.2.4

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to include Regulatory Guide 1.150 as reference for ultrasonic testing of reactor vessel welds.

Generic Letter 83-15 transmitted Regulatory Guide 1.150, Revision 1, for implementation in preservice and inservice inspection programs. Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds during Preservice and Inservice Examination," describes procedures for conducting preservice and inservice examinations of reactor vessel welds by ultrasonic testing. In the ABWR FSER, the staff found use of Regulatory Guide 1.150 to be part of an acceptable ISI and PSI program.

Consideration should be given to revising existing Review Procedures to incorporate a citation of Regulatory Guide 1.150.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14308/RG 1.150; 23779/FINAL SER ABWR CH 5; 23838/NRC GENERIC LETTER 83-15

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**Integrated Impact Number:** 494      **SRP Section Number:** 5.2.4

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of BWR augmented inservice inspections developed to address intergranular stress corrosion cracking (IGSCC).

Generic Issue A-42 addressed the issue of pipe cracking in the heat-affected zones of welds in primary system piping in BWRs. As indicated in NUREG-0933, this issue was resolved with the initial issuance of NUREG-0313 which was transmitted to BWR license holders. NUREG-0313 contains the technical basis for staff positions with regards to IGSCC, including augmented inspections of piping and welds. NRC Bulletin 82-03, Bulletin 83-02 and Generic Letter 88-01 and its supplement provide staff positions for minimizing the probability of cracking in BWR austenitic stainless steel and associated welds. The bulletins and generic letter outline staff positions including guidelines for augmented inspections. Generic Letter 88-01 also transmitted Revision 2 to NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

Consideration should be given to developing Review Procedures for review of augmented inservice inspections developed in response to the above documents.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11075/NUREG 0933; 14168/NRC GENERIC LETTER 81-03; 14181/NRC GENERIC LETTER 88-01; 14407/NRC BULLETIN 82-03; 14415/NRC BULLETIN 83-02; 21798/NRC GENERIC LETTER 88-01 SUP 1; 23772/NUREG 0313

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**Integrated Impact Number:** 495      **SRP Section Number:** 3.9.3

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address Interfacing Systems Loss-of-Coolant Accidents (ISLOCA).

Generic Issue 105 was established to address the potential for ISLOCAs.

In SECY-90-016, dated January 12, 1990, the NRC staff stated that future evolutionary ALWR designs can reduce the possibility of a LOCA outside of containment by designing (to the extent practicable) all systems and subsystems connected to the RCS to an ultimate rupture strength at least equal to the full RCS pressure. In its June 26, 1990, Staff Requirements Memorandum (SRM), the Commission approved the staff's position provided that all elements of the low

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pressure system are considered. In the ABWR ASER and CE 80 + ASER, the staff described its review of the designs to the above position. In addition, in both ASERS, the staff described the detailed criteria used to verify compliance with the above position.

In the resolution of Generic Issue 105 contained in NUREG 0933, the staff noted that for existing plants there is little risk contribution from ISLOCA sequences for BWRs, and that while ISLOCAs at PWRs were plant-specific in nature, the ongoing IPE program (GL 88-20) includes licensee analysis of ISLOCA sequences. For future applicants the staff noted a new SRP Section had been proposed in Memorandum for F. Gillespie from W. Minners, "Proposed Resolution of Generic Issue 105, 'Interfacing Systems LOCA in LWRs,'" April 2, 1993.

Consider modifying the Review Procedures to incorporate the staff position that future designs should reduce the possibility of a LOCA outside of containment by designing (to the extent practicable) all systems and subsystems connected to the RCS to an ultimate rupture strength at least equal to the full RCS pressure. Consideration should also be given to developing a new SRP Section on interfacing system LOCA and appropriate review interfaces for SRP Sections that address systems with potential RCS interfaces. This future work recommendation will be tracked with IPD 7.0 Form 6.3-2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23040/SECY 90-016; 23789/ADVANCE SER CE CH 3; 23812/ADVANCE SER ABWR CH 3; 23832/NRC MEMORANDUM 04/02/93, from W. Minners to F. Gillespie; 23833/NRC MEMORANDUM 06/03/93, from E. Beckjord to J. Taylor; 23835/NUREG 0933

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**Integrated Impact Number:** 496      **SRP Section Number:** 3.9.3

#### **Suggested Changes to the SRP Section:**

Revise the existing loading combination discussion in Acceptance Criteria to allow single-earthquake design.

SRP Section 3.9.3 Acceptance Criteria refers to SRP Section 3.9.3 Appendix A, to define the acceptability of design and service loadings applicable to the design of Class 1, 2, and 3 components, component supports and core support structures. Appendix A specifies separate loading combinations that include the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE).

In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," the staff requested the Commission approve staff positions that decouple the level of the OBE ground motion from that of the SSE. The

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Commission approved the staff's position in its staff requirements memorandum (SRM) of June 26, 1990. In SECY-93-087, the staff recommended eliminating the OBE from the design of SSCs in advanced reactors. The Commission approved the staff position in its SRM dated July 21, 1993.

In the CE 80 + ASER and ABWR ASER, the staff found elimination of the OBE to be acceptable when the guidance provided in SECY-93-087 was followed. In addition, in the CE 80 + ASER, the staff noted it had provided NRC Letter of September 11, 1992, "Safety Evaluation on the Use of a Single Earthquake Design for Systems, Structures and Components in the Advanced Boiling Water Reactor," as a guidance document which identified the necessary changes to existing seismic design criteria.

Consider adding discussions of single-earthquake design to the existing Acceptance Criteria discussion for loading combinations based on the documents cited above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23052/SECY 93-087; 23787/ADVANCE SER CE CH 3; 23818/ADVANCE SER ABWR CH 3

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**Integrated Impact Number:** 497      **SRP Section Number:** 3.9.3

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address cumulative fatigue damage in ASME Code Class 2 or 3 or Quality Group D components.

In the EPRI Evolutionary Plant FSER, the staff noted the proposed design life of 60 years raises questions relative to the margins available to the current ASME fatigue design curves. In the ABWR ASER and the CE 80+ ASER, the staff noted that the margins built into the ASME fatigue design curves might not be sufficient to account for variations in the original fatigue test data because of various environmental effects. For ASME Code Class 2 and 3 piping for which a fatigue analysis is performed, the analysis shall consider environmental effects in the fatigue analysis. In Generic Issue 78, "Monitoring of Fatigue Transient Limits for Reactor Coolant System," the staff expressed a concern that repeated thermal cycling of RCS components produces some degree of fatigue degradation of the materials which could lead to failure.

In the ABWR ASER, the staff reviewed and accepted vendor provided supplemental guidelines specific to the ABWR design which enhanced the design margin beyond the requirements of the ASME Code, Section III for fatigue evaluation. The CE80+ ASER describes the vendor's position that the expected environmental conditions and lack of carbon or low alloy steel directly

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exposed to the primary coolant allowed use of the existing fatigue curves contained in ASME Code Section III. The staff found this position acceptable.

In both the ABWR ASER and the CE80+ ASER, the staff noted that it is assessing the potential generic implication of this issue on all operating plants, and depending on the severity of the issue, certain actions might be required to generically address this concern.

Consider modifying the Review Procedures to verify that applicants have addressed the issue of cumulative fatigue damage in ASME Code Class 2, 3 or Quality Group D components throughout the plant's proposed design life. This modification should identify that application of the ASME Code Class 1 requirements of Subsection NB of the ASME Code, Section III, is an acceptable approach for fatigue evaluation of Code 2, 3 or Quality Group D components.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23803/ADVANCE SER CE CH 3; 23807/ADVANCE SER ABWR CH 3; 23817/ADVANCE SER ABWR CH 3; 23822/FINAL SER EPRI CH 1; 23944/NUREG 0933

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**Integrated Impact Number:** 499      **SRP Section Number:** 3.9.3

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures and Appendix A to include NUREG-1367 as guidance for piping functional capability.

Position C.3 of Appendix A to SRP Section 3.9.3 states that the appendix provides guidance for consideration of piping functional capability and operability of active pumps and valves under service loading combinations resulting from postulated events. NUREG-1367, "Functional Capability of Piping Systems" was issued to evaluate whether existing Code rules, and potential changes to the Code rules, were sufficient to ensure maintenance of piping system functional capability. In the CE 80 + ASER, the staff found functional capability limits for piping acceptable provided that all of the NUREG-1367 conditions are met. In the ABWR ASER the staff noted GE's commitment that all ASME Code Class 1, 2, and 3 piping systems that are essential for safe shutdown under the postulated events are designed to meet the criteria of NUREG-1367, and noted NUREG-1367 contains methodology acceptable to the staff for ensuring the functional capability of essential piping systems.

Consideration should be given to incorporating NUREG-1367 as guidance in Appendix A and in the Review Procedures for functional capability.



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#### **Potential Impacts/Documents supporting the Suggested Changes:**

23788/ADVANCE SER CE CH 3; 23814/ADVANCE SER ABWR CH 3; 23954/NUREG 1367

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**Integrated Impact Number: 502      SRP Section Number: 15.2.8**

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to identify additional information for confirming the conservatism of mathematical models used in the Feedwater Line Break (FLB) analysis.

SRP Section 15.2.8 states that the analytical methods are reviewed to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. The values of parameters used in the analytical model are also reviewed,

NUREG/CR-4945 summarized significant results from the Semi scale test program. The document describes a concern that test data from Semi scale indicates that the CE 80 FSAR FLB analysis, although using previously accepted methodologies, appears to be nonconservative.

In the CE 80+ DSER, although the applicant used the previously approved CESEC-III code to analyze a spectrum of break sizes, the staff required the applicant to discuss the FLB method and justify that the method is conservative in comparison with the Semi scale test data from NUREG/CR-4945. If the method is found to be nonconservative, the applicant would be required to reanalyze the FLB event by using the model that is supported by test data, including the Semi scale data.

Consideration should be given to revising Review Procedures to reference NUREG/CR-4945 as a source of information for confirming the conservatism of mathematical models used in the feed water line break analysis.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23209/DRAFT SER CE80 CH 15

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**Integrated Impact Number: 503      SRP Section Number: 9.2.2**

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures to verify reactor coolant pump (RCP) seal integrity following an extended loss of seal injection or cooling.

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SRP Section 9.2.2 currently provides Review Procedures (III.3.g and III.4.d) addressing certain RCP seal cooling issues. These Review Procedures are related to verifying the applicant's compliance with GDC 44 (Acceptance Criteria II.4).

GI-23 addresses concerns related to RCP seal failures that could result in a small-break loss-of-coolant-accident. The potential for, and consequences of, RCP seal failures due to a loss of seal cooling in connection with certain multiple safety system failures has been considered during the evaluation of several issues including GI-65 (related to component cooling water), GI-130 (related to service water), and GSI A-44 (related to station blackout). The resolution of these issues recognized that further work on addressing the specific concern of RCP seal failures due to an extended loss of cooling is proceeding under GI-23. As indicated in NUREG 0933, GI-23 currently has a High priority.

Regulatory position C.3.3.4 of Regulatory Guide 1.155 specifies that a system required for reactor coolant pump seal cooling or injection specifically to meet the station blackout duration should be capable of being actuated and controlled from the control room, or if other means of control are required, it should be demonstrated that these steps can be carried out in a timely fashion, and if the system must operate within 10 minutes of a loss of all ac power, it should be capable of being actuated from the control room.

In its review of EPRI's proposal for the resolution of GI-23 for evolutionary plants, the staff indicated that they expect applicants to submit a design that provides for independent seal cooling during station blackout or to provide adequate testing of the proposed seal design to demonstrate integrity following extended loss of seal injection and cooling. In the CE 80+ DSER, the staff indicated that CE should either submit adequate test data to demonstrate seal integrity during a station blackout for an extended period or provide a diverse seal injection system, which should be a safety-grade system independent of the CVCS and associated support systems to the extent practicable.

Draft regulatory guide DG-1008 proposes guidance on the scope and content of actions to ensure that a plant has the capability to withstand loss-of-seal-cooling events, given the potential for failure of RCP seals. However, the draft regulatory guide is proposed guidance that has not been finalized.

Consideration should be given to developing Review Procedures to verify that evolutionary plant applicants either submit adequate test documentation to demonstrate seal integrity following extended loss of seal injection and cooling or submit a design that provides for independent RCP seal cooling.

PNL NOTE: Information on draft regulatory guide DG-1008 and PI-25121 was added on 6/6/95. Pending final issuance of the Regulatory Guide for use, the changes outlined above that

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are specific to draft regulatory guide DG-1008 could be considered a Type II change.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

10149/RG 1.155; 10386/NRC GENERIC LETTER 91-07; 22407/DRAFT SER CE80 CH 20; 22408/FINAL SER EPRI CH 1; 22614/NUREG 0933; 22615/NRC GENERIC LETTER 91-13; 22721/NUREG 0933; 22722/FINAL SER EPRI CH 3; 22723/FINAL SER EPRI CH 8; 25121/DRAFT RG DG-1008

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**Integrated Impact Number:** 504      **SRP Section Number:** 9.2.2

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to accommodate proposed designs where component cooling water (CCW) supply and return lines to the reactor coolant pumps do not isolate on a safety injection signal.

SRP Section 9.2.2 currently describes a review of RCP cooling issues associated with anticipated transients and DBAs (Review Procedures III.3.g and III.4.d). These reviews are related to verifying the applicant's compliance with GDC 44 and the positions described in TMI Action Plan Items II.K.2.16 and II.K.3.25 (Acceptance Criteria II.4).

The EPRI FSER describes the staff's review of EPRI's proposal that CCW to the reactor coolant pumps and motors not be isolated on an automatic containment isolation signal. The staff indicated that this proposal had merit on the basis that not requiring automatic isolation of pump cooling water, component degradation due to inadvertent or test actuation of containment isolation can be avoided and continued long-term pump operation in an actual event can be permitted. However, the staff recommended that EPRI include provisions to ensure that the main control room operator has the necessary information and bases to determine when it is appropriate to isolate the affected line by remote manual means and how fast the line should be isolated. The CE DSER indicates that in the CE 80+ design, supply and return headers for the RCP support systems do not isolate on a safety injection signal and that instrumentation was included in the design to alert the control room operators if a problem should occur in these lines.

Consider modifying Review Procedures to accommodate proposed designs where component cooling water (CCW) supply and return lines to the reactor coolant pumps do not isolate on a safety injection signal consistent with the information indicated above.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22616/DRAFT FINAL SER ABWR CH 20; 22617/DRAFT SER CE80 CH 9; 22714/FINAL SER EPRI CH 3

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**Integrated Impact Number: 505      SRP Section Number: 9.5.3**

#### **Suggested Changes to the SRP Section:**

This impact will not be processed further based upon PRB comments on the related draft revision of SRP Section 9.5.3.

Modify the SRP Section to address the application of Class 1E, seismic, and separation/redundancy criteria for the review of lighting systems and associated components and power supplies for new applications. The SRP currently does not include reviews of lighting systems and associated components for conformance to such design criteria.

In the SERs for evolutionary plants, the staff discussed design criteria for lighting systems serving safety-related areas and associated access routes. The staff indicated that portions of lighting systems serving such areas should be designed as Class 1E and meet requirements associated with Class 1E systems/equipment. The staff also asserted that certain portions of lighting systems should be designed and qualified as seismic Category I and that associated light fixtures should be qualified if possible or seismically supported as a minimum. Portions of AC lighting systems in such areas should be designed such that power supply circuits to individual light fixtures are staggered with the staggered circuits fed from separate electrical divisions and backed by appropriate onsite AC supplies (e.g. diesel generators or combustion turbine generator). At least one of the staggered lighting power supply circuits should be Class 1E. Control room emergency lighting should be powered from two redundant, independent safety uninterruptible power supplies to ensure that the system meets the single-failure criterion. In addition to these general criteria, the staff describes a number of positions in the evolutionary plant SERs regarding specific application of these criteria to lighting systems.

In sections of the SERs related to SRP Sections 8.3.1 and 8.3.2, the staff indicated that lighting loads should be considered for the purposes of sizing onsite sources (e.g. diesel-generators, batteries, etc.) and that lighting circuits should be identified/labeled and routed to conform with Class 1E circuit separation criteria.

Consideration should be given to modifying the SRP Section to address the application of Class 1E, seismic, and separation/redundancy criteria for the review of lighting systems and associated components and power supplies, for new applications, based upon information from the SERs for evolutionary plants.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

23094/FINAL SER CE80 CH 9; 23100/FINAL SER EPRI CH 11; 23103/FINAL SER EPRI CH 11; 23107/FINAL SER ABWR CH 8; 24569/FINAL SER ABWR CH 8

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**Integrated Impact Number: 506      SRP Section Number: 9.5.3**

#### **Suggested Changes to the SRP Section:**

Add Review Procedures and Areas of Review (Review Interfaces) with SRP Section 13.6 for review of the adequacy of evolutionary plant security lighting systems and associated power supply arrangement. SRP Section 13.6 includes Acceptance Criteria for security systems, including regulations applicable to security lighting (e.g. 10 CFR 73.55), but does not explicitly discuss review of security lighting systems. The SRP currently does not include explicit review of security lighting systems or their power supplies.

In the SERs for evolutionary plants, the staff reviewed proposed security lighting system designs and design criteria in conjunction with other plant lighting systems. The staff accepted the EPRI evolutionary plant proposed design criteria for security lighting systems and applied the criteria to the review of the proposed CE System 80+ security lighting system design. The staff also accepted design criteria proposed by CE which was not discussed in the EPRI Evolutionary Plant FSER. Design criteria accepted by the staff identifies portions of security lighting systems to be provided uninterruptible power (as opposed to alternate AC source backed power), identifies minimum acceptable illumination levels and lighting coverage, and addresses compatibility with security closed-circuit television systems. The staff indicated that the proposed requirements are consistent with the requirements of 10 CFR 73.55(c)(5) for security lighting.

Consideration should be given to adding Review Procedures and Areas of Review (Review Interfaces) with SRP Section 13.6 for review of the adequacy of security lighting systems and associated power supply arrangement, based upon the design criteria discussed in the evolutionary plant SERs.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23094/FINAL SER CE80 CH 9; 23095/FINAL SER CE80 CH 9; 23096/FINAL SER CE80 CH 9; 23098/FINAL SER EPRI CH 9; 23106/FINAL SER EPRI CH 11

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**Integrated Impact Number:** 507      **SRP Section Number:** 9.5.3

#### **Suggested Changes to the SRP Section:**

This impact will not be processed further based upon PRB comments on the related draft revision of SRP Section 9.5.3.

Add Review Procedures for review of evolutionary plant fixed, self-contained battery powered emergency and personnel safety lighting systems. The SRP does not provide overall reviews of the adequacy of such systems. SRP Section 9.5.1 provides reviews of the maintenance/testing of emergency lighting systems and their adequacy to support safe shutdown in the event of a fire and access/egress to/from fire areas.

In the SERs for evolutionary plants, the staff reviewed proposed fixed, self-contained battery powered emergency and personnel safety lighting system design criteria and testing requirements in conjunction with other plant lighting systems. In the EPRI Evolutionary Plant FSER, the staff accepted proposed design criteria for self-contained emergency and personnel safety lighting which address seismic design, battery capacity, minimum areas of coverage, and minimum illumination provided at locations where emergency operations involving reading are performed. In the ABWR FSER, the staff also accepted the applicant's proposed design and requirements for periodic inspection, testing, and bulb replacement for the fixed, self-contained battery powered guide lamp system. In the CE System 80+ FSER, the staff indicated (related to review of lighting systems) that the staff will verify that the completion of hazard analyses and plant emergency procedures are included as commitments and that appropriate inspections, tests, and/or analyses are included in ITAAC to verify implementation of design commitments.

Consideration should be given to adding Review Procedures for review of fixed, self-contained battery powered emergency and personnel safety lighting systems, based upon information provided in evolutionary plant SERs.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23094/FINAL SER CE80 CH 9; 23102/FINAL SER EPRI CH 11; 23103/FINAL SER EPRI CH 11; 23105/FINAL SER EPRI CH 11; 23107/FINAL SER ABWR CH 8

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**Integrated Impact Number:** 509      **SRP Section Number:** 15.2.7

#### **Suggested Changes to the SRP Section:**

Modify Acceptance Criteria and Review Procedures to include General Design Criterion (GDC) 17, "Electric Power Systems" and staff guidance for the assumption of the loss of offsite power

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(LOOP), in addition to the limiting single-failure event, for the analysis of transients and accidents. This section reviews the sequence of events, analytical model and input thereto, and the predicted consequences of a loss of normal feedwater flow (LFW).

In the CE 80+ FSER, the staff stated that, in accordance with the requirements of GDC 17, LOOP should not be considered as a single-failure event, but should be assumed in the analysis for each event without changing the event category. In addition, the staff stated that the applicant is required to discuss each of the transient and accident analyses to justify that the analyses meet the GDC 17 requirements.

Consider modifying Acceptance Criteria to include GDC 17, and Review Procedures to incorporate staff guidance for the assumption of the LOOP, in addition to the limiting single-failure event, for the analysis of transients and accidents.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22920/DRAFT SER CE80 CH 15; 22921/CODE OF FED. REGS 10CFR50; 22922/DRAFT SER CE80 CH 15

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**Integrated Impact Number:** 511      **SRP Section Number:** 6.1.2

#### **Suggested Changes to the SRP Section:**

Substitute ASTM D3842 and ASTM D3911 for citation of ANSI N101.2-1972. ANSI N101.2-1972 is cited under Acceptance Criteria in SRP Section 6.1.2 and listed as Reference 2, but the standard has been withdrawn.

In the Evolutionary Plant FSER, the staff encouraged EPRI to revise the Evolutionary Plant URD to invoke ASTM D3842-1980, "Standard Guide for Selection of Test Methods for Coatings for Use in Light Water Nuclear Power Plants," for the qualification of paints and coatings exposed to the containment atmosphere. ASTM D3842-1980 was revised in 1986 and reaffirmed in 1991.

In the CE80+ DFSE, the staff endorsed the CE proposal that containment coatings be selected based on ASTM D3842-1986 and ASTM D3911-1989, "Standard Test Method for Evaluating Coatings Used in Light Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions." The staff has reviewed ASTM D3842-86 and ASTM D3911-89 and found them acceptable. In the CE80+ FSER, the staff noted "These ASTM specifications represent the current industry technology and experience and are intended to replace the ANSI standards referenced in ANSI N101.2."

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Consideration should be given to substituting ASTM D3842 and ASTM D3911 in place of ANSI N101.2-1972.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22867/FINAL SER EPRI CH 6; 22869/C&S: ANSI N101.2-72; 23178/DRAFT SER CE80 CH 6; 23188/C&S: ASTM D3842; 23189/C&S: ASTM D3911; 24464/FINAL SER CE80 CH 6

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**Integrated Impact Number: 512      SRP Section Number: 4.3**

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to verify a non-positive moderator temperature coefficient (MTC) over the entire fuel cycle when the reactor is critical. This SRP Section reviews the nuclear design for conformance to relevant General Design Criteria (GDC). In particular, reactivity coefficients are reviewed for conformance to the requirements of GDC 11, "Reactor Inherent Protection."

While there are no explicit criteria to prohibit a positive MTC, such as may exist in a PWR at the beginning of core life, the staff has accepted positive MTCs based on the results of transient and accident analyses. In the EPRI Evolutionary Plant FSER and the CE 80+ FSER, the staff explicitly stated its intent to require a non-positive moderator temperature coefficient (MTC) over the entire fuel cycle when the reactor is critical.

Consider modifying Review Procedures to include explicit verification, for new applicants, of a non-positive MTC over the entire fuel cycle when the reactor is critical.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22957/FINAL SER CE80 CH 4; 22958/FINAL SER EPRI CH 4; 25601/FINAL SER ABWR CH 4

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**Integrated Impact Number: 513      SRP Section Number: 3.5.1.1**

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to address missile protection features, such as those concerning pressurized gas bottles, that are outside of the design certification scope. In addition, consideration should be given to developing a Review Procedure concerning the control of pressurized gas bottles that would be applicable to a broader class of plants.



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The Final SER for the ABWR specifically addresses reviews for design details that are the responsibility of the combined license applicant. The review documented in the Final SER indicated that the combined license applicant should provide design details of all pressurized gas bottles as well as details of missile protection features for structures systems and components (SSCs) that are outside of the ABWR scope. This issue was designated a COL Action item. In addition, the NRC has issued a broader class of documents providing information on pressurized gas bottle missile hazards. NRC Inspection Manual Chapter 71707, NRC Notice 91-37, and NUREG/CR-3551 provide information that can be used to develop a Review Procedure addressing pressurized gas bottle missile hazards which would be applicable to a broader class of plants.

Consideration should be given to adding a Review Procedure and an associated Evaluation Finding to address missile protection features, including those concerning pressurized gas bottles, that are outside of the design certification scope. In addition, consideration should be given to developing a Review Procedure to address the control of pressurized gas bottles for a broader class of plants.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22815/FINAL SER ABWR CH 3; 24278/NRC NOTICE 91-37; 24281/NUREG CR-3551;  
24282/INSPECTION MANUAL CH 71707

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**Integrated Impact Number: 514      SRP Section Number: 3.5.1.1**

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to address the use of probabilistic criteria in evaluating the need to provide missile protection.

The ABWR FSER documents a review method utilizing a combined probability to determine the statistical significance of an identified missile. Once a potential missile is identified its statistical significance is determined by calculating the combined probability of missile occurrence, impacting a significant target, and causing significant damage. If the combined probability is less than  $10^{-7}$  per year, the missile is not considered significant, if the combined probability is greater than  $10^{-7}$  per year, missile protection of safety-related SSCs is provided by one or more of the following: (1) locating the system or component in a missile-proof structure, (2) separating redundant systems or components for the missile path or range, (3) providing local shields and barriers for systems and components, (4) designing the equipment to withstand the impact of the most damaging missile, (5) providing design features to prevent the generation of missiles, (6) orienting missile sources to prevent missiles from striking equipment important to safety.

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Consideration should be given to adding a new Review Procedure to address the use of probabilistic criteria in evaluating the need to provide missile protection.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24279/FINAL SER ABWR CH 3

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**Integrated Impact Number: 515      SRP Section Number: 3.5.3**

#### **Suggested Changes to the SRP Section:**

Revise REVIEW PROCEDURES and REFERENCES to add a citation for Regulatory Guide 1.115 as a source of guidance for evaluation of missile barriers.

RG 1.115 describes methods acceptable to the NRC staff for protecting safety-related structures, systems, and components against low-trajectory missiles resulting from turbine failure by appropriate orientation and placement of the turbine-generator set. References are cited which may be useful for barrier design that are not cited in SRP 3.5.3.

Consideration should be given to revising Review Procedures and References of Section 3.5.3 to add Regulatory Guide 1.115 as a source of guidance for evaluation of missile barriers.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

17326/RG 1.115

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**Integrated Impact Number: 516      SRP Section Number: 2.4.10**

#### **Suggested Changes to the SRP Section:**

ANSI N170 is cited in SRP Section 2.4.10. This standard is cited only as Reference 8 in subsection VI. Reg. Guide 1.59, which is cited as guidance for SRP Section 2.4.10 reviews, endorses ANSI N170-1976 with exceptions. The current version of this standard is ANSI/ANS 2.8-1992. Consideration should be given to performing a detailed side-by-side comparison between ANSI N170-1976 and ANSI/ANS 2.8-1992 to allow SRP reviewers to use the more current version of the standard.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

ANSI N170

No comparison needed. Per 11/94 conversation with Civil Engineering and Geosciences Branch

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staff, Part 100 is under revision and has been issued for public comment. A regulatory guide is being written. Because of related ongoing work, do not perform comparison at this time.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20394/RG 1.59; 22817/C&S: ANSI N170; 22818/C&S: ANS 2.8

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**Integrated Impact Number: 517      SRP Section Number: 2.4.11**

#### **Suggested Changes to the SRP Section:**

Update citation of an existing guidance document to reflect the approval and issuance of the document.

SRP Section 2.4.11 Review Procedures cite Reference 25 for reviews of spray pond analysis techniques for determining adequate water inventory, for selecting minimum heat transfer conditions, and for performing transient analyses. At the time of issuance of SRP Section 2.4.11, Reference 25 was not identified by document number in subsection VI, References. This document was approved and issued in August, 1981, as NUREG-0733. NUREG-0733 develops models which can be utilized in the design of certain types of spray ponds used in ultimate heat sinks at nuclear power plants, and ways in which the models may be employed to determine the design basis.

Consideration should be given to updating Reference 25 to reflect the approval and issuance of NUREG-0733.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22833/NUREG 0733

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**Integrated Impact Number: 518      SRP Section Number: 2.4.11**

#### **Suggested Changes to the SRP Section:**

SRP Section 2.4.11 currently reviews the capability of the ultimate heat sink to provide adequate cooling water under normal and emergency conditions.

10 CFR 52 provides requirements for site parameter envelopes that are to be included in applications for design certifications and manufacturing licenses for standard plant designs. Applications which reference standard plant designs approved under 10 CFR 52 are required to address the conformance of site specific parameters with the site parameter envelope for the

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approved, certified standard design. In SERs documenting staff review of evolutionary plant applications for design certification, the staff addressed the requirements related to the site parameter envelope in Section 2.6.

Consideration should be given to developing a new SRP Section for review of the site parameter envelope associated with standard plant applications, as a candidate for future work.

Consideration should also be given to revising existing SRP Sections for review of site-specific parameters to reflect the site parameter-related requirements of 10 CFR 52, for applications referencing a standard plant design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

13289/CODE OF FED. REGS 10CFR52; 23086/DRAFT SER CE80 CH 2

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**Integrated Impact Number:** 519      **SRP Section Number:** 2.4.12

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to reflect the amended construction/design defect reporting requirements of 10 CFR Part 50, Section 50.55. SRP Section 2.4.12 currently cites 10 CFR Part 50, Section 50.55 as Acceptance Criteria for review of permanent dewatering systems whose design may substantially change based upon site characteristic data which becomes available after issuance of a Construction Permit (CP).

As cited in Acceptance Criteria, subsection II.1, 10 CFR Part 50, Section 50.55 required that significant deficiencies in construction of or significant damage to a structure, system, or component which will require extensive redesign, or extensive repair to meet the criteria of the construction permit be reported to the Commission.

In July, 1991, the Commission amended 10 CFR Part 50, Section 50.55, paragraph (e) to require that a CP holder who obtains information reasonably indicating the existence of any defect found in construction or any defect found in the final design of a facility as approved and released for construction notify the Commission of the defect. The notification requirements of 10 CFR Part 50, Section 50.55 apply to all defects and failures to comply associated with a substantial safety hazard regardless of whether extensive evaluation, redesign, or repair is required to conform to the criteria and bases stated in the safety analysis report or CP. Note that the paragraph numbering of the regulation was also amended with respect to current Branch Technical Position HGEB-1 citations.

Consideration should be given to revising Acceptance Criteria to reflect the amended construction/design defect reporting requirements of 10 CFR Part 50, Section 50.55.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22845/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 520      SRP Section Number: 2.4.12**

#### **Suggested Changes to the SRP Section:**

SRP Section 2.4.12 currently reviews site groundwater conditions, onsite use of groundwater, design bases for groundwater levels, and hydrodynamic effects of groundwater on safety-related structures and components.

10 CFR 52 provides requirements for site parameter envelopes that are to be included in applications for design certifications and manufacturing licenses for standard plant designs. Applications which reference standard plant designs approved under 10 CFR 52 are required to address the conformance of site-specific parameters with the site parameter envelope for the approved, certified standard design. In SERs documenting staff review of evolutionary plant applications for design certification, the staff addressed the requirements related to the site parameter envelope in Section 2.6.

Consideration should be given to developing a new SRP Section for review of the site parameter envelope associated with standard plant applications, as a candidate for future work. Action will be tracked on IPD-7.0 Form 2.3.1-1. Consideration should also be given to revising existing SRP Sections for review of site-specific parameters to reflect the site parameter-related requirements of 10 CFR 52, for applications referencing a standard plant design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21027/CODE OF FED. REGS 10CFR52; 23087/DRAFT SER CE80 CH 2; 23088/CODE OF FED. REGS 10CFR52

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**Integrated Impact Number: 521      SRP Section Number: 3.3.1**

#### **Suggested Changes to the SRP Section:**

Update and add further guidance for review of methods of calculating the pressure loading profile imposed upon structures by high winds. SRP Section 3.3.1 currently cites ANSI A58.1 (without reference to version/date) and ASCE paper No. 3269 as guidance for review of procedures for transforming design wind velocity into pressure loading imposed upon structures. ANSI A58.1-1972 was the version of the standard in effect at the time of issuance of SRP Section 3.3.1.

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In the CE System 80+ DSER, the staff found that the applicant's procedures for determining the design wind loadings on structures using ANSI A58.1-1982 and ASCE Papers 3269 and 4933 are acceptable.

The current version of ANSI A58.1 has been revised and redesignated as ANSI/ASCE- 7-1988. PNL has proposed a detailed code comparison to evaluate the differences between cited and current versions of the standard. This proposal is being evaluated by the staff.

Consideration should be given to updating citations of ANSI A58.1 to reflect the results of the code comparison, if performed. In addition, consideration should be given to adding citation of ASCE Paper 4933 as guidance for review of methods of calculating the pressure loading profile imposed upon structures by high winds.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22644/DRAFT SER CE80 CH 3; 22648/C&S: ANSI A58.1; 22651/C&S: ASCE 7-88

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**Integrated Impact Number: 522      SRP Section Number: 3.3.2**

#### **Suggested Changes to the SRP Section:**

Update and add further guidance for review of methods of calculating the pressure loading imposed upon structures by tornadic winds. SRP Section 3.3.2 currently cites ANSI A58.1 (without reference to version/date) and ASCE Paper No. 3269 as guidance for review of procedures for transforming design basis tornado wind velocity into pressure loading imposed upon structures. ANSI A58.1-1972 was the version of the standard in effect at the time of issuance of SRP Section 3.3.2.

The CE System 80+ DSER provides a position that the procedures used to determine the design wind loadings on structures using ANSI A58.1-1982 and ASCE Papers 3269 and 4933 to transform the wind velocity into an effective pressure on structures are acceptable.

The current version of ANSI A58.1 has been revised and redesignated as ANSI/ASCE- 7-1988. PNL has proposed detailed code comparisons to evaluate the differences between cited and current versions of the standard. This proposal is being evaluated by the staff.

Consideration should be given to updating citations of ANSI A58.1 to reflect the results of the code comparisons, if performed. In addition, consideration should be given to adding citation of ASCE Paper 4933 as guidance for review of methods of calculating the pressure loading imposed upon structures by tornadic winds.

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### **Integrated Impacts**

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22649/C&S: ANSI A58.1; 22650/C&S: ASCE 7-88; 22656/DRAFT SER CE80 CH 3

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**Integrated Impact Number: 525      SRP Section Number: 12.2**

#### **Suggested Changes to the SRP Section:**

ANSI N237-1976 is cited in the Acceptance Criteria subsection as guidance related to the establishment of typical long-term concentrations of principal radionuclides in fluid streams of light-water-cooled nuclear power plants. The standard has been superseded by ANS 18.1, the latest version of which is dated 1984. A side-by-side comparison for these standards is currently scheduled by PNL.

Pending completion and review of the side-by-side comparison, revise the citation to the latest standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22848/C&S: ANSI N237-76; 23108/C&S: ANS 18.1; 23109/ADVANCE SER ABWR CH 12

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**Integrated Impact Number: 526      SRP Section Number: 5.4.2.2**

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure for improved eddy current testing.

Reg. Guide 1.83 is currently identified in the SRP as guidance for the review of steam generator (SG) tube ISI programs. Various USIs/GSIs (A-3, A-4, A-5, 66, and 67) were initiated with regard to steam generator issues and were incorporated in GI-135. One of the subissues addressed under GI-135 is improved steam generator tube eddy current testing.

In the CE 80+ FSER, the staff indicated that the improved eddy current testing issue of GI-135 was being deferred to the development of a new revision of Reg. Guide 1.83, and that improved eddy current testing will be addressed on a plant specific basis via review of the steam generator tube ISI program in conjunction with applicable surveillance requirements defined in the Technical Specifications.

In the EPRI FSER, the staff indicated that it requested EPRI to address improved eddy current testing in their proposed resolution of GI-67.

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Consideration should be given to incorporating a Review Procedure to confirm implementation of improved eddy current testing in connection with GI-135. Consideration should also be given to completing the Reg. Guide 1.83 revision activity including the incorporation of positions relative to improved eddy current testing. An IPD 7.0 form (number 5.4.2.2-1) has been issued for this recommendation.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1474/NUREG 0933; 1480/NUREG 0933; 1486/NUREG 0933; 14678/NUREG 0933;  
14950/RG 1.83; 15164/NUREG 0933; 15190/NUREG 0933; 15482/NUREG 0933;  
22172/FINAL SER CE80 CH 20; 22173/FINAL SER EPRI CH 3

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**Integrated Impact Number:** 527      **SRP Section Number:** 5.4.2.2

#### **Suggested Changes to the SRP Section:**

Update existing references to Standard Technical Specifications (STS).

Standard Technical Specifications (NUREGs 0212, 0452, and 0103) are currently identified in the SRP as part of the guidance for the review of SG tube ISI programs. Later versions of STS have been developed (NUREGs 1430, 1431, 1432). The STS cited in the SRP as guidance contain specific criteria for steam generator tube inspections including eddy current test acceptance criteria, results categorization, inspection frequencies, and supplemental inspections. The more current versions of STS do not contain specific steam generator tube inspection program criteria.

In the CE System 80+ FSER, the staff discusses its review and acceptance of new Standard Technical Specifications developed in conjunction with the certified design.

Consideration should be given updating references to STS to include the current versions of STS. Consideration should also be given to evaluating the approach to be taken with regard to review guidance in the case where an applicant references the current versions of STS.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22169/NUREG 1430; 22170/NUREG 1431; 22171/NUREG 1432; 25885/FINAL SER CE80  
CH 16

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### **Integrated Impacts**

**Integrated Impact Number: 528      SRP Section Number: 5.4.2.2**

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure for steam generator tube inspection recommendations in Generic Letter 85-02.

USIs A-3, A-4, and A-5 were initiated with regard to steam generator tube integrity. Resolution of these USIs addressed, in part, steam generator tube inspection programs. These USIs were resolved without new requirements based on the safety significance of these issues and licensee responses to Generic Letter 85-02. Generic Letter 85-02 documents staff recommended actions resulting from study of these USIs including recommendations related to steam generator tube ISI. The CE 80+ FSER indicates that the staff reviewed the CE 80+ submittal for conformance with the recommendations of Generic Letter 85-02.

Consideration should be given to adding a Review Procedure to determine conformance with the portions of Generic Letter 85-02 applicable to steam generator tube inspection.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1474/NUREG 0933; 1480/NUREG 0933; 1486/NUREG 0933; 14965/NRC GENERIC LETTER 85-02; 22174/FINAL SER CE80 CH 20; 24520/NUREG 0844; 25634/NRC GENERIC LETTER 95-03

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**Integrated Impact Number: 529      SRP Section Number: 5.4.2.2**

#### **Suggested Changes to the SRP Section:**

Add 10 CFR 50.55a to Acceptance Criteria and associated Review Procedure.

10 CFR 50.55a establishes, in part, ISI requirements for ASME Code Class 1 (Quality Group A) components, including steam generator tubes, and establishes the applicability of ASME Section XI inspection requirements to these components. 10 CFR 50.55a(b)(2)(iii) specifically addresses steam generator tubes and describes the relationship between certain ASME Code inspection requirements and inspection requirements implemented through Technical Specifications.

Consideration should be given to identifying 10 CFR 50.55a (and as a result, ASME Section XI) as Acceptance Criteria for SRP Section 5.4.2.2. Consideration should also be given to adding a Review Procedure for determining compliance with the applicable portions of ASME Section XI related to the steam generator tube ISI program.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

14935/CODE OF FED. REGS 10CFR50; 14936/CODE OF FED. REGS 10CFR50;  
14937/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 531      SRP Section Number: 5.4.2.2**

#### **Suggested Changes to the SRP Section:**

Add Review Procedure related to steam generator tube ISI for 24 month fuel cycle applications.

Standard Technical Specifications (NUREGs 0212, 0452, and 0103) and Reg. Guide 1.83 are currently identified in the SRP as the primary guidance for the review of SG tube ISI programs. Inspection intervals for steam generator tube ISI are described in these documents. Generic Letter 91-04 was issued to provide guidance with regard to changes in Technical Specifications in support of a 24 month fuel cycle. This letter provides specific guidance with regard to steam generator tube ISI intervals for a 24 month fuel cycle in relation to the intervals specified in Reg. Guide 1.83 and the STS documents.

Consideration should be given to adding a Review Procedure to identify the guidance in Generic Letter 91-04 related to steam generator tube inspection intervals for 24 month fuel cycle applications.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14950/RG 1.83; 14968/NRC GENERIC LETTER 91-04

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**Integrated Impact Number: 532      SRP Section Number: 5.4.2.2**

#### **Suggested Changes to the SRP Section:**

Identify Reg. Guide 1.121 as a guidance document in the Acceptance Criteria subsection and add associated Review Procedure.

Reg. Guide 1.121 guide describes a method acceptable to the staff for establishing the limiting safe conditions of degraded steam generator tubing, beyond which defective tubes as established by inservice inspection should be removed from service. The applicants steam generator tube ISI program is reviewed in Section 5.4.2.2 which includes the applicant's proposed actions when defects are observed.

Consideration should be given to identifying Reg. Guide 1.121 as a guidance document for

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reviewing steam generator tube ISI programs. Consideration should also be given adding a Review Procedure to verify that tube plugging criteria described in the applicants ISI program conforms with the guidance of Reg. Guide 1.121 or that acceptable alternative criteria have been established.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14951/RG 1.121

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**Integrated Impact Number:** 533      **SRP Section Number:** 2.2.1 - 2.2.2

#### **Suggested Changes to the SRP Section:**

Incorporate Review Procedures to include positions on storage of liquid hydrogen, liquid oxygen, and propane.

The review in SRP 2.2.1 - 2.2.2 focuses on potential external hazards or hazardous materials that are present or which may be present during the lifetime of the plant. The purpose of this review is to establish the information concerning the presence and magnitude of potential hazards so that the reviews and evaluations described in other SRP sections can be performed. Generic Issue 136 was established to address concerns with storage and use of large quantities of combustibles on plant sites due to new plant features. This issue notes that guidance currently provided in SRP Sections 2.2.1 - 2.2.2 and Regulatory Guide 1.91 related to these concerns may not be adequate.

Staff positions regarding the onsite use and storage of large volumes of propane were developed during the staff review of a Dresden 2/3 license amendment request for the onsite use of a mobile volume reduction system (MVRS). Staff positions regarding the onsite use and storage of large volumes of liquid hydrogen and liquid oxygen are part of the staff's review of the EPRI topical report on BWR hydrogen water chemistry control. The resolution of Generic Issue 136 indicates that the staff positions established for these specific reviews are suitable for general application.

Consider incorporating Review Procedures to include appropriate portions of the staff positions developed through the Dresden 2/3 MVRS review and the EPRI hydrogen water chemistry control guidelines topical report review. In addition, revisions to Regulatory Guide 1.91 should be considered a candidate for future work as indicated on IPD-7.0 Form No. 2.2.1-1.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

14687/NUREG 0933

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**Integrated Impact Number:** 534      **SRP Section Number:** 10.4.2

#### **Suggested Changes to the SRP Section:**

Replace the citation to Regulatory Guide 1.123 with Regulatory Guide 1.28, as appropriate.

In a June 17, 1991 letter, the NRC withdrew Regulatory Guide 1.123 because it had become obsolete. The ANSI standard endorsed by Regulatory Guide 1.123 (N45.2.13-1976) has been incorporated into ANSI/ASME NQA-1-1983 which is endorsed by Regulatory Guide 1.28, Rev. 3, issued in August 1985.

Consider replacing the citation to Regulatory Guide 1.123 in the acceptance criteria subsection with one to Regulatory Guide 1.28 following an evaluation of the continued applicability of the new standard to SRP Section 10.4.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

6215/RG 1.123; 24073/RG 1.28

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**Integrated Impact Number:** 535      **SRP Section Number:** 3.5.1.2

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to address the controls provided to ensure equipment undergoing maintenance and ancillary support equipment used during maintenance does not become an internally generated missile.

The Final SER for the ABWR specifically addresses COL action items to ensure that equipment undergoing maintenance will be removed from containment during operation or will be seismically restrained to prevent it from becoming a missile. The COL applicant will have to provide controls that ensure all equipment inside containment that is required during maintenance will either be removed prior to operation, moved to a location where it is not a potential hazard to safety-related equipment, or seismically restrained to prevent it from becoming a missile. NRC Notice 80-21 contains a discussion on the control of non-seismic ancillary equipment (dollys, gas bottles, block and tackle gear, ductwork, etc.) and the potential for dislodging, impacting, and damaging safety related equipment during an earthquake.

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Consideration should be given to adding a Review Procedure and an associated Evaluation Finding to address the controls provided to ensure equipment undergoing maintenance and ancillary support equipment does not become an internally generated missile.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22816/FINAL SER ABWR CH 3; 24292/NRC NOTICE 80-21

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**Integrated Impact Number:** 536      **SRP Section Number:** 10.4.6

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to reference EPRI NP-4947-SR on hydrogen water chemistry.

In the ABWR FSER, the staff states that GE should state conformance with EPRI NP-4947-SR, "BWR Hydrogen Water Chemistry Guidelines," (1987 Revision, October 1988) in addition to the positions stated in Regulatory Guide 1.56.

Consider adding a reference to EPRI NP-4947-SR in review procedures.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24095/FINAL SER ABWR CH 10

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**Integrated Impact Number:** 537      **SRP Section Number:** 10.4.6

#### **Suggested Changes to the SRP Section:**

Update the Reference to ASTM D2187-71 to the latest version.

Regulatory Guide 1.56 is used as acceptance criteria in SRP Section 10.4.6 for the design of condensate demineralizer systems. In Regulatory position C.3, RG 1.56 endorses use of a portion of ASTM D2187-71 for the measurement of total exchange capacity of cation resin. The latest version of this standard is dated 1993.

Consider updating the citation to ASTM D2187 from the 1971 version to the 1993 version. A detailed standard comparison of the two versions is needed to support such a change. A related Regulatory/Research Needs Form (IPD 7.0 Form 10.4.6-1) has been initiated for a conforming change to RG 1.56.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24096/RG 1.56

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**Integrated Impact Number: 539      SRP Section Number: 3.8.1**

#### **Suggested Changes to the SRP Section:**

Add a discussion of RG 1.35.1 to Section 3.8.1.II.7, "Testing and Inservice Surveillance Requirements."

The recommended inservice inspection programs of prestressed concrete containment structures are described in RG 1.35, "Inservice Inspection of UngROUTed Tendons in Prestressed Concrete Containment Structures," Rev. 3, July 1990.

RG 1.35 recommends the comparison of measured prestressing forces with the predicted forces of randomly selected tendons. The predicted forces at a given time are based on a measurement of prestressing forces during installation minus the losses in the prestressing forces that were predicted to have occurred since that time because of material and structural characteristics. Due to complexity of the problem, it is unlikely that the measured prestress force will closely agree with the predicted value.

RG 1.35.1 clarifies the NRC's position on tolerance bands for groups and subgroups of tendons so that the small-sample inspection program of RG 1.35 can provide better confidence in the integrity of prestressing tendons. RG 1.35.1 also supplements the ASME "Code for Concrete Reactor Vessels and Containments," ACI-359, 1986, by describing in more detail the factors to be considered in determining the effective prestress in tendons, the measured prestressing force and determination of the prestress losses.

Consider modifying Acceptance Criteria II.7 to include a discussion of the guidance provided in RG 1.35.1.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

9887/RG 1.35.1

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**Integrated Impact Number: 540      SRP Section Number: 3.8.1**

#### **Suggested Changes to the SRP Section:**

Revise SRP Section 3.8.1 to comply with the 10 CFR 50.34.

Code of Fed. Regs 10 CFR 50.34, "Contents of applications; technical information," Subparagraph (f)(3)(v), covers requirements for containment design.

Subarticle (f)(3)(v) requires meeting the provisions of the ASME B&PV Code, Section III, Division 2, Subarticle CC-3720, Factored Load Category, considering pressure and dead load alone, during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide as the inerting agent.

Consider modifying SRP Section 3.8.1 Parts I, "Areas of Review;" II, "Acceptance Criteria;" III, "Review Procedures;" IV, "Evaluation Findings;" and V, "References" to conform with the above requirements.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20930/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 542      SRP Section Number: 3.8.1**

#### **Suggested Changes to the SRP Section:**

Update SRP Sections 3.8.1.II.7, "Testing and Inservice Surveillance Requirements," Section 3.8.1.I.2, "Applicable Codes, Standards and Specifications," and Section 3.8.1.VI, "References."

SRP Section 3.8.1.IV, "Evaluation," lists RG 1.136 as one of the documents that containment design and construction must be in compliance with. It is also referenced in SRP 3.8.1.VI, "References."

With the issuance of RG 1.136 the following RGs have been withdrawn:

1.10 "Mechanical (Cadmium) Splices in Reinforcing Bars of Category I Concrete Structures"

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1.15 "Testing of Reinforcing Bars for Category I Concrete Structures"

1.18 "Structural Acceptance Test for Concrete Primary Reactor Containments"

1.19 "Nondestructive Examination of Primary Containment Liner Welds"

1.55 "Concrete Placement in Category I Structures" and

1.103 "Post-tensioned Prestressing Systems for Concrete Reactor Vessels and Containments."

Consider deleting RGs 1.10, 1.15, 1.18, 1.19, 1.55 and 1.103 from SRP Section 3.8.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

10030/RG 1.136

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**Integrated Impact Number: 543      SRP Section Number: 3.8.1**

#### **Suggested Changes to the SRP Section:**

Revise SRP Section 3.8.1 to conform with SECY-93-087, Item I.M.

Item I.M. of SECY-93-087 states that the Commission approves the staff's recommendation to eliminate the OBE from the design of systems, structures and components. The OBE is discussed in several places in SRP Section 3.8.1, e.g., discussion of loads for severe environmental conditions.

The SECY further states that with the elimination of the OBE the equipment should be qualified in accordance with the provisions of the IEEE Standard 344-1987, with either (1) five one-half SSE events followed by one full SSE event, or (2) a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events, in accordance with Appendix D of IEEE Standard 344-1987 when followed by one full SSE.

The OBE is to be continued as a threshold criterion for conducting inspections following an earthquake event.

Consider modifying SRP Section 3.8.1 to reflect the current staff position regarding OBE.



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#### **Potential Impacts/Documents supporting the Suggested Changes:**

23047/SECY 93-087; 24150/SECY 93-087; 25681/FINAL SER CE80 CH 1; 25682/FINAL SER ABWR CH 1; 25683/FINAL SER ABWR CH 3; 25684/FINAL SER CE80 CH 3

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**Integrated Impact Number:** 544      **SRP Section Number:** 3.8.1

#### **Suggested Changes to the SRP Section:**

Consider specifying the current version of the ASME B&PV Code, Section III, Division 2 (ACI 359) in SRP 3.8.1.

The SRP does not reference specific issues of industry standards so that the latest RG endorsed standards could be used. RG 1.136, "Materials, Construction, and Testing of Concrete Containments," Rev. 2, June 1981, endorses articles CC-1000, CC-2000, CC-4000, CC-5000, and CC-6000 of the 1980 edition of the ASME B&PV Code. In SRP Section 3.8.1, Acceptance Criteria, the staff endorses article CC-3000 with exceptions. In Section 3.8.1 of the ABWR FSER, the staff states that it has reviewed the adequacy of the 1989 edition of ASME Code, Section III, Division 2, and found it acceptable with an exception for tangential shear design.

The ASME B&PV Code has been revised and the latest revision, as indicated in NUREG/CR 5973, was issued in 1992.

A comparison between the 1980 and 1992 versions of the ASME B&PV Code (ACI 359) should be performed to support endorsement of the current version in SRP 3.8.1. The comparison should reflect the endorsement of article CC-3000 in SRP Section 3.8.1 and the staff review of the 1989 edition of the Code in the ABWR FSER.

An IPD 7.0 form has been initiated.

Consider modifying SRP 3.8.1 to include a discussion of the latest edition of the ASME B&PV Code (1992).

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24153/NUREG CR-5973

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**Integrated Impact Number:** 548      **SRP Section Number:** 10.4.7

#### **Suggested Changes to the SRP Section:**

Incorporate in SRP 10.4.7 a discussion on the modifications to fluid system design and assign the staff recommendations to the proper reviewer.

NUREG-0619 details the staff's efforts in resolving Generic Technical Issue A-10, "BWR Nozzle Cracking," and contains the solutions recommended by the staff with respect to the nozzles, spargers, cladding, leakage, operating procedures, and inservice testing. Implementing these recommendations could have a major effect on the design of the BWR fluid systems as well some effect on PWR fluid systems. Generic Letter 81-11 provides clarification to portions of NUREG-0619.

Consider incorporating in Areas of Review the paragraphs describing the evaluations performed by other branches discussions on the nozzles, spargers, cladding, leakage, operating procedures, and inservice testing.

Consider incorporating in Review Procedures, paragraph 2, a discussion on nozzles, spargers, and cladding, and in paragraph 5 a discussion of sparger leakage.

Also in Review Procedures consider incorporating a requirement that the reviewer will review the system design for its ease of inservice testing with respect to the nozzles and spargers.

Consider incorporating in Evaluation Findings a discussion verifying the acceptability of the nozzle and sparger design, leakage control design and the inservice testing procedures.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1711/NRC GENERIC LETTER 81-11; 22465/NRC GENERIC LETTER 80-95

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**Integrated Impact Number:** 549      **SRP Section Number:** 10.4.7

#### **Suggested Changes to the SRP Section:**

Revise review procedures to address the need for a program to monitor and control erosion/corrosion induced wall thinning in the condensate and feedwater systems.

Generic Letter 89-08 discusses instances of significant degradation of piping and components of high energy carbon steel piping systems (two phase as well as single phase) due to erosion/corrosion. The staff reviewed implementation of erosion/corrosion monitoring

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programs, including a review of whether adequate guidance was provided for corrective actions and other activities regarding repair and replacement of degraded piping and components. The program should include formalized procedures or administrative controls to ensure continued long-term implementation of the erosion/corrosion monitoring program.

Consider revising review procedures to incorporate the staff positions on erosion/corrosion monitoring as discussed in Generic Letter 89-08.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1706/NRC BULLETIN 87-01; 1720/NRC GENERIC LETTER 89-08

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**Integrated Impact Number: 550      SRP Section Number: 10.4.7**

#### **Suggested Changes to the SRP Section:**

Revise areas of review for review of steam generator and reactor vessel overfill protection control systems.

The current areas of review in SRP Section 10.4.7 states that upon request, the Instrumentation and Control Branch will review the instrumentation and controls associated with the feedwater control system or steam generator level control system. This review interface needs to be strengthened to reflect the review of overfill protection control systems related to the resolution of USI A-47, "Safety Implications of Control Systems," as discussed in Generic Letter 89-19. This review will be addressed in SRP Section 7.7.

Consider revising areas of review to reflect the review of overfill protection systems in SRP Chapter 7.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1723/NRC GENERIC LETTER 89-19

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**Integrated Impact Number: 551      SRP Section Number: 10.4.4**

#### **Suggested Changes to the SRP Section:**

Update Subsection VI, References, to include the revised title for GDC 4.

Rulemaking in 1987 revised GDC 4, including the title. GDC 4 as it relates to pipe breaks or malfunctions in the turbine bypass system and the resultant affect on essential systems is

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addressed in II.1 of Acceptance Criteria, II.3.b of Review Procedures, and IV.1 of Evaluation Findings. However, in Subsection VI, References, the title of GDC 4 is "Environmental and Missile Design Bases."

Consider changing Subsection VI, References, to read GDC 4, "Environmental and Dynamic Effects Design Bases."

#### **Potential Impacts/Documents supporting the Suggested Changes:**

6301/CODE OF FED. REGS 10CFR50; 21778/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 552      **SRP Section Number:** 9.5.6

#### **Suggested Changes to the SRP Section:**

Revise the review procedures to address the system recharging capacity of the diesel engine air starting system compressor.

In their reviews of the ABWR and CE System 80+ designs as documented in the respective FSERs, the staff noted that starting air system interfaces with the diesel engines, such as cranking devices, duration of the cranking cycle and revolutions per start attempt would dictate design parameters including volume and design pressure of the air receivers (sufficient for five start cycles per receiver) and compressor size (sufficient discharge flow to recharge the system in under 30 minutes). Specific criteria associated with GDC 17 is included in the Acceptance Criteria subsection of SRP Section 9.5.6 for five start cycles per receiver. There currently is no specific criteria in the SRP related to compressor sizing.

Consider revising the review procedures instead of the specific criteria associated with GDC 17, to include a 30 minute recharging criteria for the air start system compressor.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22380/FINAL SER ABWR CH 9; 22394/FINAL SER CE80 CH 9

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**Integrated Impact Number:** 553      **SRP Section Number:** 3.2.1

#### **Suggested Changes to the SRP Section:**

Add Regulatory Guide 1.151 as a guidance document and develop an associated Review Procedure to address seismic classification of safety-related instrumentation sensing lines.

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Regulatory Guide 1.151 describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to the seismic classification of safety-related instrument sensing lines. Regulatory Guide 1.151 positions C.2 and C.3 provide detailed guidance for the proper seismic classification of instrument sensing lines connected to safety-related systems.

Consideration should be given to adding Regulatory Guide 1.151 as a guidance document and developing an appropriate Review Procedure to address the seismic classification of safety-related instrument sensing lines.

This integrated impact is referenced in IPD 7.0 Future Regulatory Action Needs forms 3.2.1-1 and 3.2.1-2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

10906/RG 1.151

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**Integrated Impact Number: 554      SRP Section Number: 6.5.2**

#### **Suggested Changes to the SRP Section:**

Revise the Review Procedures to incorporate new source term assumptions found acceptable to the staff in the design certification reviews for the CE 80+ and ABWR.

In 1962 the U.S. Atomic Energy Commission published TID-14844, "Calculation of Distance Factors for Power and Test Reactors" which specified a release of fission products from the core to the reactor containment in the event of a postulated accident involving "substantial meltdown of the core." This "source term" is the basis for the NRC's Regulatory Guides 1.3 and 1.4, and has been used to determine compliance with the NRC's reactor site criteria, 10 CFR Part 100, and to evaluate other important plant performance requirements including the containment spray system.

NUREG 1465 (draft) incorporates new information with regard to fission product releases that has developed since TID 14844 was issued. NUREG 1465 (draft) was developed to provide more realistic estimates of the source term release into containment including timing, nuclide types, quantities, and chemical form, given a severe core-melt accident. The staff utilized the NUREG 1465 source term information in their review of the CE System 80+ containment spray system in lieu of Regulatory Guide 1.4.

Consider revising the Review Procedures to include NUREG-1465 (draft), "Accident Source Terms for Light-Water Nuclear Power Plants", June 1992, as an acceptable approach to defining source terms for evaluating the fission product removal capability of the containment spray

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system.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23131/SECY 90-016; 24247/FINAL SER CE80 CH 15; 24248/FINAL SER CE80 CH 1

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**Integrated Impact Number: 555      SRP Section Number: 5.2.2**

#### **Suggested Changes to the SRP Section:**

SRP Section 5.2.2 cites IEEE 279 with no date specified.

The latest version is IEEE 603 1991.

PNL plans to conduct a detailed side-by-side comparison between the cited and latest versions of this standard. Pending completion and review of the side-by-side comparison, consider citing the latest version of this standard.

INSPECTION PROGRAM BRANCH (PIPB) COMMENT: Compliance with IEEE 603-1980 as supplemented by Regulatory Guide 1.153 is considered by the NRC staff to satisfy the provisions of IEEE 279-1971. The NRC staff is in the process of revising Regulatory Guide 1.153 to update its endorsement from IEEE 603-1980 to IEEE 603-1991. No further action is necessary pending revision of Regulatory Guide 1.153.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1250/RG 1.153; 23987/C&S: IEEE 279

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**Integrated Impact Number: 556      SRP Section Number: 4.2**

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to incorporate revisions to 10 CFR 50.46 and Appendix K to Part 50 that allow use of best-estimate calculational techniques.

The review in SRP 4.2 focuses on the thermal, mechanical, and materials design of the fuel system and currently cites 50.46 and Appendix K. On September 16, 1988, the NRC staff amended the requirements of 10 CFR 50.46 and Appendix K "ECCS Evaluation Models" to allow the use of either Appendix K assumptions or realistic (best estimate) evaluation models. These realistic models must include sufficient supporting justification to demonstrate that the analytic techniques employed realistically describe the behavior of the reactor system during a

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postulated LOCA. Paragraph 50.46(a)(1) also requires that the uncertainty in the realistic evaluation model be quantified and considered when comparing results of the best estimate calculations with the applicable limits in paragraph 50.46(b). The NRC staff issued Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance" to support the amendments. This regulatory guide describes models, correlations, data, model evaluation procedures, and methods that are acceptable to the staff for meeting the requirements for a realistic or best-estimate analysis of ECCS performance during a LOCA and for estimating the uncertainty in that calculation.

Consideration should be given to adding Regulatory Guide 1.157 as guidance and revising Acceptance Criteria to incorporate provisions for use of realistic (best-estimate) calculational techniques.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

9748/CODE OF FED. REGS 10CFR50; 9777/RG 1.157; 20702/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 557      SRP Section Number: 4.2**

#### **Suggested Changes to the SRP Section:**

ASTM C776-76, Part 45, 1977 is referenced as guidance in Acceptance Criteria related to hydrogen content of fuel pellets. The current version of this standard is ASTM C776, 1989.

Consideration should be given to performing a detailed side-by-side comparison between ASTM C776-76, 1977 and ASTM C776, 1989 to allow SRP reviewers to use the more current version of the standard.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

No comparison needed. C776 was not endorsed as providing acceptance criteria in SRP 4.2. C776 provided information only.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23417/C&S: ASTM C776-77

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**Integrated Impact Number: 558      SRP Section Number: 4.2**

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to incorporate fuel burnup limits and justification based on material properties versus exposure data.

In the EPRI Evolutionary Plant FSER and in the ABWR FSER, the staff objected to specified minimum fuel burnup requirements greater than NRC-approved fuel burnup levels. The staff considers the burnup limit a safety question and has several fuel operating concerns at burnup levels above those currently approved. These concerns impact normal operation, off-normal transients, and accidents. Fuel burnup limits must be specified and justified based on material properties versus exposure data for each fuel type used.

Consideration should be given to revising Review Procedures to address fuel burnup specification and justification.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23418/FINAL SER ABWR CH 4; 23419/FINAL SER EPRI CH 1; 25542/RG 1.70

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**Integrated Impact Number: 559      SRP Section Number: 10.3**

#### **Suggested Changes to the SRP Section:**

For Evolutionary BWRs (that have eliminated the MSIV leakage control system) revise Review Procedures for seismic design bases, quality, and seismic classification of main steam system piping and add appropriate new Review Interfaces.

In response to an EPRI proposal to eliminate the MSIVLCS as a plant optimization feature, the staff stated, in SECY 93-087, that preventing gross structural failure of the piping and condenser hotwell would provide assurance that leakage from the MSIVs following a design-basis accident would not exceed the 10 CFR Part 100 guidelines. In addition, this would ensure the integrity of the main steam drain lines and bypass piping from the first valve to the main condenser hotwell. On this basis, the staff recommended that the main steam drain and bypass lines and the piping between the turbine stop valve and the turbine inlet be evaluated for seismic concerns in lieu of classification as safety-related or as seismic Category I. These lines should be analyzed using a dynamic seismic analysis to demonstrate structural integrity under SSE loading conditions and should meet all of the quality group and quality assurance guidelines specified in SRP Section 3.2.2, Appendix A. In the SRM for SECY 93-087 the Commission approved the approach and positions for classifying the main steam lines and systems for evolutionary boiling water reactors



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In the EPRI FSER and ABWR FSER, the staff took the position that proposals to eliminate the main steam isolation valve leakage control system (MSIVLCS) in Evolutionary BWRs by taking credit for fission product plate-out and holdup in main steam lines and the condenser would be acceptable, subject to complying with additional requirements. These additional requirements include a suitable seismic dynamic analysis of the main steamline from the seismic interface restraint up to the turbine stop valve to demonstrate its structural integrity, and meeting all pertinent quality assurance requirements of 10 CFR 50 Appendix B.

In the ABWR FSER, the staff described additional requirements for eliminating the MSIVLCS. This FSER stated that to process MSIV leakage through the main condenser a leakage path must be assured either through the MS drain line to the condenser or through the turbine bypass system to the condenser. Whichever of these two paths is chosen, a reliable power source must be available to the appropriate control and isolation valves so that a control operator can establish the flow path assuming a single active failure.

In other SRP Sections, different aspects of the MSIVLCS are reviewed. These sections include criteria that should be addressed whether a formal MSIVLCS is installed or not.

Consideration should be given to revising Review Interfaces and Review Procedures (applicable to evolutionary BWRs) to verify that the additional requirements consistent with the staff positions regarding elimination of the MSIVLCS in SECY 93-087 and the EPRI Evolutionary Plant and ABWR FSERs have been met.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23064/SECY 93-087; 23195/FINAL SER EPRI CH 1; 23665/FINAL SER EPRI CH 1;  
23666/FINAL SER ABWR CH 10; 25118/NUREG 0800; 25119/NUREG 0800;  
25120/NUREG 0800

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**Integrated Impact Number: 560      SRP Section Number: 10.3**

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address the effects of N-1 loop operations on steam supplies to turbine-driven safety system pumps.

Review Procedure III.5.a considers the adequacy of steam supplies to safety system pumps.

The NRC staff studied N-1 loop operation under Generic issue B-59. Generic Letter 86-09

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documents technical resolution of B-59. For PWRs, the staff stated that specific design characteristics of each plant must be reviewed in detail to assure that all safety considerations relative to (N-1) loop operation are evaluated. The Generic Letter provided examples of considerations which are highly plant specific in nature, including the effects of the down loop on safety systems such as steam to turbine-driven safety system pumps.

In the EPRI Evolutionary Plant FSER, the staff noted the EPRI requirement that acceptability of operation with one secured reactor coolant pump is subject to plant-specific evaluation to address the concerns delineated in GL 86-09.

Consider revising Review Procedures to address the effects of N-1 loop operations on steam supplies to turbine-driven safety system pumps as discussed in Generic Letter 86-09.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

3898/NRC GENERIC LETTER 86-09; 23663/FINAL SER EPRI CH 1; 23664/NUREG 0933

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**Integrated Impact Number:** 561      **SRP Section Number:** 10.3

#### **Suggested Changes to the SRP Section:**

Add Acceptance Criteria and Review Procedures to include station blackout requirements and guidance.

10 CFR 50.63, the Station Blackout (SBO) rule, requires that each light-water-cooled nuclear power plant be able to withstand and recover from an SBO of a specified duration. Regulatory Guide 1.155 describes a means acceptable to the NRC staff for implementing the requirements of 10 CFR 50.63. Regulatory Guide 1.155 position C.3 contains staff positions related to systems and components required for decay heat removal.

A review of compliance with station blackout requirements is the subject of a new SRP Section 8.4 (proposed). A review of the main steam supply system conformance with Regulatory Guide 1.155 positions, as applicable, would be coordinated with the review proposed for SRP Section 8.4.

Consideration should be given to adding 10 CFR 50.63 and Regulatory Guide 1.155 as Acceptance Criteria and developing applicable Review Procedures to address station blackout requirements in regard to the main steam supply system.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24240/RG 1.155; 24241/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 562      SRP Section Number: 11.5**

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria, Review Procedures and Evaluation Findings to replace citations of superseded sections in 10 CFR Part 20.

Acceptance Criterion II.A cites part 20.106 of 10 CFR 20 as it relates to radioactivity in effluent to unrestricted areas. On May 21, 1991, the NRC issued a revision to these requirements for protection against ionizing radiation. The revised part (Sections 20.1001-20.2401) became effective for implementation on June 20, 1991 and mandatory on January 1, 1994.

The staff, in the ABB-CE System 80+ ASER, directed its evaluation at determining whether the system meets the requirements of 10 CFR Part 20, Section 20.1302.

Consider revising the Acceptance Criteria, Review Procedures and Evaluation Findings to replace citations of superseded sections in 10 CFR Part 20.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23399/ADVANCE SER CE CH 11; 23405/CODE OF FED. REGS 10CFR20; 23628/CODE OF FED. REGS 10CFR20; 23740/CODE OF FED. REGS 10CFR20

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**Integrated Impact Number: 563      SRP Section Number: 11.5**

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to address the requirements of 10 CFR 50.36a and modify Review Procedures, and Evaluation Findings to address overall review of the offsite dose calculation manual (ODCM).

Review Procedure III.2 evaluates proposed technical specifications to determine that the content and intent of the technical specifications prepared by the applicant are in agreement with requirements developed as the result of the staff's review of process and effluent monitoring instrumentation and sampling systems.

In order to keep releases of radioactive materials to unrestricted areas as low as is reasonably

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achievable, 10 CFR Part 50.36a requires each license to include radiological effluent technical specifications (RETS) on effluent from nuclear power reactors. The NRC staff issued Generic Letter 89-01, which allows procedural detail and programmatic controls of RETS to be removed from the main body of the technical specifications (TS) and placed in the ODCM and the process control program (PCP). An Integrated Impact has been written for SRP 11.4 that recommends that the procedural details of the current TS on solid radioactive wastes, being relocated to the Process Control Program, be evaluated as part of SRP Section 11.4, Solid Waste Management System.

Programmatic controls for radioactive effluents and radiological environmental monitoring are to conform to the regulatory requirements of Appendix I to 10 CFR Part 50. Reg. Guide 4.8 identified ongoing staff activity in developing model technical specifications and for radiological environmental monitoring programs. Generic Letter 89-01 Sup. 1 forwarded NUREG-1301 as guidance to PWR licensees who elected to implement Generic Letter 89-01. NUREG-1302 provides similar guidance for BWRs.

The staff stated, in the ABB-CE System 80+ ASER, that they will review the plant-specific RETS that will be provided in a plant-controlled document as well as setpoints in the plant-specific ODCM on plant-specific basis.

Consider revising Acceptance Criteria to address the requirements of 10 CFR 50.36a and modify Review Procedures, and Evaluation Finding to address overall review of the Offsite Dose Calculation Manual. In addition, consideration should be given to updating Reg. Guide 4.8 to reference model technical specifications (NUREGs 1301 and 1302) for radioactive effluents and for radiological environmental monitoring programs. This will be tracked as future work with IPD-7.0 form number 11.5-2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11601/NRC GENERIC LETTER 89-01; 11625/RG 4.8; 23630/CODE OF FED. REGS 10CFR50; 23631/ADVANCE SER CE CH 11; 23633/CODE OF FED. REGS 10CFR50; 23634/NUREG 1301; 23635/NUREG 1302; 23637/NRC GENERIC LETTER 89-01 Sup 1

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**Integrated Impact Number: 564      SRP Section Number: 4.6**

#### **Suggested Changes to the SRP Section:**

Add 10 CFR Part 50.62(c)(3) to the acceptance criteria and associated review procedures.

As resolution of Generic Issue A-9, 10 CFR Part 50.62 established requirements for the reduction of risk from anticipated transients without scram (ATWS) events for light water

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nuclear reactors. In doing so, these requirements impact the hardware, testing and functional design of the CRDS. For BWRs, the ATWS rule stipulates requirements for an Alternate Rod Injection (ARI) system including redundant scram air header exhaust valves.

Consider adding 10 CFR Part 50.62(c)(3) as an acceptance criterion and developing associated review procedures to verify implementation of ATWS Rule requirements that impact CRDS functional design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1503/NUREG 0933; 22056/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 565      SRP Section Number: 4.6**

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to address staff guidance to improve overall BWR scram discharge system design.

In June of 1980, an event occurred at TVA's Brown's Ferry Unit 3 which called into question the ability of the Control Rod Drive System (CRDS) to fully meet the criteria contained in section 4.6 of the SRP. During shutdown of the Brown's Ferry Unit a manual scram failed to insert a number of the control rods. Subsequent investigations narrowed the cause of the problem to an accumulation of water in the scram discharge volume (SDV) header. The accumulation of water in the SDV was primarily contributed to inadequate design of the supporting control air systems, inadequate venting and slow or no draining of the associated discharge volumes. Followup investigations identified a number of other system design deficiencies requiring corrective actions. Information regarding this event and similar events at other plants (such as the event documented in NRC Bulletin 80-14) were disseminated, and actions for licensees operating BWRs were required in NRC IE Bulletin 80-17 and Supplements 1-5. Also, Generic Letters 80-066 and 80-111 issued information similar to NRC IE Bulletin 80-17 Supplements 1 and 4, respectively. Generic Issue 25 dealt with the control air problems which resulted in the slow filling of the SDV with water, this issue was resolved by staff action as documented in the SER enclosed in Generic Letter 80-107. Generic Issue 41, dealing with the long term program to improve the SDV design, was also resolved as documented in the generic SER disseminated by Generic Letter 80-107. A supplement to the SER and clarification of staff positions were disseminated in Generic Letters 81-09 and 81-18.

In the ABWR design, water displaced from the CRDs during scram is routed directly to the reactor pressure vessel and the SDV has been eliminated. The aspects of this issue related to the SDV are not relevant to the evolutionary ABWR. However, the ABWR design incorporates two

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features to prevent the loss or impairment of the scram function because of a slow loss of control air in the system (Generic Issue 25): (1) a low pressure alarm to alert the operator to trouble in the scram air header and (2) an accumulator charging header low pressure scram to automatically shut down the plant before the accumulator is depleted.

Consideration should be given to developing a Review Procedure to address the staff guidance on acceptable BWR scram discharge system design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20823/NRC BULLETIN 80-17; 20824/NRC BULLETIN 80-17 Sup 1; 20825/NRC BULLETIN 80-17 Sup 2; 20827/NRC BULLETIN 80-17 Sup 3; 20828/NRC BULLETIN 80-17 Sup 4; 20834/NRC GENERIC LETTER 81-09; 20837/NRC GENERIC LETTER 81-18; 23278/NRC GENERIC LETTER 80-107; 23292/NUREG 0933; 23308/FINAL SER ABWR CH 4; 24515/NRC BULLETIN 80-14; 24516/NRC GENERIC LETTER 80-66; 24517/NRC GENERIC LETTER 80-111

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**Integrated Impact Number:** 566      **SRP Section Number:** 9.2.6

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to include station blackout requirements and guidance.

The Station Blackout (SBO) rule (10CFR50.63) requires that each light-water-cooled nuclear power plant be able to withstand and recover from an SBO of a specified duration. Regulatory Guide 1.155 contains staff positions related to systems and components required for decay heat removal during an SBO. Certain positions in C.3.2, C.3.3, and C.3.5 would apply to portions of the condensate storage facility necessary for decay heat removal in response to a station blackout and for the associated instrumentation and controls.

A review of the applicant's analysis demonstrating capability to withstand or cope with an SBO event is the subject of new SRP Section 8.4 developed under the SRP-UDP. A review of the condensate storage facility conformance with Regulatory Guide 1.155 positions, as applicable, would be coordinated with the review proposed for SRP Section 8.4.

Consider citing 10 CFR 50.63 and Regulatory Guide 1.155 requirements and guidance in Acceptance Criteria and developing a Review Procedure that coordinates the condensate storage facility review with the SBO review of SRP Section 8.4, as appropriate.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

271/RG 1.155; 23077/FINAL SER CE80 CH 8; 24350/FINAL SER ABWR CH 9

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**Integrated Impact Number: 568      SRP Section Number: 14.2**

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to cite Reg. Guide 1.68.1 as guidance for review of the initial testing of BWR feedwater and condensate systems.

SRP Section 14.2 currently cites several Reg. Guides which provide guidance used as a basis for determining compliance with the Acceptance Criteria and/or provide more detailed information pertaining to the tests called for in Reg. Guide 1.68.

Tests for boiling water reactor (BWR) power conversion systems are described in Regulatory Guide 1.68 to provide assurance that these systems will perform as designed and to aid in minimizing the probability of system malfunctions during subsequent plant operations. Reg. Guide 1.68.1 describes in more detail the type and nature of BWR feedwater and condensate system tests that are acceptable to the staff.

In the ABWR FSER, the staff applied regulatory positions of Reg. Guide 1.68.1 to the review of proposed initial testing of the ABWR feedwater control system.

Consideration should be given to revising Acceptance Criteria to cite Reg. Guide 1.68.1 as guidance for review of the initial testing of BWR feedwater and condensate systems.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4283/RG 1.68.1; 22775/FINAL SER ABWR CH 14

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**Integrated Impact Number: 569      SRP Section Number: 14.2**

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to cite Reg. Guide 1.68.2 as guidance for review of the initial startup testing of remote shutdown capabilities.

SRP Section 14.2 currently cites several Reg. Guides which provide guidance used as a basis for determining compliance with the Acceptance Criteria and/or provide more detailed information pertaining to the tests called for in Reg. Guide 1.68.

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Regulatory Guide 1.68 lists, as one of its initial startup tests, the demonstration of "shutdown from outside the control room." Reg. Guide 1.68.2 provides amplification of guidance for startup test program demonstration of remote shutdown capabilities. Reg. Guide 1.68.2 describes an initial startup test program acceptable to the NRC staff for demonstrating hot standby capability and the potential for cold shutdown from outside the control room.

In the CE System 80+ FSER, the staff applied Reg. Guide 1.68.2 to the review of proposed startup testing of the CE System 80+ remote shutdown capabilities.

Consideration should be given to revising Acceptance Criteria to cite Reg. Guide 1.68.2 as guidance for review of the initial startup testing of remote shutdown capabilities.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4953/RG 1.68.2; 22734/FINAL SER CE80 CH 14; 22737/FINAL SER CE80 CH 14

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**Integrated Impact Number:** 570      **SRP Section Number:** 14.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to cite Reg. Guide 1.68.3 in place of Reg. Guide 1.80 as guidance for review of the preoperational testing of instrument and control air systems and other compressed gas systems.

SRP Section 14.2 currently cites several Reg. Guides, including Reg. Guide 1.80, which provide guidance used as a basis for determining compliance with the Acceptance Criteria and/or provide more detailed information pertaining to the tests called for in Reg. Guide 1.68.

As indicated in Reg. Guide 1.68.3, Reg. Guide 1.68.3 replaces Reg. Guide 1.80 and describes methods acceptable to the NRC staff for complying with the Commission's regulations with respect to preoperational testing verification that instrument and control air systems and the loads they supply will operate properly. The guide also applies to compressed gas systems that supply loads that could affect the overall safety and performance of the plant, compressed gas systems that supply specific system loads that are important to safety, and systems important to safety that use compressed gases other than air.

In the ABWR and CE System 80+ FSERs, the staff applied the regulatory positions of Reg. Guide 1.68.3 to reviews of proposed initial testing of air and compressed gas systems.

Consideration should be given to revising Acceptance Criteria to cite Reg. Guide 1.68.3 as guidance for review of the preoperational testing of instrument and control air systems and other



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compressed gas systems.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4168/RG 1.68.3; 22727/FINAL SER ABWR CH 14; 22734/FINAL SER CE80 CH 14;  
22738/FINAL SER CE80 CH 14; 22776/FINAL SER ABWR CH 14

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**Integrated Impact Number:** 571      **SRP Section Number:** 14.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to cite Reg. Guide 1.136 in place of Reg. Guide 1.18 as guidance for review of the initial testing of concrete containments.

SRP Section 14.2 currently cites several Reg. Guides, including Reg. Guide 1.18, which provide guidance used as a basis for determining compliance with the Acceptance Criteria and/or provide more detailed information pertaining to the tests called for in Reg. Guide 1.68.

Reg. Guide 1.136 describes bases acceptable to the NRC staff for implementing the requirements of the Commission's regulations with regard to the testing of concrete containments. As stated in Reg. Guide 1.136, Reg. Guide 1.18 has been withdrawn. The regulatory position of Reg. Guide 1.18 is considered to be covered by national standard ACI 359 (ASME Section III, Division 2), "Code for Concrete Reactor Vessels and Containments," which is endorsed by Reg. Guide 1.136.

Consideration should be given to revising Acceptance Criteria to cite Reg. Guide 1.136 in place of Reg. Guide 1.18 as guidance for review of the initial testing of concrete containments.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20367/RG 1.136

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**Integrated Impact Number:** 572      **SRP Section Number:** 14.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to cite Reg. Guide 1.9 as guidance for review of the initial testing of diesel-generator units.

SRP Section 14.2 currently cites several Reg. Guides, including Reg. Guide 1.108, which provide guidance used as a basis for determining compliance with the Acceptance Criteria and/or provide more detailed information pertaining to the tests called for in Reg. Guide 1.68.

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Reg. Guide 1.68 describes preoperational tests of emergency or standby AC power supplies and loss of offsite power initial tests. Reg. Guide 1.68 references Reg. Guide 1.9 as guidance for applicable portions of these tests. Reg. Guide 1.9 provides guidance acceptable to the NRC staff for complying with the Commission's requirements that diesel-generator units intended for use as onsite emergency power sources in nuclear power plants be qualified and have the necessary reliability and availability. Diesel-generator units are unique in that explicit regulatory guidance for initial reliability testing exists. Portions of the qualification testing of diesel-generator units are also addressed under initial testing programs. Reg. Guide 1.9 thus provides guidance directly applicable to the initial testing program for diesel-generator units. Revision 3 of Reg. Guide 1.9 integrates guidance previously addressed in Revision 2 of Reg. Guide 1.9, Revision 1 of Reg. Guide 1.108, and pertinent staff generic communications.

In the EPRI Evolutionary Plant FSER, the EPRI-proposed resolution of Issue B-56, "Diesel Reliability" included discussion of the NRC upgrade of Reg. Guide 1.9. In the CE System 80+ FSER, the staff verified that the applicant's listing of Reg. Guides applicable to the initial test program included Reg. Guide 1.9.

Consideration should be given to revising Acceptance Criteria to cite Reg. Guide 1.9 as guidance for review of the initial testing of diesel-generator units.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22734/FINAL SER CE80 CH 14; 22758/RG 1.9; 23674/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 573      **SRP Section Number:** 14.2

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of initial test program compliance with TMI Item I.G.1, "Special Low Power Testing and Training."

SRP Section 14.2 currently cites TMI Item I.G.1 as specific criteria for review of the inclusion of operator training during the performance of certain initial tests. TMI Item I.G.1 requires a low-power testing program, consisting of tests beyond those called for in Reg. Guide 1.68, for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training.

NRC Generic Letter 83-24 provides discussion of the staff's acceptance of the BWR Owner's Group proposed additional testing to meet TMI Item I.G.1. NRC Generic Letter 83-24 also briefly discusses PWR testing to meet TMI Item I.G.1.

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In the ABWR FSER, the staff verified that the SSAR, including applicable test abstracts, addresses the testing outlined in Appendix E of a letter dated February 4, 1981 from D.B. Waters to D.G. Eisenhut. This letter documents the BWR Owner's Group response to TMI Item I.G.1.

Westinghouse letter NS-EPR-2465 describes a special low power test program consisting of natural circulation tests, associated operator training, and procedure validation which the staff considers acceptable to meet TMI Item I.G.1 requirements for all PWR applicants.

Consideration should be given to developing Review Procedures for review of initial test program compliance with TMI Item I.G.1, based upon information from the documents discussed above.

Add the following paragraph, under subsection 12, "Individual Test Descriptions/Abstracts", page 14.2-9:

"g. Will be used in special low power testing program to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program as required for the resolution of TMI Action Item I.G.1. (References 29 and 30)."

Reference No. 30 should be deleted and replaced with the staff's letter that actually approved the Westinghouse proposal to address TMI Action Item I.G.1 - NRC Letter to the Carolina Power & Light Company, "Shearon Harris Nuclear Power Plant Units 1, 2, 3 and 4 Special Low Power Test Program - TMI Action Plan Item I.G.1," January 21, 1982.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4575/NRC GENERIC LETTER 83-24; 22747/NUREG 0694; 22773/FINAL SER ABWR CH 14; 25572/NRC LETTER From Ziemann, DL to Miraglia, FJ

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**Integrated Impact Number:** 577      **SRP Section Number:** 15.6.5

#### **Suggested Changes to the SRP Section:**

Add appropriate Review Procedures, applicable to PWRs, to address boron dilution during LOCA.

The staff, in the ABB-CE System 80+ ASER, stated that experimental evidence and analysis results show that an inherent mechanism for boron dilution in the PWR reactor coolant pump (RCP) loop seals could exist. During a small break loss of coolant accident, deborated water in

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the RCP loop seals could be transported to the core through natural circulation or startup of RCPs. The deborated water could add significant reactivity that, in turn, could result in damage to the core. ABB-CE was requested to address the applicability of this boron dilution event to the System 80+ design and provide resolutions to this issue.

Consider adding appropriate Review Procedures, applicable to PWRs, to address boron dilution during LOCA.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23534/ADVANCE SER CE CH 15

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**Integrated Impact Number: 578      SRP Section Number: 15.6.5**

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to accommodate realistic or best-estimate evaluation models.

Acceptance Criterion II.a. cites Part 50.46 and Appendix K of 10 CFR 50 as it relates to ECCS equipment functions. Review Procedure III.1 verifies that the LOCA calculations were performed using an approved evaluation model. 10 CFR 50.46 and Appendix K were revised on September 16, 1988 to permit the use of acceptable realistic or best-estimate evaluation models in lieu of Appendix K. Reg. Guide 1.157 was issued to describe models, correlations, data, model evaluation procedures, and methods that are acceptable to the NRC staff for meeting the requirements for a realistic or best-estimate calculation of ECCS performance and for estimating the uncertainty in that calculation.

Consider revising Acceptance Criteria and Review Procedures to accommodate realistic or best-estimate evaluation models as allowed by 10 CFR 50.46 and as described in Reg. Guide 1.157. Further analysis of the above documents and the supporting references will be required to ensure the Review Procedures are consistent and compatible with 10 CFR 50.46 and Reg. Guide 1.157.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

8341/RG 1.157; 8356/CODE OF FED. REGS 10CFR50; 18205/NUREG 0933;  
18211/NUREG 0933; 18225/NUREG 0933; 19328/CODE OF FED. REGS 10CFR50

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## **APPENDIX I**

### **Integrated Impacts**

**Integrated Impact Number:** 579      **SRP Section Number:** 15.6.5

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address staff positions related to evaluations of reactor coolant pump (RCP) operation during loss of coolant accidents (LOCAs).

Review Procedure III.4.f requires the reviewer to verify that the applicant's analysis conservatively addresses the operation of the RCPs.

Generic Letters 83-10A through 83-10F document the results of a comparative review of vendor and NRC analyses to determine whether RCP trip is necessary during LOCAs. The NRC concluded that there is a wide range of transients and LOCAs where it is beneficial for the operators to maintain forced circulation cooling and mixing through operation of the RCPs. However, for certain small break LOCAs, continued operation of the RCPs or delayed RCP trips could lead to core damage. Generic Letter 83-10A through 83-10F encouraged PWR owners' group studies of this concern.

Generic Letters 85-12, 86-05 and 86-06 provide guidance concerning implementation of the reactor coolant pump trip criteria safety evaluation for each of the three PWR owners' group submittals. The NRC concluded that the need for RCP trip following a transient or accident should be determined by each licensee on a case-by-case basis to ensure that whatever decision is made regarding pump operation, it will result in safe, reliable operation of reactors and will not adversely affect the ability of the plant to comply with rules and regulations.

Consider revising Review Procedures to address staff positions related to RCP operation during a LOCA as described in safety evaluations promulgated in Generic Letters 85-12, 86-05 and 86-06. Further analysis of these safety evaluations and supporting references will be required to ensure the Review Procedures are consistent and compatible with Part 50.46 and Appendix K of 10 CFR 50.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

8326/NRC GENERIC LETTER 83-10A; 8327/NRC GENERIC LETTER 83-10B; 8328/NRC GENERIC LETTER 83-10C; 8329/NRC GENERIC LETTER 83-10D; 8330/NRC GENERIC LETTER 83-10E; 8331/NRC GENERIC LETTER 83-10F; 8333/NRC GENERIC LETTER 85-12; 8334/NRC GENERIC LETTER 86-05; 8335/NRC GENERIC LETTER 86-06

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### **Integrated Impacts**

**Integrated Impact Number: 581      SRP Section Number: 3.7.3**

#### **Suggested Changes to the SRP Section:**

Revise existing discussion in the Acceptance Criteria subsection to add additional techniques for considering independent support movements.

SRP 3.7.3 delineates acceptable means of accounting for independent support movements in equipment and components. The staff noted in both the CE80+ FSER and EPRI Evolutionary Plant FSER that the applicants had accounted for these movements using independent support motion response spectrum analysis techniques. While not covered in the SRP, the staff found this approach acceptable as long as the "approved techniques" were in accordance with NUREG-1061, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," Volume 4, "Evaluation of Other Loads and Load Combinations," Section 2, "Staff Recommendations on Response Combinations."

Consider including guidance from NUREG-1061, Volume 4, Section 2 on Independent Support Motion Methods in the existing discussion in the Acceptance Criteria subsection for "Multiply-Supported Equipment and Components with Distinct Inputs."

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22099/FINAL SER CE80 CH 3; 22100/FINAL SER EPRI CH 1; 22105/NUREG 1061 VOL 4

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**Integrated Impact Number: 583      SRP Section Number: 3.7.3**

#### **Suggested Changes to the SRP Section:**

Revise existing discussion in the Acceptance Criteria subsection regarding analysis of non-Category I systems' effects on Category I systems.

For seismic interaction between Category I systems and other systems, RG 1.29 Criterion C.3 requires Category I design requirements extend to the first seismic restraint. In addition, the RG's Criterion C.2 requires SSCs that could affect Category I systems to be designed and constructed so that the SSE would not cause a failure in the Category I system. The Acceptance Criteria associated with seismic interaction in SRP 3.7.3 subsection 8 are consistent with RG 1.29.

In the CE80+ DSER and the ABWR FSER the staff found analyses that were not constrained to the first anchor/seismic restraint to be consistent with the intent of RG 1.29 under certain conditions. Specifically, in the CE80+ DSER "restraining as required" was an acceptable

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alternative to anchoring (defined by the staff as restrained in six degrees of freedom) if the non-seismic portion of the piping system was restrained in such a manner so as not to invalidate the analysis for the seismic portion of the system. In the ABWR FSER the staff determined that an analysis could go to the first anchor or a sufficient distance in the non-seismic system so as not to degrade the validity of the Category I analysis.

Consider allowing analyses similar to the approaches taken in the ABWR FSER and the CE80+ DSER. Consideration should also be given to adding RG 1.29 as a reference to the Acceptance Criteria for "Interaction of Other Systems With Category I Systems."

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11023/RG 1.29; 22096/FINAL SER ABWR CH 3; 22097/DRAFT SER CE80 CH 3; 25563/FINAL SER CE80 CH 3; 25564/FINAL SER CE80 CH 3; 25567/FINAL SER CE80 CH 3; 25568/FINAL SER CE80 CH 3

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**Integrated Impact Number: 584      SRP Section Number: 9.5.1**

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures and Branch Technical Position (BTP) CMEB 9.5-1 to incorporate additional staff positions from Generic Letters (GL) 81-12, 83-33, 86-10 and Supplement 1 to GL 86-10.

The current SRP incorporates, as Acceptance Criteria, the Fire Protection Rule, 10 CFR 50.48, which became effective in February 1981. After 10 CFR 50.48 and Appendix R to 10 CFR Part 50 were published, the staff revised BTP APCSB 9.5-1, which became BTP CMEB 9.5-1, to include the provisions of Appendix R.

10 CFR 50.48(b) requires plants licensed to operate prior to January 1, 1979 to comply with applicable requirements of Appendix R to 10 CFR 50. To assist the staff in their assessment of compliance with these requirements, they issued GL 81-12 which included, as enclosure 1, a "Staff Position Safe Shutdown Capability." Further analysis is required to determine what changes, if any, to BTP CMEB 9.5-1 are appropriate to incorporate the guidance of GL 81-12.

During staff reviews of Appendix R exemption requests and applications for operating licenses, it became apparent that certain requirements of Appendix R to 10 CFR 50 and the corresponding guidelines in SRP 9.5-1 were not being interpreted correctly by some licensees. In response to these differences in interpretation, the staff documented their positions in GL 83-33. The staff issued additional guidance on implementation of Appendix R in GL 86-10, stating that, to the extent that this guidance may be inconsistent with prior guidance (including Generic Letter

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83-33), it is intended that GL 86-10 take precedence. Since GL 86-10 was issued, a special NRC review team was established to evaluate technical issues related to qualification testing of the Thermo-Lag 330-1 fire barrier system. A number of NRC concerns related to these fire barriers have been identified to the industry (eg. Bulletin 92-01, Supplement 1 to Bulletin 92-01, and GL 92-06). Subsequently, the staff issued Supplement 1 to GL 86-10 to refine and clarify fire barrier testing acceptance criteria.

In the EPRI Evolutionary Plant FSER, the ABWR FSER and the CE80+ FSER, the staff indicated they used GL 81-12 and GL 86-10 to supplement the guidance in BTP CMEB 9.5-1 in their reviews of evolutionary plant fire protection.

Consider incorporating appropriate positions from Generic Letters 81-12, 83-33, 86-10, and Supplement 1 to GL 86-10 in Review Procedures and in BTP CMEB 9.5-1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

3817/NRC GENERIC LETTER 81-12; 3823/NRC GENERIC LETTER 83-33; 3842/NRC GENERIC LETTER 86-10; 21527/NRC BULLETIN 92-01; 21528/NRC BULLETIN 92-01 SUP 1; 22120/NRC GENERIC LETTER 92-08; 23701/NRC GENERIC LETTER 86-10 SUP 1; 23847/FINAL SER CE80 CH 9; 23848/FINAL SER EPRI CH 9; 23849/FINAL SER ABWR CH 9; 25280/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 585      SRP Section Number: 9.5.1**

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to include discussion of the "applicable regulation" related to fire protection requirements for evolutionary plants as outlined in SECY 90-016 and SECY 93-087.

In SECY-90-016, the staff concluded that fire issues that have been raised through operating experience and through the Individual Plant Examination for External Events (IPEEE) Program (GL 88-20) must be resolved for evolutionary ALWRs. To minimize fire as a significant contributor to the likelihood of severe accidents for evolutionary advanced reactors, the staff proposed enhancements to NRC's current fire protection guidance in the form of an "applicable regulation." In its SRM of June 26, 1990, the Commission approved the staff's position regarding review criteria for fire protection design, as discussed in SECY-90-016 and supplemented by the staff's April 27, 1990, response to comments by the Advisory Committee on Reactor Safeguards. This issue was also discussed in SECY 93-087 which provides information on the staff's handling of ACRS concerns regarding shared HVAC systems.

In the EPRI Evolutionary FSER, the ABWR FSER and the CE80+ FSER, the staff applied the



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"applicable regulation" criteria in their reviews.

Consider revising Review Procedures to incorporate the "applicable regulation" for evolutionary plant fire protection as outlined in SECY 90-016 and SECY 93-087.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22982/SECY 90-016; 23847/FINAL SER CE80 CH 9; 23848/FINAL SER EPRI CH 9;  
23849/FINAL SER ABWR CH 9

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**Integrated Impact Number: 586      SRP Section Number: 9.5.1**

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures and/or Branch Technical Position (BTP) CMEB 9.5-1 for review of provisions for biocide treatment of fire protection systems that use raw service water as a source.

NRC Bulletin 81-03 concerned flow blockage of water supplies by asiatic clams and mussels. The bulletin required licensees to review fire water supplies and lines to ensure they were not fouled. Generic Issue 51 was established to address the subject of service water system (SWS) fouling at operating plants primarily by aquatic bivalves. Generic Letter (GL) 89-13 transmitted, as enclosure 1, "Recommended Program to Resolve Generic Issue 51" which included recommendations for biocide treatment of systems that use raw service water as a source, such as fire protection systems. Enclosure 1 also recommended periodic flushing and flow testing of systems, such as fire protection systems, that use raw service water as a supply. Supplement 1 to GL 89-13 provided additional staff positions regarding fire protection system biofouling.

In the ABWR DFSE, the staff requested the applicant to address recommendations from GL 89-13 including biocide treatment before layup of systems such as fire protection which use raw service water as a source.

Consider revising Review Procedures and/or Branch Technical Position (BTP) CMEB 9.5-1 to review provisions for biocide treatment and periodic flushing and flow testing of fire protection systems that use raw service water as a source.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

3667/NRC BULLETIN 81-03; 4158/NRC GENERIC LETTER 89-13; 4159/NRC GENERIC LETTER 89-13 Sup 1; 23851/DRAFT FINAL SER ABWR CH 20

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**Integrated Impact Number: 587      SRP Section Number: 9.5.1**

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures and/or Branch Technical Position (BTP) CMEB 9.5-1 to incorporate staff positions regarding design of water supply systems and separation of seismic and non-seismic portions of the fire protection system.

BTP CMEB 9.5-1, Positions C.1.c and C.6.c include seismic criteria for certain suppression systems.

In the ABWR FSER and the CE80+ FSER, the staff included the following positions: 1) that the sprinkler systems in the reactor building and the wet standpipe systems in the reactor and control buildings must be designed to ANSI B31.1 and analyzed to remain functional following a safe-shutdown earthquake; 2) a portion of the water-supply system, including a tank, a pump, and part of the yard supply main must be designed to these requirements also; and 3) during normal operation, the seismically designed and non-seismically designed systems must be separated by normally closed valves and a check valve, so that a break in the non-seismically analyzed portion of the system cannot impair the operation of the seismically designed portion of the system. Although item 1) is consistent with BTP CMEB 9.5- 1, the remaining positions appear to go beyond the current review criteria of SRP 9.5.1.

Consider revision of BTP CMEB 9.5-1 to incorporate the positions stated above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23852/FINAL SER CE80 CH 9; 23854/FINAL SER ABWR CH 9

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**Integrated Impact Number: 588      SRP Section Number: 9.5.1**

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures and/or Branch Technical Position (BTP) CMEB 9.5-1 to incorporate staff positions regarding use of seismically sensitive relays for fire protection systems.

BTP CMEB 9.5-1, Positions C.1.c and C.6.c include seismic criteria for fire suppression and detection systems.

In the EPRI Evolutionary Plant FSER, the staff discussed resolution of a confirmatory issue regarding seismically sensitive relays. The staff established that seismically sensitive relays will not be used in fire protection systems in which seismic induced failure could lead to

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unacceptable consequences.

Consider revision of BTP CMEB 9.5-1 to incorporate the staff positions regarding use of seismically sensitive relays in fire protection, detection, alarm, and suppression systems.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23855/FINAL SER EPRI CH 9

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**Integrated Impact Number: 589      SRP Section Number: 9.5.1**

#### **Suggested Changes to the SRP Section:**

Revise Branch Technical Position, BTP CMEB 9.5-1, to update the NFPA Code references.

Several NFPA standards referenced in BTP CMEB 9.5-1 attached to SRP Section 9.5-1 have been withdrawn, updated, and/or revised since the last revision of SRP Section (See Table 1). The NFPA Standards provide the basis for many of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Standards which have been withdrawn or revised may constitute changes in requirements from those standards previously accepted or approved for use by the NRC. Consideration should be given to performance of standard comparisons for the purpose of evaluating the differences between the cited and current versions of the code to determine the extent of regulatory impacts, if any.

Consider updating the standards referenced to indicate which revision of the standards should be used in the updated SRP.

Table 1  
Referenced Codes and Recommended Comparisons  
for SRP Section 9.5.1

Standard Version Performance Reference In SRP Section 9.5.1	Current Version	Recommended  of Standard Comparison yes/no
NFPA 4-1977 yes NFPA 6-1974	NFPA-1201-1994  Withdrawn	
DRAFT Revision - April 1996	I - 242	

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No	
NFPA 7-1974	Withdrawn
No	
NFPA 8-1974	Withdrawn
No	
NFPA 10-1975	1994
yes	
NFPA 11-1975	1994
yes	
NFPA 11A-1970	1994
yes	
NFPA 11B-1974	Combined with NFPA 11
yes	
NFPA 12-1973	1993
yes	
NFPA 12A-1973	1992
yes	
NFPA 12B-1973	1990
yes	
NFPA 13-1976	1994
yes	
NFPA 14-1974	1993
yes	
NFPA 15-1973	1990
yes	
NFPA 16-1973	1995
yes	
NFPA 20-1973	1993
yes	
NFPA 24-1973	1992
yes	
NFPA 26-1958	1988
yes	
NFPA 27-1975	NFPA 600-1992
yes	
NFPA 30-1973	1993
yes	
NFPA 50A-Not Specified[a]	1994
yes	
NFPA 51B-1976[b]	1994
yes	

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NFPA 69-1973[b]	1992
yes	
NFPA 70-1975	1993
yes	
NFPA 72D-1975	NFPA 72-1993
yes	
NFPA 72E-1974	NFPA 72-1993
yes	
NFPA 80-1975	1992
yes	
NFPA 90A-Not Specified[a]	1993
yes	
NFPA 92-1972[c]	Withdrawn
no	
NFPA 197-1966	NFPA 1410-1995
yes	
NFPA 204-1968	NFPA 204M-1991
yes	
NFPA 220-1975[b]	1992
yes	
NFPA 251-1975	1990
yes	
NFPA 259-1976[b]	1993
yes	
NFPA 802-1974[b]	1993
yes	
NFPA 1962-Not Specified[a]	1993
yes	

[a] These standards appear in the text of BTP CMEB 9.5-1 to SRP Section 9.5.1, but do not appear in the BTP list of references.

[b] These standards are listed in the BTP CMEB 9.5-1 list of references, but do not appear in the text of the BTP.

[c] This standard appears in the BTP CMEB 9.5-1 text as NFPA 92 and NFPA 92M. The BTP list of references only includes NFPA 92M. NUREG/CR 5973, Revision 1 determined the "M" to be a typographical error and the proper reference should be NFPA 92.

INSPECTION PROGRAM BRANCH COMMENT:

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### **Integrated Impacts**

No comparisons needed. Plant Systems Branch does not recommend comparisons of NFPA standards.

#### **PNL COMMENT:**

The entry for NFPA 4 was modified to identify the latest version as NFPA 1201 1994. The latest versions for NFPA 10, NFPA 16 and NFPA 1410 were updated to 1994, 1995 and 1995 respectively (KLB 6-26-95). No changes to the NFPA citations in the SRP will be made pending further direction from the NRC regarding outdated standards citations.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23857/C&S: NFPA 4-77; 23858/C&S: NFPA 4A-69; 23859/C&S: NFPA 6-74; 23860/C&S: NFPA 7-74; 23861/C&S: NFPA 8-74; 23862/C&S: NFPA 10-75; 23863/C&S: NFPA 11-75; 23864/C&S: NFPA 11A-70; 23865/C&S: NFPA 11B-74; 23866/C&S: NFPA 12-73; 23867/C&S: NFPA 12A-73; 23868/C&S: NFPA 12B-73; 23869/C&S: NFPA 13; 23870/C&S: NFPA 14-74; 23871/C&S: NFPA 15; 23872/C&S: NFPA 16-73; 23873/C&S: NFPA 20-73; 23874/C&S: NFPA 24-73; 23875/C&S: NFPA 26-58; 23876/C&S: NFPA 27-75; 23877/C&S: NFPA 30; 23878/C&S: NFPA 50A; 23879/C&S: NFPA 51B-76; 23880/C&S: NFPA 69-73; 23881/C&S: NFPA 70-75; 23882/C&S: NFPA 72D-75; 23883/C&S: NFPA 72E-74; 23884/C&S: NFPA 80-75; 23885/C&S: NFPA 90A; 23886/C&S: NFPA 92-72; 23887/C&S: NFPA 197-66; 23888/C&S: NFPA 204-68; 23889/C&S: NFPA 220; 23890/C&S: NFPA 251; 23892/C&S: NFPA 259-76; 23893/C&S: NFPA 802-74; 23894/C&S: NFPA 1962; 25278/FINAL SER CE80 CH 9; 25279/FINAL SER ABWR CH 9

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**Integrated Impact Number: 590      SRP Section Number: 9.5.1**

#### **Suggested Changes to the SRP Section:**

ASTM E-119-1976 is cited in Branch Technical Position (BTP) CMEB 9.5-1. ASTM D-3286-1973 is referenced in the BTP.

The current versions of these standards are as follows:

- ASTM E-119-1988, "Standard Test Methods for Fire Tests of Building Construction and Materials"
- ASTM D-3286-1991, "Standard Test Method for Gross Calorific Value of Coal and Coke by the Isoperibol Bomb Calorimeter"

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PNL is currently performing a detailed side-by-side comparison between the cited and current versions of these standards. Consider revising the reference to ASTM E-119-1976 to ASTM E-119-1988 and revising the reference to ASTM D-3286-1973 to ASTM D-3286-1991, pending completion and review of the side-by-side comparisons.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23903/C&S: ASTM E119; 23906/C&S: ASTM D3286-73

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**Integrated Impact Number: 591      SRP Section Number: 9.5.1**

#### **Suggested Changes to the SRP Section:**

ANSI B31.1-1973, and the Factory Mutual System Approval Guide are cited in Branch Technical Position (BTP) CMEB 9.5-1.

The current versions of these standards are as follows:

- ASME B31.1-1992, "Power Piping".
- Factory Mutual System Approval Guide-1990

Consideration should be given to performing a detailed side-by-side comparison to allow SRP reviewers to use the current versions of these standards.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

ANSI B31.1 - No comparison needed. This standard applies to non-safety related applications for new plants.

Factory Mutual - No comparison needed. FM is cited in RG 1.120, RG 5.44, and SRP 9.5.1. In RG 1.120, the citation to FM does not specify a version and is limited to two items. In RG 5.44, the citation to FM is not an explicit endorsement and only provides a general statement. In SRP 9.5.1, the citation to FM does not specify a version and is used only for one item on page 31. Thus, there appears to be minimal benefit from doing a comparison.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23902/C&S: ANSI B.31.1; 23907/C&S: Factory Mutual Approval Guide

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**Integrated Impact Number:** 592      **SRP Section Number:** 9.5.1

#### **Suggested Changes to the SRP Section:**

Revise Branch Technical Position (BTP) CMEB 9.5-1 to address the concerns of Generic Issues 106 and 136.

Generic Issue 106 addressed the onsite storage, piping, and use of highly combustible gases. Generic Issue 136 addressed the onsite storage and use of large quantities of cryogenic combustibles. Acceptance criteria for the onsite storage and use of large liquid propane systems were developed in the staff review of the Dresden license amendment for the onsite use of the Mobile Volume Reduction System and were documented in a letter to D. Farrar, Commonwealth Edison Company, from J. Zwolinski, USNRC, "Technical Specifications Relating to the Use of a Mobile Volume Reduction System at Dresden Station," August 13, 1986. Acceptance criteria for the onsite storage and use of large liquid hydrogen and liquid oxygen systems are documented in EPRI NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations," 1987 Revision, September 1987. In a letter to G. H. Neils, BWR Owners Group II, from J. Richardson, USNRC, "Acceptance for Referencing of Licensing Topical Report Titled 'Guidelines for Permanent BWR Hydrogen Water Chemistry Installations,' 1987 Revision, July 13, 1987, the staff approved the EPRI guidelines and recommended that the guidelines be extended to include hydrogen systems supplying hydrogen to the volume control tank in PWRs and for cooling the main electric generators in PWRs and BWRs. These acceptance criteria are in addition to or more restrictive than those in BTP CMEB 9.5-1 Revision 2.

Consider revising BTP CMEB 9.5-1 to incorporate the acceptance criteria developed for onsite storage and use of large quantities of cryogenic combustibles and for hydrogen systems supplying hydrogen to the volume control tank in PWRs and for cooling the main electric generators in PWRs and BWRs.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24242/NRC LETTER From Zwolinski, J. To Farrar, D.; 24243/NRC LETTER From Richardson, J. To Neils, G.H.; 24244/C&S: EPRI NP-5283-SR-A; 24245/NUREG 0933; 24246/NRC MEMORANDUM 12/14/92, From Beckjord, E. to Gillespie, F.

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**Integrated Impact Number:** 593      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures that address Task Action Plan Items II.E.2.1.



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The SRP currently cites TMI Task Action Plan Item II.E.2.1 in Acceptance Criteria step II.3 and in Review Procedures step III.26(c). NUREG-0737 does not identify II.E.2.1 as an item applicable to operating license holders or applicants. NUREG-0718 indicated that construction permit applicants need not address II.E.2.1. NUREG-0933 indicates that Task Action Plan Item II.E.2.1 was covered under Item II.K.3.17 which was implemented as a part of NUREG-0737.

NUREG-0737 item II.K.3.17 required a detailed report from operating licensees on ECCS outages and any proposed changes to improve the availability of the ECCS equipment, if required. In addition, applicants for an operating license shall establish a plan to meet the requirements of Item II.K.3.17. Generic Letter 83-36 provided further clarification on the implementation of item II.K.3.17. The ABWR DSFER states, "Section 1.9 of the SSAR commits the COL holder to report ECCS outages in an annual summary report to the NRC," indicating that the staff intends to review for compliance with this item during the COL review.

Consideration should be given to replacing Task Action Plan Item II.E.2.1 with Task Action Plan Item II.K.3.17. Acceptance Criteria for II.K.3.17 would be applicable to OL/COL applicants and the Review Procedures should reflect the positions in NUREG-0737 as updated by Generic Letter 83-36.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22312/NUREG 0660; 22313/NUREG 0933; 23177/NRC GENERIC LETTER 83-36;  
23370/NUREG 0718; 25571/FINAL SER ABWR CH 20

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**Integrated Impact Number: 595      SRP Section Number: 6.3**

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures to address NRC staff positions and guidance on safety depressurization systems and automatic depressurization systems (ADS).

In SECY 90-016, the staff concludes that the ALWR designs should include a depressurization system. In the staff requirements document (SRM) for SECY 90-016 the Commission approved the staff's positions that the evolutionary light water reactor designs include a depressurization system. In SECY 93-087 the staff recommends that the Commission approve the general criteria that the evolutionary LWR designs provide a reliable depressurization system. The EPRI FSER contains staff guidance on acceptable design objectives for the depressurization system that are intended to preclude direct containment heating (DCH). The ABWR design incorporates a safety grade automatic depressurization system in the ECCS systems. The CE80+ design also includes a safety grade depressurization system to minimize the possibility of high-pressure molten-core ejection.

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Consideration should be given to developing Acceptance Criteria and Review Procedures to address the staff positions and guidance on the safety depressurization and ADS systems for ALWR plant designs.

Alternatively, consideration should be given to including the above positions into a proposed new SRP section that would address safety depressurization systems. IPD 7.0 form 6.3-1, has been initiated to track the the preparation of a proposed SRP section to address safety depressurization systems. Incorporation of the above positions and guidance into SRP section 6.3 would be unnecessary if this approach is chosen.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22373/SECY 90-016; 22374/FINAL SER EPRI CH 5; 22887/SECY 93-087; 25611/FINAL SER ABWR CH 6

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**Integrated Impact Number: 596      SRP Section Number: 6.3**

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criterion that addresses Task Action Plan Item II.K.3.18.

The current Acceptance Criteria for this Task Action Plan Item states that it involves ECCS outages for all plants (see integrated impact number 593 for issues related to ECCS outages). In Task Action Plan Item II.K.3.18 all BWR plants were required to modify the Automatic Depressurization System (ADS) logic to eliminate the need for manual actuation to assure adequate core cooling. Feasibility and risk studies are also required to determine the best approach.

The rules 10 CFR 50.34(f)(1)(vii) and 10 CFR 52.47(a)(1)(ii) include references for information to some of the TMI Action Items including Item II.K.3.18. The ABWR FSER addresses this issue under 10 CFR 50.34(f)(1)(vii). The ABWR design bypasses the need for a high drywell pressure signal to initiate the ADS system through use of an eight minute timer that bypasses the high drywell pressure permissive.

Consideration should be given to revising the Acceptance Criteria and Review Procedures for Task Action Plan Item II.K.3.18 to address the requirements for the ADS system in BWR plants identified above.

Alternatively, consideration should be given to including the above requirements for the ADS system in BWR plants into a proposed new SRP section that would address safety depressurization systems. IPD 7.0 form 6.3-1, has been initiated to track the preparation of a

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proposed SRP section to address safety depressurization systems. Incorporation of the above requirements into SRP section 6.3 would be unnecessary if this approach is chosen.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

12767/CODE OF FED. REGS 10CFR50; 24825/NUREG 0737; 25054/NUREG 0933;  
25593/FINAL SER ABWR CH 20; 25620/NRC GENERIC LETTER 83-36

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**Integrated Impact Number: 597      SRP Section Number: 6.3**

#### **Suggested Changes to the SRP Section:**

Add Acceptance Criteria and associated Review Procedures to address conformance of the Automatic Depressurization System (ADS) design to the requirements contained in NUREG-0737 Task Action Plan Item II.K.3.28 and 10 CFR 50.34(f)(1)(x).

This requirement is applicable to BWR plants only. A study is required to ensure that the ADS valves, accumulators, and associated equipment and instrumentation will be capable of performing their functions during and following an accident situation and taking no credit for non-safety-related equipment or instrumentation. Additionally air (or nitrogen) leakage through the valves must be accounted for in order to assure that enough inventory of compressed air is available to cycle the ADS valves. The ABWR FSER addresses this issue under 10 CFR 50.34(f)(1)(x). In addition, the ADS system in an ABWR design is part of the ECCS.

Consideration should be given to adding NUREG-0737 Task Action Plant Item II.K.3.28 and 10 CFR 50.34(f)(1)(x) as Acceptance Criteria and developing the associated Areas of Review and Review Procedures to address this requirement.

Alternatively, consideration should be given to including the above requirements for the ADS system in BWR plants into a proposed new SRP section that would address safety depressurization systems. IPD 7.0 form 6.3-1, has been initiated to track the preparation of a proposed SRP section to address safety depressurization systems. Incorporation of the above requirements into SRP section 6.3 would be unnecessary if this approach is chosen.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

12768/CODE OF FED. REGS 10CFR50; 22317/NUREG 0737; 24827/NRC BULLETIN 80-01; 25061/NUREG 0933; 25594/FINAL SER ABWR CH 20

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**Integrated Impact Number:** 599      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

Revise the Review Procedures associated with TMI Task Action Plan Item II.K.3.15 involving isolation of HPCI and RCIC for BWR plants, to incorporate the staff guidance of Generic Letter 83-02.

The HPCI and RCIC systems use differential pressure sensors on elbow taps in the steam lines to their turbine drives to detect and isolate pipe breaks in the systems. The TMI item required that the applicant or licensee modify the pipe-break-detection circuitry so that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent system isolation.

Generic Letter 83-02 provides staff guidance on the minimum and maximum expected response times for this function. The minimum expected response time is a plant specific value. The maximum expected response time should not be higher than seven seconds unless the licensee provides proper justification for selecting a higher response time. Plants that don't have isolation system response time in their Technical Specifications, should include the setpoint and the surveillance requirements on the time delay relay in their Technical Specifications.

Consideration should be given to augmenting the current Review Procedure for TMI Task Action Plan Item II.K.3.15 to address the guidance provided by Generic Letter 83-02.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

12269/NRC GENERIC LETTER 83-02

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**Integrated Impact Number:** 600      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria, Review Procedures, Evaluation Findings and References to address the use of realistic, or best-estimate, evaluation models in accordance with 10 CFR 50.46 and Regulatory Guide 1.157.

At present, SRP section 6.3 Acceptance Criteria step II.G. and Evaluation Findings step IV.(9) specify 10 CFR 50.46 and Appendix K as the sources of requirements for the ECCS being designed so that its cooling performance is in accordance with an acceptable evaluation model. 10 CFR 50.46 and Appendix K were revised on September 16, 1988 to permit the use of an acceptable evaluation model in lieu of Appendix K. Regulatory Guide 1.157 was issued to describe models, correlations, data, model evaluation procedures, and methods acceptable to the

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NRC staff for meeting the requirements for a realistic or best-estimate evaluation models for ECCS performance and for estimating the uncertainty in the calculations.

Consideration should be given to revising the Acceptance Criteria, Evaluation Findings and References to accommodate use of realistic, or best-estimate evaluation models as allowed by 10 CFR 50.46 and as described in Regulatory Guide 1.157. Further analysis of the above documents and the supporting references will be required to ensure the Review Procedures are consistent and compatible with 10 CFR 50.46 and Regulatory Guide 1.157.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

74/RG 1.157; 183/CODE OF FED. REGS 10CFR50; 193/CODE OF FED. REGS 10CFR50; 18204/NUREG 0933; 18209/NUREG 0933; 18221/NUREG 0933

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**Integrated Impact Number:** 601      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to ensure that the ECCS system design has provisions to ensure that thermal stratification and thermal stresses, which may occur in the unisolable portions of piping connected to the RCS, are properly accounted for.

Thermal stratification and the resultant thermal stresses, which can occur when cold water leaks into or hot water leaks out of the RCS, can cause premature failure of the unisolable sections of connected piping subject to the stresses. NRC Bulletin 88-08 provides staff positions and guidance directed at identifying, preventing and correcting this problem. The ABWR FSER requested that applicants review their design in accordance with NRC Bulletin 88-08 to determine if any sections of unisolable piping connected to the RCS could be subject to unacceptable thermal stresses.

Consideration should be given to adding a Review Procedure to address provisions in the ECCS system design to account for potential thermal stratification and resultant thermal stresses.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1284/NRC BULLETIN 88-08; 25573/FINAL SER ABWR CH 3; 25574/FINAL SER CE80 CH 3

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**Integrated Impact Number:** 602      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to address the review of the ECCS systems capability to support inservice inspection and testing guidance and requirements. In addition, add an Area of Review (Review Interface) to address the coordination and review of general inservice inspection and testing requirements.

SECY 90-016 section IV.B, the CE80+ FSER section 3.9.6 and the EPRI FSER chapter 1 section 12.2 contain discussions on staff positions and guidance affecting the inservice testing of safety-related pumps and valves. Some of these guidelines and positions, which supplement the Section XI requirements, will have a direct impact on the ECCS systems. Two examples are full flow testing (capability to test the system at 100 percent of the design flow) for safety related pumps and full flow testing of ECCS check valves. 10 CFR 50.55a(b)(2)(iv) and Bulletin 79-17 provide examples of specific ECCS inservice inspection requirements. Acceptance Criteria are currently provided to ensure that the ECCS design meets the requirements of GDC 36 and 37; adding a Review Procedure will provide reasonable verification that the applicable criteria will be met. The Review Interface is recommended to address the general inservice inspection and testing program requirements contained in 10 CFR 50.55a and as detailed in Section XI of the ASME Code.

Consideration should be given to developing a Review Procedure that will ensure verification that the specific criteria, staff positions and guidance on inservice testing and inspection that have a direct impact on the ECCS systems will be met. Consideration should also be given to adding an Area of Review (Review Interface) which will direct the reviewer to SRP sections 3.9.6 and 6.6 which contain the general inservice testing and inspection program requirements.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

195/CODE OF FED. REGS 10CFR50; 1326/NRC BULLETIN 79-17; 22371/FINAL SER EPRI CH 1; 22970/SECY 90-016; 25575/FINAL SER ABWR CH 5; 25576/FINAL SER ABWR CH 3; 25577/FINAL SER CE80 CH 6; 25578/FINAL SER CE80 CH 3; 25579/STAFF REQ. MEMO 9007160185; 25580/STAFF REQ. MEMO 9308270107; 25581/SECY 93-087

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**Integrated Impact Number:** 603      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to address proper design of the miniflow systems required to ensure

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ECCS pump protection.

NRC Bulletins 88-04, 86-03, 80-18, and 79-24 address various miniflow design concerns. NRC Bulletin 88-04 addressed a concern that safety related centrifugal pumps could interact during parallel pump operation under miniflow conditions and result in dead headed operation of the weaker pump. NRC Bulletin 86-03 addressed a design deficiency that created a single-failure vulnerability in the minimum flow recirculation line of ECCS pumps that could cause a failure of more than one ECCS pump. NRC Bulletin 80-18 addressed corrective actions for maintaining adequate miniflow to the Centrifugal Charging Pumps (CCPs), designed for ECCS purposes, during all conditions including when the CCPs are automatically started, and where the miniflow isolation valves are automatically isolated upon safety injection initiation. NRC Bulletin 79-24 addresses design deficiencies that allowed freezing of an ECCS miniflow recirculation line. The CE-80+ FSER addressed the staff's concerns regarding the adequacy of miniflow systems for safety-related pumps, including those concerns addressed in NRC Bulletin 88-04. Finally, NRC Generic Letter 89-04 contains staff positions and guidance concerning miniflow instrumentation.

Branch Technical Position (BTP) RSB 6-1 position B.4 currently addresses valves and piping between the RWST and the safety injection pumps to ensure no single active failure will result in damage to pumps such that the minimum flow requirements for long-term core and containment cooling after a LOCA are not satisfied. Although this position in the BTP addresses miniflow requirements, it does not adequately address the positions identified above.

Consideration should be given to adding a Review Procedure to address applicable miniflow issues for the safety injection pumps.

Part A was revised to include a discussion covering the staff positions of Generic Letter 89-04 on 9/21/95.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1288/NRC BULLETIN 88-04; 1328/NRC BULLETIN 80-18; 1331/NRC BULLETIN 86-03; 12254/NRC BULLETIN 79-24; 25582/FINAL SER ABWR CH 3; 25583/FINAL SER CE80 CH 3; 25584/FINAL SER CE80 CH 6; 25617/NRC GENERIC LETTER 89-04

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**Integrated Impact Number:** 604      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

Modify SRP section 6.3 to address the concerns of Generic Issue 105, and consider issuance of a new SRP section on interfacing systems LOCA.

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Generic Issue 105 addressed interfacing systems loss-of-coolant accidents. In Memorandum for F. Gillespie from W. Minners, "Proposed Resolution of Generic Issue 105, 'Interfacing Systems LOCA in LWRs,'" April 2, 1993, RES proposed a new SRP Section as part of the resolution of the issue. Some of the language in the proposed section is not fully consistent with the NRC position as expressed in SECY-93-087 and should be modified accordingly. The inconsistencies relate to (1) the need to design, to the extent practicable, all systems and subsystems connected to the reactor coolant system (RCS) to withstand full RCS pressure and (2) the inclusion of piping runs and all associated elements of the systems.

Consider incorporating applicable portions of the resolution of GI 105 into SRP Section 6.3. Consideration should also be given to developing a new SRP Section on interfacing system LOCA as a future work item. This future work recommendation will be tracked with IPD 7.0 Form 6.3-2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

15470/NUREG 0933; 21820/FINAL SER EPRI CH 5; 22969/SECY 90-016; 23313/NRC MEMORANDUM 04/02/93, from W. Minners to F. Gillespie; 23314/NRC MEMORANDUM 06/03/93, from E. Beckjord to J. Taylor; 23315/SECY 93-087; 25586/FINAL SER ABWR CH 20; 25587/FINAL SER CE80 CH 6

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**Integrated Impact Number: 605      SRP Section Number: 6.3**

#### **Suggested Changes to the SRP Section:**

Add Acceptance Criteria and Review Procedures to address the capability of the ECCS systems to provide injection and core cooling following a station blackout event.

10 CFR 50.63, the Station Blackout (SBO) rule, requires that each light-water-cooled nuclear power plant be able to withstand and recover from an SBO of a specified duration. Regulatory Guide 1.155 describes a means acceptable to the NRC staff for implementing the requirements of 10 CFR 50.63. Regulatory Guide 1.155 position C.3 contains staff positions related to systems and components required for decay heat removal and the ability to maintain adequate reactor coolant system inventory.

Consideration should be given to adding 10 CFR 50.63 and Regulatory Guide 1.155 as Acceptance Criteria and developing applicable Review Procedures to address station blackout requirements in regard to the ECCS.



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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22368/RG 1.155; 23316/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 606      SRP Section Number: 6.3**

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to address the recommendations and staff positions concerning ECCS capability to provide reactor coolant system (RCS) inventory additions during reduced inventory operations.

Generic Letter 88-17 detailed a set of actions to be implemented by operating licensees prior to reduced inventory operations. Those recommendations that could impact the ECCS include: (1) provide at least two available or operable means of adding inventory to the RCS that are in addition to pumps that are a part of the normal decay heat removal (DHR) systems, including at least one high pressure injection pump, (2) the water addition rate capable of being provided by each of the means should be at least sufficient to keep the core covered, (3) procedures for the use of these systems during loss of DHR events should be provided, and (4) the path of water addition must be specified to assure the flow does not bypass the reactor vessel before exiting any opening in the RCS.

Consideration should be given to adding a Review Procedure to address the recommendations and staff positions concerning ECCS capability to provide RCS inventory additions during reduced inventory operations.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23148/NRC GENERIC LETTER 88-17; 25610/FINAL SER CE80 CH 5

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**Integrated Impact Number: 607      SRP Section Number: 6.5.1**

#### **Suggested Changes to the SRP Section:**

This Integrated Impact initiated an IPD 7.0 form recommending a revision to Regulatory Guide 1.52 and will not be processed further.

The SRP currently cites ANSI N509 and ANSI N510 in conjunction with Reg. Guide 1.52 with regard to the design of ESF air filtration systems. In revision 2 of Regulatory Guide 1.52, the staff recommends that ductwork be designed, constructed, and tested in accordance with Section 5.10 of ANSI N509-1976. The current version is ASME N509-1989. Regulatory Guide 1.52

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also endorses ANSI N-510-1975 the current version of which is ASME N510-1989.

The current versions of these standards are referenced in the EPRI evolutionary FSER in a summary of key requirements for safety-related HVAC systems. In addition, Article AA-4000, "Structural Design," in the 1988 revision of ANSI/ASME AG-1 provides minimum design requirements for the structural design of HVAC equipment and supports. The Staff has concluded that Article AA-4000 provides minimum design requirements that are acceptable. However, because other portions of ANSI/ASME AG-1-1988, such as rules for the design of HVAC ductwork, remain to be developed, the staff has not fully endorsed this standard.

Consideration should be given to revising Regulatory Guide 1.52 to update the references for ANSI N509-1976, ANSI N510-1975 and to incorporate those portions of ANSI/ASME AG-1 that have been deemed to be acceptable. IPD 7.0 Form No. 6.5.1-1 has been initiated to track the recommended revision to Regulatory Guide 1.52.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4167/RG 1.52; 22668/FINAL SER EPRI CH 9; 22682/DRAFT FINAL SER ABWR CH 3

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**Integrated Impact Number:** 608      **SRP Section Number:** 6.5.1

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to address the impact of normal operation of the ESF atmospheric cleanup systems on the functional capability of the systems during a design basis accident (DBA).

The ABWR ASER indicated the staff requested an analysis to demonstrate that the use of the Standby Gas Treatment System (SGTS) during normal plant operation does not impair its functional capability during a DBA. The staff considers that the subject analysis need not be performed, provided the use of SGTS during power operation is limited to no more than 90 hours per year (approximately 1 percent of the time). However, if 90 hours of operation per year for either train is to be exceeded, the COL applicant is required to provide functional damage analyses to demonstrate that SGTS is capable of performing its intended function in the event of a LOCA.

Consideration should be given to adding a Review Procedure to address the staff positions regarding analysis of the impacts of normal operation on the functional capability of ESF atmosphere cleanup systems during a DBA.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

23306/ADVANCE SER ABWR CH 6

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**Integrated Impact Number: 609      SRP Section Number: 6.5.1**

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to review for the proper construction of charcoal absorber cells to preclude the potential for the loss of charcoal.

NRC Bulletin 80-03 requested specific actions by licensees to determine if charcoal absorber cells in use, or proposed for use, had the potential for loss of charcoal incidental to handling, storage or use. NRC Generic Letter 80-11 transmitted this NRC Bulletin to licensees for action. In the ABWR ASER, the staff requested that the SSAR be revised to address IE Bulletin 80-03 to state that the charcoal tray and screen will be all welded construction to preclude potential loss of charcoal from absorber cells per IE Bulletin 80-03.

Consideration should be given to adding a Review Procedure to address the staff positions contained in IE Bulletin 80-03 in regard to precluding the potential for loss of charcoal from charcoal absorber cells.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23173/NRC GENERIC LETTER 80-11; 23174/NRC BULLETIN 80-03; 23175/ADVANCE SER ABWR CH 6

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**Integrated Impact Number: 610      SRP Section Number: 6.5.1**

#### **Suggested Changes to the SRP Section:**

ANSI N509-1980 is cited in SRP section 6.5.1 in conjunction with Reg. Guide 1.52 as guidance pertaining to the design, construction, and qualification and acceptance testing of the components which make up Engineered Safety Feature (ESF) air-cleaning systems. Reg. Guide 1.52 endorses ANSI N509-1976.

The current version of this standard, ASME N509-1989, is referenced in the EPRI evolutionary FSER in a summary of key requirements for safety-related HVAC systems.

Consideration should be given to performing a detailed side-by-side comparison between the cited and current versions of standard ANSI N509 to allow SRP reviewers to use the more

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current version of the standard. This comparison would also support work to revise Reg. Guide 1.52 as recommended in IPD 7.0 Form No. 6.5.1-1.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

No comparison needed. Per 11/94 conversation with Plant Systems Branch staff, NRC has already endorsed the current version in the Improved Technical Specifications in Section 5.7.2.x.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22685/C&S: ASME N509-89; 22688/FINAL SER EPRI CH 9; 23170/C&S: ANSI N509-80

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**Integrated Impact Number:** 611      **SRP Section Number:** 6.5.1

#### **Suggested Changes to the SRP Section:**

ANSI N510-1980 is cited in SRP section 6.5.1 in conjunction with Reg. Guide 1.52 as guidance pertaining to the testing of the components which make up Engineered Safety Feature (ESF) air-cleaning systems. Reg. Guide 1.52 endorses ANSI N510- 1975. The current version is ASME N510-1989.

The current version of this standard, ASME N510-1989, is referenced in the EPRI evolutionary FSER in a summary of key requirements for safety-related HVAC systems.

Consideration should be given to performing a detailed side-by-side comparison between cited and current versions of standard ANSI N510 to allow SRP reviewers to use the more current version of the standard. This comparison would support work to revise Reg. Guide 1.52 as recommended in IPD 7.0 Form 6.5.1-1.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

No comparison needed. Per 11/94 conversation with Plant Systems Branch staff, NRC has already endorsed the current version in the Improved Technical Specifications in Section 5.7.2.x.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22687/C&S: ASME N510-89; 22688/FINAL SER EPRI CH 9; 23171/C&S: ANSI N510-80

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**Integrated Impact Number:** 612      **SRP Section Number:** 13.1.1

#### **Suggested Changes to the SRP Section:**

Standard ANSI N18.1 is cited, in conjunction with Reg. Guide 1.8, as specific criteria for the qualifications of the applicant's corporate "Engineer in Charge." Reg. Guide 1.8 endorses ANSI N18.1-1971, without exceptions, for the qualification and training of certain personnel. The current version of the ANSI N18.1 standard is ANSI/ANS 3.1-1987. A detailed code comparison has been proposed to evaluate the differences between the cited and current versions of the standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22573/C&S: ANSI N18.1; 22574/C&S: ANS 3.1

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**Integrated Impact Number:** 613      **SRP Section Number:** 12.3 - 12.4

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures as necessary to incorporate changes to 10 CFR Part 20.

SRP 12.3-12.4 Acceptance Criteria cite 10 CFR 20 paragraphs 20.1(c), 20.101, 20.103, 20.104, 20.203, and 20.207. On May 21, 1991, the Nuclear Regulatory Commission issued a revision to its standards for protection against ionizing radiation, 10 CFR Part 20. The revised Part 20 (20.1001 through 20.2401, together with new appendices) was effective for implementation on June 20, 1991. Licensees were required to comply with the new provisions by January 1, 1994.

NUREG-1446 provides comparative text of the old and new Part 20 in a side-by-side format that may be useful in determining the new paragraphs corresponding to those currently referenced in SRP 12.3-12.4. Paragraph numbers in the old and new portions of 10 CFR 20 do not exhibit one-for-one correspondence between the old and new paragraph content.

Consider revising Acceptance Criteria and Review Procedures as necessary to properly reflect the newly applicable paragraphs of 10 CFR Part 20. Further analysis is required to determine what changes are appropriate to SRP 12.3-12.4 Acceptance Criteria and Review Procedures as a result of the significant change to 10 CFR Part 20.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

15871/CODE OF FED. REGS 10CFR20; 23328/NUREG 1446; 23690/CODE OF FED. REGS 10CFR20

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**Integrated Impact Number:** 614      **SRP Section Number:** 12.3 - 12.4

#### **Suggested Changes to the SRP Section:**

ANSI N16.2-1969 and ANSI N101.6-1972 are cited in SRP section 12.3-12.4 as guidance documents in Acceptance Criteria under the Regulatory Guide and other document list at item numbers 19 and 20.

The current version of these standards are as follows:

- ANSI N16.2, "Criticality Accident Alarm Systems" is now numbered ANS 8.3. The 1986 version is current.
- ANSI N101.6, "Concrete Radiation Shields" is now called ANS 6.4, "Guidelines on Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants." The 1985 version is current.

PNL is currently performing a detailed side-by-side comparison between the cited and current versions of these standards. Consider revising the reference to ANSI N16.2-1969 to specify ANS 8.3-1986 and revising the reference to ANSI N101.6-1972 to specify ANS 6.4-1985 pending completion and review of the side-by-side comparisons.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23385/C&S: ANSI N101.6-72; 23386/C&S: ANSI N16.2

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**Integrated Impact Number:** 615      **SRP Section Number:** 12.3 - 12.4

#### **Suggested Changes to the SRP Section:**

Revise existing Acceptance Criteria to incorporate the improved Standard Technical Specifications (STS).

SRP Section 12.3-12.4 cites the STS for four LWR vendors (W, B&W, GE, and CE) in Acceptance Criteria as they relate to radiation protection considerations. Improved STS (NUREGs -1430, -1431, -1432, -1433, and -1434) have been promulgated for industry use.

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Consider citing the improved Standard Technical Specifications in Acceptance Criteria. Further review will be necessary to determine if the improved STS parameters are equivalent to the existing STS parameters in the area of radiation protection design considerations.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23387/NUREG 1430; 23388/NUREG 1431; 23389/NUREG 1432; 23390/NUREG 1433;  
23898/NUREG 1434

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**Integrated Impact Number:** 616      **SRP Section Number:** 5.2.4

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of inspection programs to ensure that boric acid corrosion does not lead to degradation of the reactor coolant pressure boundary.

NRC Generic Letter (GL) 88-05 stated that the NRC believed that boric acid leakage potentially affecting the integrity of the reactor coolant pressure boundary should be procedurally controlled to ensure continued compliance with the licensing basis. GL 88-05 requested that operating licensees provide assurances that a program had been implemented consisting of systematic measures to ensure that boric acid corrosion does not lead to degradation of the assurance that the reactor coolant pressure boundary will have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture.

The request that licensees provide assurances that a program has been implemented to address the corrosive effects of reactor coolant system leakage at less than technical specification limits constituted a new staff position. Previous staff positions had not considered the corrosion of external surfaces of the reactor coolant pressure boundary. Based on the frequency and continuing pattern of significant degradation of the reactor coolant pressure boundary, the staff concluded that, in the absence of such a program, compliance with General Design Criteria 14, 30 and 31 could not be ensured.

Consider developing Review Procedures for review of inspection programs to ensure that boric acid corrosion does not lead to degradation of the reactor coolant pressure boundary.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14391/NRC BULLETIN 82-02; 23555/NRC GENERIC LETTER 88-05

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**Integrated Impact Number:** 617      **SRP Section Number:** 3.8.2

#### **Suggested Changes to the SRP Section:**

Add a discussion of 10 CFR 50.34(f)(3)(v)(B), containment structure loading, to SRP Section 3.8.2.

10 CFR 50.34(f)(3)(v)(B)(1) addresses containment structure loadings produced by an inadvertent full actuation of the post-accident inerting hydrogen control system (assuming carbon dioxide). Under these conditions the stresses in the steel containment should not exceed the limits set forth in the ASME B&P Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level A Limits.

The regulation creates a scenario in which there are effects of full actuation of the post-accident hydrogen control system but not including seismic or design basis accident loadings. Evaluation of instability is not required. It also requires that the containment has the capability to safely withstand pressure tests at 1.10 times the pressure calculated to result from carbon dioxide inerting.

SRP Section 6.2.5, "Combustible Gas Control in Containment," Rev. 2, addresses control of combustible gases in containment following a LOCA, thus not corresponding to the requirements of 10 CFR 50.34(f)(3)(v)(B)(1), as described above .

Review of SRP Section 3.8.2, "Steel Containment," Rev. 1, load combination equations definitions and load combination equations for Level A service discloses that loads resulting from the conditions defined in 10 CFR 50.34(f)(3)(v)(B)(1) and (B)(2) are not included.

Consider modifying the Areas of Review, the Acceptance Criteria, and Evaluation Findings in SRP Section 3.8.2 to incorporate the loadings defined in SRP Section 6.2.5 that will reflect the requirements of 10 CFR 50.34(f)(3)(v)(B)(1) and (B)(2).

#### **Potential Impacts/Documents supporting the Suggested Changes:**

9135/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 618      **SRP Section Number:** 3.8.2

#### **Suggested Changes to the SRP Section:**

Add a discussion of 10 CFR 50.34(f)(3)(v)(A)(1), integrity of steel containments, to SRP Section 3.8.2.

This Integrated Impact concerns the level of stresses in steel containments in case of an increased pressure inside of the structure.

10 CFR 50.34(f)(3)(v)(A)(1) requires that the integrity of steel containments be maintained during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by hydrogen burning or the added pressure from post-accident inerting, assuming carbon dioxide is the inerting agent. Under these conditions the requirements of the B&P Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level C Limits, should be satisfied, considering pressure and dead load alone, except that an evaluation of instability is not required. As a minimum to satisfy the above Code requirements the structure must be designed for a combination of dead load and an internal pressure of 45 psig. The CFR states also that the systems necessary to ensure containment integrity shall be demonstrated to perform their functions under these conditions.

SRP Section 6.2.5 "Combustible Gas Control in Containment" defines the loads generated by hydrogen burning under different conditions but does not specifically address the requirements of the CFR described above. In Paragraph II, "Acceptance Criteria," paragraph "a", it states that "As a result of the TMI-2 accident a reevaluation of the hydrogen that may be generated following an accident is being undertaken."

Review of SRP Section 3.8.2, "Steel Containment," Rev. 1, load combination definitions and load combination equations for Level C service discloses absence of the loads resulting from pressure and dead load alone in conjunction with the loads resulting from hydrogen burning as required by the conditions described above.

Consider modifying SRP Section 3.8.2, the Areas of Review, Acceptance Criteria, Review Procedure, and Evaluation Findings, as appropriate, to reflect the requirements of 10 CFR 50.34(f)(3)(v)(A)(1).

#### **Potential Impacts/Documents supporting the Suggested Changes:**

9136/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 619      **SRP Section Number:** 3.8.2

#### **Suggested Changes to the SRP Section:**

Keep the discussion of GDC 4 but change the reference title.

GDC 4 requires that the structures systems and components important to safety be designed to withstand dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from external events.

SRP Section 3.8.2.II.3 states that the acceptance criteria for the design of steel containments are based, among others, on GDC 4, and include the dynamic effects of equipment failures, missiles, pipe whip, and blowdown loads associated with the loss-of-coolant accident.

SRP Section 3.8.2.IV.3 also states that the staff concludes that the requirements of GDC 4 are met.

However, Rule making in 1987 revised GDC 4, including its title.

10 CFR 50, GDC 4 is entitled "Environmental and dynamic effects design bases." SRP Section 3.8.2.VI.6, refers to GDC 4 as "Environmental and Missile Design Bases."

Consider updating the reference to GDC 4 to be consistent with 10 CFR 50.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21044/CODE OF FED. REGS 10CFR50; 21732/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 620      **SRP Section Number:** 3.8.2

#### **Suggested Changes to the SRP Section:**

Incorporate staff position on shell buckling.

In Section 3.8.1 of the ABWR design certification FSER, the staff discusses its review of the applicant's evaluation of the potential buckling of the drywell head and the buckling criteria used. The staff compared the evaluation to the staff position for shell buckling due to internal pressure described in Appendix E of the FSER.

Consider incorporating Appendix E staff positions on shell buckling in acceptance criteria and review procedures.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24251/FINAL SER ABWR CH 3; 25898/FINAL SER ABWR APPENDIX E

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**Integrated Impact Number: 622      SRP Section Number: 5.2.4**

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for inspection programs to monitor bottom mounted instrumentation thimble tube performance in Westinghouse designed reactors.

NRC Bulletin 88-09 describes degradation of incore flux monitoring system thimble tubes in Westinghouse designed reactors. The bulletin states that the NRC staff believed that lack of requirements for inservice inspection of thimble tubes may have resulted in significant thimble tube degradation having gone undetected, creating a condition that may be adverse to safety. To ensure compliance with General Design Criteria "Reactor Coolant Pressure Boundary" of 10 CFR 50, Appendix A and to minimize (through early detection of thimble tube thinning) the likelihood of a potentially non-isolable leak of reactor coolant, the NRC staff requested that licensees of Westinghouse designed reactors with bottom mounted instrumentation establish an inspection program to monitor thimble tube performance.

Consider developing Review Procedures for inspection programs to monitor bottom mounted instrumentation thimble tube performance in Westinghouse designed reactors.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24086/NRC BULLETIN 88-09

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**Integrated Impact Number: 623      SRP Section Number: 3.9.3**

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address cumulative fatigue damage in ASME Code Class 1 piping.

In Generic Issue 78, "Monitoring of Fatigue Transient Limits for Reactor Coolant System," the staff expressed a concern that repeated thermal cycling of RCS components produces some degree of fatigue degradation of the materials which could lead to failure. In the EPRI Evolutionary Plant FSER, the staff noted the proposed design life of 60 years raises questions relative to the margins available to the current ASME fatigue design curves. In the ABWR ASER and the CE 80 + ASER, the staff noted the cumulative fatigue usage factor should take into consideration all cyclic effects caused by the plant operating transients for a 60-year design

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life. In addition, the staff further noted a comparison of test data with the Code requirements indicates that the margins in the ASME Code fatigue design curves might be less than originally intended. In the ABWR ASER, the staff reviewed and accepted vendor provided supplemental guidelines specific to the ABWR design which enhanced the design margin beyond the requirements of the ASME Code, Section III for fatigue evaluation. The CE 80 + ASER describes the vendor's position that the expected environmental conditions and lack of carbon or low alloy steel directly exposed to the primary coolant allowed use of the existing fatigue curves contained in ASME Code Section III. The staff found this position acceptable.

In both the ABWR ASER and the CE 80 + ASER, the staff noted that it is assessing the potential generic implication of this issue on all operating plants, and depending on the severity of the issue, certain actions might be required to generically address this concern.

Consider modifying the Review Procedures to verify that applicants have addressed the issue of cumulative fatigue damage in ASME Code Class 1 piping throughout the plant's proposed design life.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23801/ADVANCE SER ABWR CH 3; 23806/ADVANCE SER CE CH 3; 23817/ADVANCE SER ABWR CH 3; 23822/FINAL SER EPRI CH 1; 23944/NUREG 0933

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**Integrated Impact Number:** 624      **SRP Section Number:** 3.9.3

#### **Suggested Changes to the SRP Section:**

Add Review Procedures to address stresses caused by thermal stratification, thermal oscillations and thermal striping, as described in NUREG-0619 and by IE Bulletins 88-08 and 88-11.

Generic Letters 80-95 and 81-11 and NUREG-0619 discuss the results of analysis of BWR reactor vessel nozzle cracking caused by cycling of water temperature in systems connected to the vessel. Corrective actions involved re-design and re-analysis of nozzles and thermal sleeves to accommodate expected water temperature variations in the feedwater and control rod drive systems.

In NRC Bulletin 88-08, the staff requested that licensees and applicants review systems connected to the RCS to determine whether any sections of such piping that cannot be isolated can be subjected to temperature oscillations that could be induced by leaking valves. Supplements were issued which broadened the concern to all lines in which stratified flow could occur. NRC Bulletin 88-11 was issued in response to the results of an inspection of the pressurizer surge line at an operating facility that showed large, unexpected movements that

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closed the gaps between the line and pipe whip restraints. The Bulletin required all PWR licensees to establish and implement a program to assure the structural integrity of the surge line when subjected to thermal stratification.

In both ABWR ASER and the CE 80+ ASER, the staff requested that the applicants evaluate the design with respect to thermal stratification, thermal oscillations, and thermal striping to determine if any sections of such piping connected to the RCS could be subject to unacceptable thermal stresses as described in the above bulletins.

Consider adding Review Procedures to verify the applicant has addressed stress caused by thermal stratification, thermal oscillations, and thermal striping, as described in NUREG-0619 and by IE Bulletins 88-08 and 88-11.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

13646/NRC BULLETIN 88-11; 23786/ADVANCE SER CE CH 3; 23808/ADVANCE SER ABWR CH 3; 23828/NRC BULLETIN 88-08; 23831/NRC BULLETIN 88-08 Sup 3; 24617/NRC GENERIC LETTER 80-95; 24618/NRC GENERIC LETTER 81-11; 24619/NUREG 0619

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**Integrated Impact Number:** 626      **SRP Section Number:** 3.9.3

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Appendix A to address the concerns of Generic Issue 113.

Generic Issue 113 addressed the dynamic qualification and testing of large bore hydraulic snubbers. In memorandum for J. Norberg from R. Baer, "Recommendations for SRP Revisions Related to Snubbers," May 5, 1992, RES proposed SRP changes as part of the resolution of the issue.

Consider revising Acceptance Criteria and Appendix A to incorporate RES recommendations on design and testing of snubbers.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

15544/NUREG 0933; 24249/NRC MEMORANDUM 05/05/92, from Robert L. Baer to James A. Norberg

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**Integrated Impact Number:** 627      **SRP Section Number:** 11.5

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures to assure that monitoring systems are adequate to detect contamination of non-radioactive systems and to prevent unmonitored and uncontrolled release of radioactive material to the environment, as discussed in Bulletin 80-10.

Bulletin 80-10 required licensees to establish a routine sampling/analysis or monitoring program to promptly identify systems that are considered as nonradioactive that could possibly become radioactive through interfaces with radioactive systems. In the ABB-CE System 80+ and the ABWR ASERs, the staff determined that the designs were adequate to: (1) detect the contamination of non-radioactive systems, and (2) prevent the potential for unmonitored and uncontrolled release of radioactive material to the environment. Based on these evaluations, the staff concluded that the designs satisfactorily address the concerns raised in IEB 80-10.

Consider developing Review Procedures to assure that monitoring systems are adequate to detect contamination of non-radioactive systems and to prevent unmonitored and uncontrolled release of radioactive material to the environment, as discussed in Bulletin 80-10.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23391/NRC BULLETIN 80-10; 23394/ADVANCE SER ABWR CH 11; 23396/ADVANCE SER ABWR CH 9; 23400/ADVANCE SER CE CH 11; 23401/ADVANCE SER CE CH 20

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**Integrated Impact Number:** 628      **SRP Section Number:** 11.5

#### **Suggested Changes to the SRP Section:**

ANSI N13.10-74 is cited in SRP Section 11.5 in connection with review of applicant's monitoring instrumentation specifications and performance criteria. ANSI N13.10-74 is also cited by Reg. Guide 4.15 which is, in turn, cited as specific criteria in SRP Section 11.5.

ANSI N13.10-1974 was reaffirmed and redesignated as ANSI/IEEE N42.18-1980. In 1991, ANSI/IEEE N42.18-1980 was reaffirmed. PNL is performing a side-by-side comparison between the version of the code cited in the SRP (i.e. ANSI N13.10- 74) and the current version (i.e. ANSI/IEEE N42.18-80)(R91)).

Pending completion and review of the side-by-side comparison, consider replacing citation of ANSI N13.10-74 with ANSI/IEEE N42.18-80(R91). In addition, consideration should be given to updating the citation of ANSI N13.10-74 in Reg. Guide 4.15. This will be tracked as future

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work with IPD- 7.0 form number 11.5-1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11582/RG 4.15; 23402/C&S: ANSI N13.10-74

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**Integrated Impact Number: 629      SRP Section Number: 11.5**

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to add 10 CFR 50.34a and Appendix I of 10 CFR 50.

Part 50.34a addresses the equipment and procedures used to control the release of radioactive material to the environment within the requirements specified in Appendix I.

Appendix I sets numerical objectives for the offsite dose due to effluents and requires monitoring to assess compliance with those objectives. Plants will be required to provide the associated setpoints for the applicable radiation monitors in the plant-specific offsite dose calculation manual (ODCM). (Another Integrated Impact for SRP Section 11.5 proposes establishment of Review Procedures to address overall review of the ODCM.)

The staff, in the ABB-CE System 80+ ASER, reviewed the radiation monitoring system capability to provide early warning of equipment, component, or system malfunction or misoperation, or potential radiological hazards within the station consistent with 10 CFR Part 20 and 10 CFR Part 50, Appendix I.

Consider adding 10 CFR 50.34a and Appendix I of 10 CFR 50 as Acceptance Criteria.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11487/CODE OF FED. REGS 10CFR50; 23631/ADVANCE SER CE CH 11; 23633/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 631      SRP Section Number: 15.6.5.A**

#### **Suggested Changes to the SRP Section:**

Revise Areas of Review (Review Interfaces) and Review Procedures for SRP Section 15.6.5A to indicate that the staff's current approach for reviewing suppression pool decontamination factors is present in SRP Section 6.5.5.

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SRP Section 15.6.5A, Acceptance Criterion 2, cites Regulatory Guide 1.3 as guidance relative to the design basis assumptions for BWRs. Regulatory Position C.1.f of Regulatory Guide 1.3 indicates that no credit should be given for retention of iodine in the suppression pool.

In both the EPRI FSER and the ABWR ASER, the staff found that a credit may be given for the removal of fission products by the suppression pool provided that suppression pool decontamination factors are evaluated in accordance with the methodology prescribed in the revised SRP Section 6.5.5, "Pressure Suppression Pools or Fission Product Cleanup Systems," (issued in December 1988). SRP Section 6.5.5 allows credit to be taken for fission product removal by the suppression pool. In the ABWR ASER, the staff states that suppression pools are capable of scrubbing airborne fission products and that to ignore this capability would be an undue conservatism.

Consideration should be given to revising Areas of Review (Review Interfaces) and Review Procedures for SRP Section 15.6.5A to indicate that the staff's current approach for reviewing suppression pool decontamination factors is present in SRP Section 6.5.5. Consideration should also be given to modifying Regulatory Guide 1.3, Regulatory Position C.1.f as a future work item. This future work recommendation will be tracked with IPD 7.0 form number 15.6.5.A-1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14516/RG 1.3; 22954/ADVANCE SER ABWR CH 15; 23682/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 633      **SRP Section Number:** 15.6.5.B

#### **Suggested Changes to the SRP Section:**

This impact will not be processed further. This issue will be tracked using IPD-7.0 form number 15.6.5B-1.

SRP Section 15.6.5B, Acceptance Criterion 2, relies on Table 1 of Regulatory Guide 1.7 which indicates that 50% of the core iodine inventory should be assumed to be mixed in the sump water being circulated through the containment external piping systems.

According to the CE 80+ ASER, in draft NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," the staff concluded that iodine entering the containment from the reactor core is composed of at least 95 percent cesium iodide (CsI) in particulate form with no more than 5 percent of iodine (I<sub>2</sub>) and hydrogen iodide (HI). Once within the containment, highly soluble cesium iodide will readily dissolve in water pools forming iodide (I<sup>-</sup>) in solution and deposit onto the interior surfaces. The staff also stated in NUREG-1465 that the radiation-induced conversion of iodide (I<sup>-</sup>) in water into elemental iodine (I<sub>2</sub>) is strongly



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dependent on the pH. Without pH control, the staff indicated that large fractions of iodine dissolved in water pools in ionic form will be converted to elemental iodine and will be released into the containment atmosphere if the pH is less than 7. On the other hand, if the pH is maintained at 7 or above, very little (less than 1 percent) of the dissolved iodine will be converted to elemental iodine. The EPRI requirements documents for evolutionary and passive plants and all ALWR designs require that the pH of the water in the containment be maintained at or above 7 (alkaline state) for the entire accident duration to minimize the formation of elemental iodine in the containment water, to reduce subsequent release of iodine into the containment atmosphere.

A review of the information provided in NUREG-1465 is needed to determine if the Regulatory Guide 1.7 assumption of 50% of the core inventory of iodine assumed to be mixed in the sump water is still appropriate. If this assumption needs revision, then Table 1 of Regulatory Guide 1.7 should be considered a candidate for future work.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21373/RG 1.7; 23694/ADVANCE SER CE CH 15; 23695/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 634      **SRP Section Number:** 12.5

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures as necessary to incorporate changes to 10 CFR Part 20.

SRP 12.5 Acceptance Criteria cite numerous paragraphs of 10 CFR Part 20. On May 21, 1991, the Nuclear Regulatory Commission issued a revision to its standards for protection against ionizing radiation, 10 CFR Part 20. The revised Part 20 (20.1001 through 20.2401, together with new appendices) was effective for implementation on June 20, 1991. Licensees were required to comply with the new provisions by January 1, 1994.

NUREG-1446 provides comparative text of the old and new Part 20 in a side-by-side format that may be useful in determining the new paragraphs corresponding to those currently referenced in SRP 12.3-12.4. Paragraph numbers in the old and new portions of 10 CFR 20 do not exhibit one-for-one correspondence between the old and new paragraph content.

Consider revising Acceptance Criteria and Review Procedures as necessary to properly reflect the newly applicable paragraphs of 10 CFR Part 20. Further analysis is required to determine what changes are appropriate to SRP 12.5 Acceptance Criteria and Review Procedures as a result of the significant change to 10 CFR Part 20.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

15905/CODE OF FED. REGS 10CFR20; 23408/NUREG 1446; 23691/CODE OF FED. REGS 10CFR20

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**Integrated Impact Number: 635      SRP Section Number: 12.5**

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to cite new Regulatory Guides.

Standard Review Plan Section 12.5 cites sixteen Division 8 Regulatory Guides in Acceptance Criteria. In the CE 80+ ASER, the staff have listed five newly issued Regulatory Guides as follows:

- 8.25 "Air Sampling in the Workplace,"
- 8.34 "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses,"
- 8.35 "Planned Special Exposures,"
- 8.36 "Radiation Doses to the Embryo/Fetus," and
- 8.38 "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants."

The staff indicated that in any application for a COL, the COL applicant shall state whether it will follow the guidance contained in the above listed regulatory guides.

Consider revising Acceptance Criteria for SRP 12.5 to include citation of these additional Regulatory Guides.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21072/RG 8.25; 22141/RG 8.35; 22142/RG 8.36; 22143/RG 8.34; 23513/RG 8.38; 23520/ADVANCE SER CE CH 12

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**Integrated Impact Number: 636      SRP Section Number: 12.5**

#### **Suggested Changes to the SRP Section:**

This integrated impact will not be processed further. Action will be tracked on IPD 7.0 Form #

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12.5-1.

SRP 12.5 Acceptance Criteria cite Regulatory Guide 8.15, "Acceptable Programs for Respirator Protection," as it relates to elements of acceptable respiratory protection programs. Specific Acceptance criteria under "Equipment, Instrumentation, and Facilities," indicate that acceptance will be based, in part, on a determination that utility-issued personnel protection equipment will include pressure demand full-face-piece air line respirators, and that respiratory protection equipment should meet the requirements of 10 CFR Part 20, Section 20.103.

On May 21, 1991, the Nuclear Regulatory Commission issued a revision to its standards for protection against ionizing radiation, 10 CFR Part 20. The revised Part 20 (20.1001 through 20.2401, together with new appendices) was effective for implementation on June 20, 1991. Licensees were required to comply with the new provisions by January 1, 1994. The revised 10 CFR 20 contains requirements for respirator protection and controls in Subpart H and protection factors for respirators in 10 CFR 20 Appendix A. These rules and associated values differ from those previously used in Regulatory Guide 8.15.

Consider, as future work, revision of Regulatory Guide 8.15 to reflect respiratory protection factors in 10 CFR Part 20.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

16923/NRC BULLETIN 78-07; 23900/CODE OF FED. REGS 10CFR20

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**Integrated Impact Number: 637      SRP Section Number: 12.5**

#### **Suggested Changes to the SRP Section:**

ANSI/ANS 3.1-1978, ANSI N13.7-1972, and ANSI N42.3-1969 are cited in SRP section 12.5 as guidance documents under the Regulatory Guide and other document Acceptance Criteria list at item numbers 23, 27, and 28.

The cited versions of the standards and their current equivalents are as follows:

- ANSI/ANS 3.1-1978 "Selection, Qualification, and Training of Personnel for Nuclear Power Plants." A 1993 version is current.
- ANSI N13.7-1972 "Radiation Protection - Photographic Film Dosimeters - Criteria for Performance." A 1989 affirmation of the 1983 version is current.
- ANSI N42.3-1969 "Test Procedure for Geiger Muller Counters," is now numbered

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ANSI/IEEE 309 - 1970. A 1991 affirmation of the 1970 version is current.

PNL is currently performing a detailed side-by-side comparison between the cited and current versions of ANSI N13.7 and ANSI N42.3. PNL is also performing a side-by-side comparison between the current version and the 1981 version of ANSI/ANS 3.1 which is cited in Regulatory Guides 1.8 (Rev. 2) and 1.149 (Rev. 0).

Consider revising the reference to the above standards to cite current versions pending completion and review of the side-by-side comparisons.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23544/C&S: ANS 3.1; 23545/C&S: ANSI N13.7; 23546/C&S: ANSI N42.3-69

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**Integrated Impact Number:** 638      **SRP Section Number:** 12.1

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria, Review Procedures and Evaluation Findings to replace citations of superseded sections in 10 CFR Part 20.

The Acceptance Criteria subsection cites paragraph 20.1(c) of 10 CFR 20 as it relates to persons involved in licensed activities making every reasonable effort to maintain radiation exposures as low as is reasonably achievable (ALARA).

On May 21, 1991, the NRC issued a revision to the standards for protection against ionizing radiation. The revised Part 20 (Sections 20.1001-20.2401) became effective January 1, 1994. Paragraph 20.1(c) of 10 CFR 20 has been replaced by paragraph 20.1101. Paragraph 20.1101(b) relies on the definition of ALARA in paragraph 20.1003.

Consider revising the Acceptance Criteria, Review Procedures, and Evaluation Findings to replace citations of superseded section 20.01(c) with Sections 20.1101(b) and 20.1003 of 10 CFR 20.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

9369/CODE OF FED. REGS 10CFR20; 9375/CODE OF FED. REGS 10CFR20; 23689/CODE OF FED. REGS 10CFR20

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**Integrated Impact Number:** 639      **SRP Section Number:** 12.1

#### **Suggested Changes to the SRP Section:**

Add paragraph 20.1101 of 10 CFR 20 as Acceptance Criteria for a Radiation Protection Program.

Paragraph 20.1101 of 10 CFR 20 requires each licensee to develop, document, and implement a radiation protection program to ensure compliance with the provisions of Part 20 including use of procedures and engineering controls to achieve occupational doses and doses to the public that are ALARA. The staff, in the ABB-CE System 80+ ASER identified a need for a Radiation Protection Program in accordance with the amended 10 CFR 20. The staff, in the ABWR ASER, acknowledged that Part 20, as amended, contains a number of new programmatic and operational radiation protection concerns that will be addressed by the COL applicant.

Consider adding paragraph 20.1101 of 10 CFR 20 as Acceptance Criteria for a Radiation Protection Program.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23331/ADVANCE SER CE CH 12; 23332/CODE OF FED. REGS 10CFR20;  
23896/ADVANCE SER ABWR CH 12

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**Integrated Impact Number:** 640      **SRP Section Number:** 3.7.3

#### **Suggested Changes to the SRP Section:**

Revise the existing discussion in the Acceptance Criteria subsection regarding determination of the number of earthquake cycles.

At present, SRP Section 3.7.3 Item II.2 and SRP Section 3.9.2 Item II.2.b both establish the number of earthquake cycles during the plant life, to be postulated for analysis purposes, as being based on at least one SSE and five OBEs. The number of cycles per earthquake is obtained from the synthetic time history, or alternately it is set at a minimum of 10 cycles per earthquake, at maximum stress amplitude. However, as discussed in SECY-90-016, the staff will consider on a case by case basis the decoupling of the OBE from the SSE. SECY-93-087 discusses the proposed regulatory positions related to this elimination of the OBE from design consideration. Included in this discussion is a proposed methodology for determining the number of earthquake cycles to be postulated for low cycle fatigue analysis when this analysis is based on the SSE only. The staff proposed two alternatives:

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1) Use an equivalent cyclic load that accounts for differences in the structural damping between OBE and SSE and for a 60-year (instead of a 40-year) plant life. Specifically, to account for earthquake cycles in the fatigue analyses of piping systems performed until new guidance is issued, the staff proposes using two SSE events with 10 maximum-stress cycles per event (20 full cycles of the maximum SSE stress range).

2) Use a number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles (with amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Standard 344-1987.

In SECY-93-087, the staff only identified the relationship to 3.9.2, the corresponding Item in 3.7.3 (which 3.9.2 cites) is not discussed. However, in the ABWR FSER and the CE80 FSER, the staff noted that their review of SSAR Section 3.9.2 was based on the information contained in SSAR Section 3.7.3. For the CE80 FSER, the staff also conducted the review in accordance with the requirements of SRP Section 3.7.3. An Integrated Impact has been proposed for SRP Section 3.9.2 to reference the appropriate Items from SRP Section 3.7.3.

Consider allowing alternate determinations of the number of earthquake cycles, based on the methodology proposed in SECY-93-087, when case by case evaluations allow removal of the OBE from the design basis. Consideration should also be given to adding IEEE Standard 344-1987 as a reference to the Acceptance Criteria for "Determination of Number of Earthquake Cycles."

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22349/FINAL SER ABWR CH 3; 22350/FINAL SER CE80 CH 3; 22351/SECY 93-087;  
22353/SECY 90-016; 22355/C&S: IEEE 344-87

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**Integrated Impact Number:** 643      **SRP Section Number:** 2.4.7

#### **Suggested Changes to the SRP Section:**

SRP Section 2.4.7 cites ANSI N170 1976.

The latest version is ANS 2.8 1992.

Consider performing a detailed side-by-side comparison between the cited and latest version of this standard to allow SRP reviewers to use the latest version.

INSPECTION PROGRAM BRANCH COMMENT:

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No comparison needed. Per 11/94 conversation with Civil Engineering and Geosciences Branch staff, Part 100 is under revision and has been issued for public comment. A regulatory guide is being written. Because of related ongoing work, do not perform comparison at this time.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23958/C&S: ANSI N170-76

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**Integrated Impact Number:** 650      **SRP Section Number:** 3.8.4

#### **Suggested Changes to the SRP Section:**

SRP Section 3.8.4 cites the following standards:

- ACI 318 No date specified
- ACI 349 No date specified
- ACI 531 1979 and Commentary ACI 531R 1979
- AISC Specifications No date specified
- ANSI N210 1976
- Uniform Building Code 1979

The latest versions are:

- ACI 318 1992
- ACI 349 1990
- ACI 530 1992
- AISC S327 1984
- ANS 57.2 1983
- Uniform Building Code 1991

Consider performing detailed side-by-side comparisons between the cited and latest versions of these standards to allow SRP reviewers to use the latest versions.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

ACI 318, ACI 349 and AISC

No comparison needed at this time. Per a 12/94 telecon with Structural and Seismic Engineering Branch, RES, staff, a commercial contractor is performing an evaluation of the latest versions of ACI 318, ACI 349 and AISC N690, with results expected in early 1995. A decision to proceed with PNL work needs to wait for results from that effort.

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ACI 531 and ICBO UBC

No comparison needed. These standards were used as interim criteria for operating plants and are not applicable to new plants.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23968/C&S: ACI 318; 23969/C&S: ICBO UBC-79; 23970/C&S: AISC Specifications;  
23971/C&S: ACI 349; 23972/C&S: ACI 531-79; 23973/C&S: ANSI N210-76

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**Integrated Impact Number: 653      SRP Section Number: 3.10**

#### **Suggested Changes to the SRP Section:**

SRP Section 3.10 cites ANSI N278.1 1975.

The cited version was redesignated ANSI/ASME278.1 and reaffirmed in 1992.

Consideration should be given to citing the reaffirmed version of this standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23977/C&S: ANSI N278.1-75

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**Integrated Impact Number: 655      SRP Section Number: 3.11**

#### **Suggested Changes to the SRP Section:**

SRP Section 3.11 cites the following standards with no date specified.

IEEE 317 No date specified

IEEE 381 No date specified

IEEE 383 No date specified

IEEE 627 No date specified

The latest versions are:

IEEE 317 1983 R92

IEEE 381 1977 R84

IEEE 383 1974 R92

IEEE 627 1980 R91



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With the exception of IEEE 317, the latest versions of the standards that were in effect at time of last SRP revision have only been reaffirmed, and therefore, are still current. With regard to IEEE 317, the 1983 version (reaffirmed in 1992) is endorsed by Regulatory Guide 1.63.

Consideration should be given to specifying the reaffirmed versions of these standards.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23979/C&S: IEEE 627; 23982/C&S: IEEE 383; 23983/C&S: IEEE 381; 23984/C&S: IEEE 317

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**Integrated Impact Number: 658      SRP Section Number: 5.3.2**

#### **Suggested Changes to the SRP Section:**

Branch Technical Position BTP MTEB 5-2 attached to SRP Section 5.3.2 cites ASTM E208 with no date specified.

The latest version is ASTM E208 1991.

PNL plans to conduct a detailed side-by-side comparison between the cited and latest versions of this standard. Pending completion and review of the side-by-side comparison, consider citing the latest version of this standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23990/C&S: ASTM E208

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**Integrated Impact Number: 659      SRP Section Number: 5.4.7**

#### **Suggested Changes to the SRP Section:**

SRP Section 5.4.7 cites IEEE 338 with no date specified.

The latest version is IEEE 338 1987 R93.

PNL plans to conduct a detailed side-by-side comparison between the cited and latest versions of this standard. Pending completion and review of the side-by-side comparison, consider citing the latest version of this standard.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

23991/C&S: IEEE 338

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**Integrated Impact Number: 661      SRP Section Number: 6.3**

#### **Suggested Changes to the SRP Section:**

SRP Section 6.3 cites ANSI N658 with no date specified. ANSI N658 is also designated as ANS 58.9. ANS 58.9 1981 was in effect at time of last SRP revision and was reaffirmed in 1987.

Consideration should be given to specifying the reaffirmed version of ANS 58.9.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23993/C&S: ANSI N658

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**Integrated Impact Number: 662      SRP Section Number: 6.5.2**

#### **Suggested Changes to the SRP Section:**

SRP Section 6.5.2 cites ANS 56.5 1979.

The cited version was reaffirmed in 1987.

Consideration should be given to citing the reaffirmed version of this standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23146/C&S: ANS 56.5-79

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**Integrated Impact Number: 664      SRP Section Number: 8.2**

#### **Suggested Changes to the SRP Section:**

SRP Section 8.2 cites the following standards:

ANSI C37.04 No date specified

ANSI C37.09 No date specified

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The cited versions were reaffirmed in the following years:

IEEE C37.04 1979 R89  
IEEE C37.09 1979 R89

Consideration should be given to citing the reaffirmed versions of these standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23995/C&S: ANSI C37.04; 23996/C&S: ANSI C37.09; 25679/C&S: ANSI C37.04

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**Integrated Impact Number:** 665      **SRP Section Number:** 8.3.1

#### **Suggested Changes to the SRP Section:**

SRP Section 8.3.1 cites the following standards:

IEEE 304 No date specified  
IEEE 308 No date specified  
IEEE 317 No date specified  
IEEE 338 No date specified  
IEEE 379 No date specified  
IEEE 384 No date specified  
IEEE 387 No date specified

The latest versions are:

IEEE 384 1992  
IEEE 308 1991  
IEEE 317 1983 R92  
IEEE 338 1987 R93  
IEEE 379 1988  
IEEE 384 1992  
IEEE 387 1984 (withdrawn 1993)

PNL plans to conduct detailed side-by-side comparisons between the cited and latest versions of these standards. Pending completion and review of the side-by-side comparisons, consider citing the latest versions of these standards.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

23997/C&S: IEEE 338; 23998/C&S: IEEE 317; 23999/C&S: IEEE 379; 24000/C&S: IEEE 384; 24001/C&S: IEEE 384; 24002/C&S: IEEE 387; 24003/C&S: IEEE 308

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**Integrated Impact Number: 671      SRP Section Number: 9.3.2**

#### **Suggested Changes to the SRP Section:**

SRP Section 9.3.2 cites ANSI N13.1 1969.

The cited version was reaffirmed in 1993.

Consideration should be given to citing the reaffirmed version of this standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22302/C&S: ANSI N13.1

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**Integrated Impact Number: 673      SRP Section Number: 9.5.4**

#### **Suggested Changes to the SRP Section:**

SRP Section 9.5.4 cites DEMA with no date specified. DEMA 1974 was in effect at time of the last SRP revision.

Consideration should be given to specifying the 1974 version of this standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24014/C&S: DEMA Standard

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**Integrated Impact Number: 674      SRP Section Number: 9.5.5**

#### **Suggested Changes to the SRP Section:**

SRP Section 9.5.5 cites DEMA with no date specified. DEMA 1974 was in effect at time of the last SRP revision.

Consideration should be given to specifying the 1974 version of this standard.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24015/C&S: DEMA Standard

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**Integrated Impact Number: 675      SRP Section Number: 9.5.7**

#### **Suggested Changes to the SRP Section:**

SRP Section 9.5.7 cites DEMA with no date specified. DEMA 1974 was in effect at time of the last SRP revision.

Consideration should be given to specifying the 1974 version of this standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24016/C&S: DEMA Standard

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**Integrated Impact Number: 676      SRP Section Number: 12.3 - 12.4**

#### **Suggested Changes to the SRP Section:**

SRP Section 12.3 - 12.4 cites ANSI N13.1 1969.

The cited version was reaffirmed in 1993.

Consideration should be given to citing the reaffirmed version of this standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24018/C&S: ANSI N13.1-69

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**Integrated Impact Number: 677      SRP Section Number: 12.5**

#### **Suggested Changes to the SRP Section:**

SRP Section 12.5 cites the following standards:

ANSI N13.2 1969

ANSI N13.5 1972

ANSI N13.6 1972

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The cited versions were reaffirmed in the following years:

ANSI N13.2 1969 R88  
ANSI N13.5 1972 R89  
ANSI N13.6 1966 R89

Consideration should be given to citing the reaffirmed versions of these standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24019/C&S: ANSI N13.6-72; 24020/C&S: ANSI N13.5-72; 24021/C&S: ANSI N13.2-69

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**Integrated Impact Number:** 679      **SRP Section Number:** 13.1.2 - 13.1.3

#### **Suggested Changes to the SRP Section:**

SRP Section 13.1.2 - 13.1.3 cites ANSI N18.7 with no date specified.

The latest version is ANS 3.2 1988.

Consider performing a detailed side-by-side comparison between the cited and latest version of this standard to allow SRP reviewers to use the latest version

#### **INSPECTION PROGRAM BRANCH COMMENT:**

ANSI N18.7

No comparison needed. ANSI N18.7 is ANS 3.2 which is already being considered for standard comparison work in the SRP-UDP, but placed on hold awaiting industry developments.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24023/C&S: ANSI N18.7

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**Integrated Impact Number:** 681      **SRP Section Number:** 13.2.2

#### **Suggested Changes to the SRP Section:**

SRP Section 13.2.2 cites ANS 3.1 with no date specified.

The latest version is ANS 3.1 1993.

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PNL plans to conduct detailed side-by-side comparisons between the cited and latest versions of these standards. Pending completion and review of the side-by-side comparisons, consider citing the latest versions of these standards.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24026/C&S: ANS 3.1

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**Integrated Impact Number:** 682      **SRP Section Number:** 13.4

#### **Suggested Changes to the SRP Section:**

SRP Section 13.4 cites ANSI N18.7/ANS 3.2. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI N18.7-1976/ANS 3.2-1976 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI N18.7/ANS 3.2 to cite the 1976 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24027/C&S: ANSI N18.7; 24030/C&S: ANS 3.2-78

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**Integrated Impact Number:** 683      **SRP Section Number:** 13.4

#### **Suggested Changes to the SRP Section:**

SRP Section 13.4 cites ANSI N18.1. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI N18.1-1971 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI N18.1 to cite the 1971 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24029/C&S: ANSI N18.1

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**Integrated Impact Number:** 686      **SRP Section Number:** 15.5.1 - 15.5.2

#### **Suggested Changes to the SRP Section:**

SRP Section 15.5.1 - 15.5.2 cites the following two standards:

ANS Trial Use Standard N212 1974 and  
ANSI N18.2 1974.

The latest versions of these two standards are:

ANS 52.1 1983 R88 and  
ANS 51.1 1983 R88.

Consider performing detailed side-by-side comparisons between the cited and latest version of these standards to allow SRP reviewers to use the latest versions.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

No comparisons needed. ANS N212 and ANSI N18.2 are used for definition only.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24037/C&S: ANS N212; 24038/C&S: ANSI N18.2-74

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**Integrated Impact Number:** 688      **SRP Section Number:** 17.2

#### **Suggested Changes to the SRP Section:**

SRP Section 17.2 cites ANSI N45.2.12 with no date specified. ANSI N45.2.12 1977 was in effect at time of last SRP revision and is still current.

Consideration should be given to specifying the 1977 version of this standard.

Consideration should also be given to adding a citation of ASME NQA-1.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

No comparison needed. Per 11/94 conversation with Quality Assurance and Maintenance Branch staff, N45.2.12 requirements are being incorporated into NQA-1 and NQA-2. RG 1.28, Revision 3 endorsed NQA-1. NRC has a program to revise the endorsement based on the results



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of an evaluation of the graded QA program. Also NQA is going through a review of both standards. Because of multiple ongoing activities within the NRC and industry, it would be inadvisable to perform a line standard comparison at this time. Per 11-10-94 telecon with Office of Research staff, two draft regulatory guides were prepared to endorse NQA-1 and NQA-2, respectively, through their 1993 addenda. Both regulatory guides were put on hold due to NRC/NEI work on the graded QA program. In the interim, NQA-1 and NQA-2 were consolidated into a new NQA-1. Work on the N45.2 standards, NQA-1, NQA-2, ANS 3.2 is premature until the work on the graded QA program is resolved.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24041/C&S: ANSI N45.2.12

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**Integrated Impact Number:** 691      **SRP Section Number:** 3.11

#### **Suggested Changes to the SRP Section:**

This impact will not be processed further. Action will be tracked by IPD-7.0 Form 3.11-3.

RG 1.131 is cited in this SRP Section. ASTM D2220 1968 is endorsed by RG 1.131. The latest version of this standard is ASTM D2220 1993.

PNL has recommended performing a detailed side-by-side comparison between the cited and the latest versions of this standard.

Consider future work to revise RG 1.131 to incorporate the results of the side-by-side comparison.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

No comparison needed. D2220 is cited in RG 1.131. RG 1.131 cites D2220 with respect to a flame resistance test. Plant Systems Branch does not recommend comparisons of ASTM fire codes.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

7225/RG 1.131; 24105/C&S: ASTM D2220-68

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**Integrated Impact Number:** 702      **SRP Section Number:** 15.4.8

**Suggested Changes to the SRP Section:**

Regulatory Guide 1.77 is cited in this SRP section. ICRP 2 1959 is endorsed by Regulatory Guide 1.77. The latest version of this standard is ICRP 30 1990.

PNL has recommended performing a detailed side-by-side comparison between the cited and latest versions of this standard.

Consider future work to revise Regulatory Guide 1.77 to incorporate the results of the side-by-side comparison.

**Potential Impacts/Documents supporting the Suggested Changes:**

24135/C&S: ICRP 2 1959; 24214/RG 1.77

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**Integrated Impact Number:** 703      **SRP Section Number:** 15.6.5.A

**Suggested Changes to the SRP Section:**

This impact will not be processed further. Action will be tracked by IPD-7.0 Form 15.6.5.A-2.

RG 1.3 is cited in this SRP Section. ICRP 2 1959 is endorsed by RG 1.3. The latest version of this standard is ICRP 30 1990.

PNL has recommended performing a detailed side-by-side comparison between the cited and the latest versions of this standard.

Consider future work to revise RG 1.3 to incorporate the results of the side-by-side comparison.

**Potential Impacts/Documents supporting the Suggested Changes:**

14516/RG 1.3; 24137/C&S: ICRP 2 1959

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**Integrated Impact Number:** 711      **SRP Section Number:** 2.4.14

**Suggested Changes to the SRP Section:**

ANSI N170, "Standards for Determining Design Basis Flooding at Power Reactor Sites" (1976), is listed in SRP Section 2.4.14 as reference 4, but is not otherwise cited in SRP Section 2.4.14.

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SRP Section 2.4.14 cites Regulatory Guide 1.59 in the Acceptance Criteria which, in turn, cites ANSI N170-1976 as an acceptable method for determining probable maximum and seismically-induced floods on streams.

ANSI N170-1976 was retitled and designated in 1981 as ANSI/ANS 2.8, "Determining Design Basis Flooding at Power Reactor Sites." The current version of this standard is dated 1992.

Consider performing a detailed side-by-side comparison between the cited and the latest version of this standard to allow use of the latest version. Note that this is equivalent to updating Regulatory Guide 1.59.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

No comparison is needed. Per 11/94 conversation with the Civil Engineering and Geosciences Branch staff, Part 100 is under revision and has been issued for public comment. A Regulatory Guide is being written. Because of related ongoing work, do not perform a comparison at this time.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22919/C&S: ANSI N170-76

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**Integrated Impact Number: 713      SRP Section Number: 9.3.1**

#### **Suggested Changes to the SRP Section:**

SRP Section 9.3.1 cites ANSI MC 11.1 / ISA S7.3 1976.

The cited version was reaffirmed in 1981.

Consideration should be given to citing the reaffirmed version of this standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

19081/C&S: ISA S7.3

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**Integrated Impact Number: 714      SRP Section Number: 9.5.6**

#### **Suggested Changes to the SRP Section:**

SRP Section 9.5.6 cites DEMA with no date specified. DEMA 1974 was in effect at time of the

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last SRP revision.

Consideration should be given to specifying the 1974 version of this standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22400/C&S: DEMA Standard

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**Integrated Impact Number: 715      SRP Section Number: 9.5.8**

#### **Suggested Changes to the SRP Section:**

SRP Section 9.5.8 cites DEMA with no date specified. DEMA 1974 was in effect at time of the last SRP revision.

Consideration should be given to specifying the 1974 version of this standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24017/C&S: DEMA Standard

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**Integrated Impact Number: 716      SRP Section Number: 11.5**

#### **Suggested Changes to the SRP Section:**

SRP Section 11.5 cites ANSI N13.1 1969.

The cited version was reaffirmed in 1993.

Consideration should be given to citing the reaffirmed version of this standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23403/C&S: ANSI N13.1-69

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**Integrated Impact Number: 736      SRP Section Number: 15.6.5.A**

#### **Suggested Changes to the SRP Section:**

This impact will not be processed further. Action will be tracked by IPD-7.0 Form 15.6.5.A-3.

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RG 1.4 is cited in this SRP Section. ICRP 2 1959 is endorsed by RG 1.4. The latest version of this standard is ICRP 30 1990.

PNL has recommended performing a detailed side-by-side comparison between the cited and the latest versions of this standard.

Consider future work to revise RG 1.4 to incorporate the results of the side-by-side comparison.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14507/RG 1.4; 24138/C&S: ICRP 2 1959

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**Integrated Impact Number:** 747      **SRP Section Number:** 6.5.1

#### **Suggested Changes to the SRP Section:**

This impact will not be processed further. Action will be tracked by IPD-7.0 Form 6.5.1-2.

RG 1.52 is cited in this SRP Section. RG 1.52 endorses the following standards:

ANSI B124.1 1971  
IEEE 279 with no date specified

The latest versions of these standards are:

UL 900 1985  
IEEE 603 1991

PNL has recommended performing detailed side-by-side comparisons between the cited and the latest versions of these standards.

Consider future work to revise RG 1.52 to incorporate the results of the side-by-side comparisons.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4167/RG 1.52; 24117/C&S: ANSI B124.1-71; 24171/C&S: ANSI N509-76; 24172/C&S: ANSI N510-75; 24195/C&S: IEEE 279

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**Integrated Impact Number:** 763      **SRP Section Number:** 3.8.3

#### **Suggested Changes to the SRP Section:**

Add a discussion of RG 1.136 to SRP Section 3.8.3

RG 1.136 describes bases acceptable to the NRC staff for implementing the requirements of GDC 1 of Appendix A to 10CFR50.

Since the last revision of SRP Section 3.8.3, RGs 1.10, 1.15, and 1.55 have been withdrawn. These RGs are referenced in Section 3.8.3, and therefore, the SRP should be revised accordingly.

RG 1.136 endorses the following of the 1980 edition of the ASME B&PV Code, Section III, Division 2 (also known as ACI Standard 359-80) with supplementary guidelines:

CC-1000, Introduction,  
CC-2000, Material,  
CC-4000, Fabrication and Construction,  
CC-5000, Construction, Testing, and Examination,  
CC-6000, Structural Integrity Test of Concrete Containment Structures.

Consider revising Subsection IV, Evaluation Findings, to incorporate the provisions of RG 1.136.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21095/RG 1.136

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**Integrated Impact Number:** 765      **SRP Section Number:** 3.8.3

#### **Suggested Changes to the SRP Section:**

Add staff position of steel embedments to acceptance criteria and review procedures.

Regulatory Guide 1.142 provides staff positions related to the use of ACI 349-76, "Code Requirements for Nuclear Safety Related Concrete Structures." RG 1.142 excludes Appendix B of ACI 349-76 in its consideration. In the staff's FSER for the System 80+ and ABWR, the staff's exceptions to Appendix B are presented to aid in the review of advanced reactor applications.

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Consider incorporating the staff position on steel embedments into SRP 3.8.3 review procedures and specific acceptance criteria.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23800/ADVANCE SER CE CH 3; 24257/FINAL SER CE80 CH 3; 24258/FINAL SER ABWR APPENDIX F

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**Integrated Impact Number:** 766      **SRP Section Number:** 3.8.3

#### **Suggested Changes to the SRP Section:**

SRP Section 3.8.3 cites AISC, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings." The citation is non-date-specific with regard to the applicable standard version.

Standard AISC N690-1969 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of AISC to cite the 1969 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23964/C&S: AISC Specifications

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**Integrated Impact Number:** 768      **SRP Section Number:** 3.8.4

#### **Suggested Changes to the SRP Section:**

Keep the discussion of GDC 4 but change the reference title.

GDC 4 requires that the structures systems and components important to safety be designed to withstand the effects of dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events outside of the nuclear power unit.

SRP Section 3.8.4.II.C states that the acceptance criteria for the design of safety related structures are based, among others, on GDC 4, including the dynamic effects of equipment failures including missiles and blowdown loads associated with the loss-of-coolant accident.

SRP Section 3.8.4.IV.3 also states that the staff concludes that the requirements of GDC 4 are met.

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However, Rule making in 1987 revised GDC 4, including its title.

10 CFR 50, GDC 4 is now entitled "Environmental and dynamic effects design bases." SRP Section 3.8.4.VI, "References," refers to GDC 4 as "Environmental and Missile Design Bases."

Consider updating the reference to GDC 4 to be consistent with 10 CFR 50.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

5477/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 769      **SRP Section Number:** 3.8.4

#### **Suggested Changes to the SRP Section:**

Consider specifying the current version of ACI-349 in the reference section of SRP 3.8.4.

Since the last revision of the SRP Section 3.8.4 many RGs and industry standards, such as ACI 349, "Code Requirements for Nuclear Safety Related Concrete Structures" have changed.

The current SRP does not reference specific issues of industry standards. For example, RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)" Rev.1, October 1981, endorses ACI 349-76 and its 1979 supplement, except for Appendix B. Meanwhile, the ACI 349 Code has been revised and the latest revision, as indicated in the NUREG/CR-5973, has been issued in 1990.

A detailed comparison between the 1979 and 1990 versions of ACI 349 should be performed to support endorsement of the current version in SRP 3.8.4.

An IPD 7.0 form (# 3.8.3-2) has been initiated for revision of RG 1.142 to endorse the current version of ACI 349.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

5524/RG 1.142

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**Integrated Impact Number:** 771      **SRP Section Number:** 3.8.4

#### **Suggested Changes to the SRP Section:**

Add a discussion of the design requirements for distant safety-related structures to SRP 3.8.4.



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10 CFR 100 Appendix A, V(d)(4) states that those structures which are not located in the immediate vicinity of the site but which are safety related shall be designed to withstand the effect of the Safe Shutdown Earthquake and the design basis for surface faulting determined on a comparable basis to that of the nuclear power plant, taking into account the material underlying the structure and the different location with respect to that of the site.

Consider including a discussion of the design requirements for distant safety- related structures in all subsections of SRP 3.8.4.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20031/CODE OF FED. REGS 10CFR100

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**Integrated Impact Number:** 772      **SRP Section Number:** 3.8.4

#### **Suggested Changes to the SRP Section:**

Update SRP Sections 3.8.4.II.2, "Applicable Codes, Standards and Specifications," and Section 3.8.4.VI, "References," to include RG 1.136, Rev. 2.

With the issuance of RG 1.136, Rev. 2, the following RGs have been withdrawn:

1.10 "Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures"

1.15 "Testing of Reinforcing Bars for Category I Concrete Structures"

1.18 "Structural Acceptance Test for Concrete Primary Reactor Containments"

1.19 "Nondestructive Examination of Primary Containment Liner Welds"

1.55 "Concrete Placement in Category I Structures" and

1.103 "Post-tensioned Prestressing Systems for Concrete Reactor Vessels and Containments.

RGs 1.10, 1.15, and 1.55 are listed in SRP Section 3.8.4.

Consider updating SRP Section 3.8.4 by deleting reference to RGs 1.10, 1.15 and 1.55.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

20033/RG 1.136; 20040/RG 1.10; 20041/RG 1.15; 20042/RG 1.55

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**Integrated Impact Number:** 773      **SRP Section Number:** 3.8.4

#### **Suggested Changes to the SRP Section:**

Provide a discussion regarding the leak-before-break provisions.

"Limited scope" rule making revised GDC 4 in 1986 to allow use of leak-before break (LBB) analyses to exclude PWR primary loop piping double-ended guillotine break as a design basis for protection against dynamic effects. "Broad scope" rule making revised GDC 4 in 1987 to extend LBB to all high-energy piping in all nuclear power plants. To support application of LBB, the staff adopted NUREG-1061, Vol.3, Section 5 into a draft SRP Section 3.6.3 titled "Leak-Before-Break Evaluation Procedures."

Loads resulting from pipe break considerations are incorporated in the load combination equations in the SRP Sections 3.8.1, 3.8.2, 3.8.3, and 3.8.4.

Consider revising these SRP Sections to provide for the new LBB load definitions when the SRP Section 3.6.3 is completed.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21734/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 774      **SRP Section Number:** 3.8.4

#### **Suggested Changes to the SRP Section:**

Add staff position on steel embedments to acceptance criteria and review procedures.

Regulatory Guide 1.142 provides staff positions related to the use of ACI 349-76, "Code Requirements for Nuclear Safety Related Concrete Structures." RG 1.142 excludes Appendix B of ACI 349-76 in its consideration. In the staff's FSERs for the System 80+ and ABWR, the staff's exceptions to Appendix B are presented to aid in the review of advanced reactor applications.

Consider incorporating the staff position on steel embedments into SRP 3.8.4 review procedures and specific acceptance criteria.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24264/FINAL SER CE80 CH 3; 24265/FINAL SER ABWR APPENDIX F

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**Integrated Impact Number:** 775      **SRP Section Number:** 3.8.4

#### **Suggested Changes to the SRP Section:**

Add staff position on use of ANSI/AISC N690-1984.

In the staff's FSERs for the System 80+ and ABWR, the staff found the use of ANSI/AISC N690-1984, "Nuclear Facilities: Steel Safety-Related Structures," to be acceptable with exceptions.

Consider incorporating the staff position on the use of ANSI/AISC N690-1984 into SRP 3.8.4 review procedures and specific acceptance criteria.

In addition, evaluate whether ANSI/AISC N690-1984 has superseded the references to AISC, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings," which are found throughout the current SRP Section 3.8.4. In the staff's FSERs for the ABWR and 80+, the emphasis was placed on N690.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24266/FINAL SER CE80 CH 3; 24267/FINAL SER ABWR APPENDIX G

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**Integrated Impact Number:** 776      **SRP Section Number:** 3.8.4

#### **Suggested Changes to the SRP Section:**

In the staff's FSER for the ABWR, the staff presented its position on dynamic lateral soil pressures on earth retaining walls and embedded walls in the design of the embedded portion of the exterior walls of all seismic category I structures.

Consider incorporating the staff position on dynamic lateral soil pressures into SRP 3.8.4 review procedures and specific acceptance criteria.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24184/NRC STAFF POSITION (Proposed) March 3, 1994; 24268/FINAL SER ABWR APPENDIX H

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**Integrated Impact Number:** 777      **SRP Section Number:** 3.8.4

#### **Suggested Changes to the SRP Section:**

Incorporate in the REVIEW PROCEDURES information regarding the exemption, accepted by the Staff in the evolutionary FSERs, allowing the elimination of the Operating Basis Earthquake (OBE) from seismic design considerations.

As discussed in the ABWR and CE 80+ FSERs, evolutionary plants may eliminate the OBE analysis from the design basis for structures, systems, and components (SSCs). In such cases, the OBE will serve as an inspection level earthquake above which the licensee would shut down the plant and inspect for potential damage to SSCs important to safety. Design certification based upon this single-earthquake design approach is predicated, in part, on the ability to ascertain the need to take action to shut down the plant in case of occurrence of the OBE, and to evaluate the performance of SSCs important to safety.

In Item I.M. of SECY 93-087, the staff presented staff positions related to the elimination of the OBE from design requirements, such as the treatment of earthquake cycles in the fatigue analyses of piping systems and the effects of seismic anchor motion stresses. In SRP Sections 3.8.1 through 3.8.5, loading combination and structural acceptance criteria included explicit treatment of operating basis earthquake loads.

Consider incorporating information in REVIEW PROCEDURES relating to the staff's acceptance of an exemption allowing the elimination of the OBE from design requirements.

In addition, it is noted that a number of Regulatory Guides provide for explicit treatment of OBE loads. An IPD 7.0 Form (Research/Regulatory Action Needs Form) (# 3.8.4-1) has been developed to address the need to revise various regulatory guides to reflect the elimination of OBE design requirements.

(NOTE: Part A of this ROC has been changed upon receipt of guidance regarding the incorporation of evolutionary plant issues. Information related to exemptions accepted by the Staff in the evolutionary FSERs should be provided in REVIEW PROCEDURES as an informational item.)

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23050/SECY 93-087; 25692/FINAL SER CE80 CH 3; 25693/FINAL SER ABWR CH 3

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**Integrated Impact Number:** 778      **SRP Section Number:** 3.8.5

#### **Suggested Changes to the SRP Section:**

Keep the discussion of GDC 4 but change the reference title.

GDC 4 requires that the structures systems and components important to safety be designed to withstand the effects of dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events outside the nuclear power unit.

SRP Section 3.8.5.II.C states that the acceptance criteria for the design of steel containments are based, among others, on GDC 4, including the dynamic effects of equipment failures including missiles and pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit.

SRP Section 3.8.5.IV.3 also states that the staff conclude that the requirements of GDC 4 are met.

However, Rule making in 1987 revised GDC 4, including its title.

10 CFR 50, GDC 4 is now entitled "Environmental and dynamic effects design bases." SRP Section 3.8.5.VI, "References," refers to GDC 4 as "Environmental and Missile Design Bases."

Consider updating the reference to GDC 4 to be consistent with 10 CFR 50.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11439/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 783      **SRP Section Number:** 15.1.1 - 15.1.4

#### **Suggested Changes to the SRP Section:**

Revise review procedures to incorporate the resolution of TMI Action Plan Items II.E.5.1 and II.E.5.2.

The current SRP section states that the resolution of these two TMI Action Plan items need to be incorporated. NUREG-0933 provides references on the resolution of these items which deal with the design sensitivity of B&W reactors to feedwater transients. Modifications for future plants and backfits to operating reactors are indicated, with related activities such as USI A-47

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and USI A-49 discussed. Further evaluation is needed to determine the appropriate changes to this SRP section to incorporate the resolution of these two TMI Action Plan items. In addition, 10 CFR 50.34(f)(2)(xvi) requires the establishment of a design criterion related to TMI Action Plan Item II.E.5.1.

Consider revising review procedures and acceptance criteria to incorporate the resolution of TMI Action Plan Items II.E.5.1 and II.E.5.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20893/CODE OF FED. REGS 10CFR50; 24273/NUREG 0933

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**Integrated Impact Number:** 784      **SRP Section Number:** 15.1.1 - 15.1.4

#### **Suggested Changes to the SRP Section:**

Revise acceptance criteria and references to reflect the staff's acceptance review for the ODYN, ODYNA and REDYA transient analysis computer codes.

SRP Section 15.1.1 - 15.1.4 cites various analytical models acceptable to the staff for transient analysis and references the standard or reference safety analysis reports for the various nuclear steam supply system (NSSS) vendors (References 5 through 8). Reference 5 is the standard safety analysis report for the BWR/6 dated April 1973.

In Generic Letter 80-91, the staff informed BWR applicants and licensees of the requirements to use the ODYN computer code for analysis of pressurization transients. The staff's SER for review of the ODYN code was transmitted to BWR applicants and licensees in Generic Letter 81-08. Additional information regarding the use of the ODYN code was provided in Generic Letter 81-37. Generic Letter 80-91 specifically states that the code is applicable to the analysis of maximum demand failures of feedwater controllers (i.e., increase in feedwater flow), which is one of the transients analyzed in SRP Section 15.1.1-15.1.4.

In the staff's FSER for the ABWR, the staff discussed its acceptance review of the ODYNA and REDYA computer codes for ABWR transient analyses. These codes are modifications of the previously approved ODYN and REDY codes. The staff concluded that these codes are acceptable for design analysis of the ABWR.

Consider revising the specific acceptance criteria and references to include the ODYN, ODYNA and REDYA computer codes as being acceptable to the staff.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24274/FINAL SER ABWR CH 15; 24433/NRC GENERIC LETTER 81-08; 24440/NRC  
GENERIC LETTER 81-37; 25886/NRC GENERIC LETTER 80-91

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**Integrated Impact Number:** 785      **SRP Section Number:** 15.1.1 - 15.1.4

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and References to reflect the staff-approved analysis methods and computer codes for ABB-CE plants.

SRP Section 15.1.1 - 15.1.4 cites various analytical models acceptable to the staff for transient analysis (References 5 through 8). For ABB-CE plants, Reference 7 lists the August 1973 CESSAR for the System 80 design. In the FSER for the System 80+ design certification review, the staff provides a listing of ABB-CE topical reports describing analytical methods and the associated NRC approval letters in Table 15.3 and the approved computer codes for transient and accident analysis in Table 15.4. A number of the topical reports and NRC approval letters have dates later than August 1973.

Consider revising the specific acceptance criteria and references to appropriately include currently-approved analytical methods and computer codes applicable to ABB-CE plants.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24275/FINAL SER CE80 CH 15

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**Integrated Impact Number:** 786      **SRP Section Number:** 9.5.4

#### **Suggested Changes to the SRP Section:**

Revise reference citations for IEEE-387 and Regulatory Guide 1.9.

The SRP Acceptance Criteria cites Regulatory Guide 1.9 and IEEE Standard 387 as related to the design of the diesel engine fuel oil system. The standard in effect in July 1981 when the SRP was issued was IEEE Std 387-77. IEEE Std 387 was revised in 1984. Regulatory Guide 1.9, Rev. 3, which was issued in July 1993, endorses guidelines set forth in IEEE Std 387-1984 with certain provisions. In addition, the title of Regulatory Guide 1.9 was revised.

Consider revising reference citations for IEEE-387 and Regulatory Guide 1.9.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22696/C&S: IEEE 387; 22798/RG 1.9

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**Integrated Impact Number: 787      SRP Section Number: 3.5.1.2**

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to address the use of probabilistic criteria in evaluating the need to provide missile protection.

The ABWR FSER documents a review method utilizing a combined probability to determine the statistical significance of an identified missile. Once a potential missile is identified, its statistical significance is determined by calculating the combined probability of missile occurrence, impacting a significant target, and causing significant damage. If the combined probability is less than  $10(-7)$  per year, the missile is not considered significant. If the combined probability is greater than  $10(-7)$  per year, missile protection of safety-related SSCs is provided by one or more of the following: (1) locating the system or component in a missile-proof structure, (2) separating redundant systems or components for the missile path or range, (3) providing local shields and barriers for systems and components, (4) designing the equipment to withstand the impact of the most damaging missile, (5) providing design features to prevent the generation of missiles, (6) orienting missile sources to prevent missiles from striking equipment important to safety.

Consideration should be given to adding a new Review Procedure to address the use of probabilistic criteria in evaluating the need to provide missile protection.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24291/FINAL SER ABWR CH 3

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**Integrated Impact Number: 788      SRP Section Number: 3.5.1.2**

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to address the use of the high-energy line separation analysis to determine adequate spatial separation for demonstrating protection from missiles generated inside containment.

The review documented in the ABWR FSER contains information on utilizing the results of the high-energy line separation analysis in determining if adequate protection is demonstrated for



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SSC important to safety from missiles that may be generated inside containment. If an applicant uses the spatial separation analysis to justify adequate protection from missiles inside containment, then that evaluation should be consistent with that same applicant's use of spatial separation in response to high-energy line breaks performed under SAR section 3.6.2. For the ABWR design adequate protection can be shown if the SSC important to safety are more than 9.14 m (30 ft) from any high-energy line. If SSC important to safety are less than 9.14 m (30 ft) from any high-energy line, protection can be shown by verifying damage can occur to only one division of safety-related systems. If damage could occur to more than one division of safety-related systems then protection is provided by utilizing barriers, shields, and enclosures.

Consideration should be given to developing a new Review Procedure to address the use of the high-energy line spatial separation analysis and the methods for evaluating adequate separation to demonstrate missile protection for SSC important to safety.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24293/FINAL SER ABWR CH 3

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**Integrated Impact Number:** 789      **SRP Section Number:** 9.5.5

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to verify that the cooling water system capacity will provide adequate volume to maintain system water level and pump NPSH after taking into account expected water loss over 7-day period without refill of the system.

Area of review Item I.1 provides for a review of the functional performance characteristics of the EDECWS with respect to supporting the operation of the Diesel Generator Unit. Review Procedure III.1 addresses specific functional performance characteristics to be verified by the reviewer.

The staff, in the ABWR and ABB-CE System 80+ FSERs, identify specific information regarding the design and capability of the cooling water system and the keep-warm system as interface issues to be provided once the DG manufacturer has been selected. In addition, the staff stated that this information must be provided as part of the application for a combined license. This information addressed the design/capacity items currently addressed by the SRP plus added a requirement to identify the expected water loss over a 7-day period and system volume capacity (needed to ensure adequate volume is available to maintain system water level and pump NPSH without refill).

Consider adding a Review Procedure to verify that the cooling water system capacity will

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provide adequate volume to maintain system water level and pump NPSH after taking into account expected water loss over 7-day period without refill of the system.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23891/FINAL SER ABWR CH 9; 23895/FINAL SER CE80 CH 9

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**Integrated Impact Number:** 790      **SRP Section Number:** 9.5.5

#### **Suggested Changes to the SRP Section:**

Revise reference citation for IEEE Std 387.

The SRP Acceptance Criteria cites Regulatory Guide 1.9 and IEEE Standard 387 as related to the design of the diesel engine cooling water system. The standard in effect in July 1981 when the SRP was issued was IEEE Std 387-1977. IEEE Std 387 was revised in 1984. Regulatory Guide 1.9, Rev. 3, which was issued in July 1993, endorses guidelines set forth in IEEE Std 387-1984 with certain provisions.

Consider revising reference citations for IEEE-387 to cite the 1984 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22613/C&S: IEEE 387

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**Integrated Impact Number:** 791      **SRP Section Number:** 15.1.5.A

#### **Suggested Changes to the SRP Section:**

Update the references to the standard technical specifications.

Technical specifications limits for coolant activity and primary to secondary system leakage are used in the radiological consequence analysis for steam line break accidents. Standard technical specifications (STS) are listed as References 2, 3, and 4 in the current SRP Section, Appendix A. These should be updated to the current standard technical specifications, NUREG-1430, NUREG-1431, and NUREG-1432, for pressurized water reactor plants. In addition, the current SRP Section includes discussions of specific STS sections, figures, and limit values which need to be verified for the updated STS.

Consider incorporating the improved STS into the list of References, and make conforming changes to the review procedures as needed.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24300/NUREG 1430; 24301/NUREG 1432; 24302/NUREG 1431

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**Integrated Impact Number:** 792      **SRP Section Number:** 15.1.5.A

#### **Suggested Changes to the SRP Section:**

Update the industry standard that provides breathing rates and dose conversion factors used in the radiological consequence calculations.

Review procedure III.9 states that the breathing rates and dose conversion factors provided in Regulatory Guide 1.4 should be used in the staff's calculation of offsite doses for the steam line break accident. Regulatory Guide 1.4 uses the corresponding values provided in ICRP 2, dated 1959. The latest version of ICRP 2 is ICRP 30, dated 1990. A standard comparison should be performed to support the update of the citation to the latest version of the ICRP standard.

Consider performing a standard comparison to support an update to ICRP 30 for breathing rate and dose conversion factor values. An associated IPD 7.0 form (# 15.6.5.a-3) for the revision of Regulatory Guide 1.4 has been initiated.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24303/C&S: ICRP 2 1959

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**Integrated Impact Number:** 793      **SRP Section Number:** 9.5.6

#### **Suggested Changes to the SRP Section:**

Revise reference citations for IEEE-387 and Regulatory Guide 1.9.

The SRP Acceptance Criteria cites Regulatory Guide 1.9 and IEEE Standard 387 as related to the design of the diesel engine starting system. The standard in effect in July 1981 when the SRP was issued was IEEE Std 387-77. IEEE Std 387 was revised in 1984. Regulatory Guide 1.9, Rev. 3, which was issued in July 1993, endorses guidelines set forth in IEEE Std 387-1984 with certain provisions. In addition, the title of Regulatory Guide 1.9 was revised.

Consider revising reference citations for IEEE-387 and Regulatory Guide 1.9.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22401/C&S: IEEE 387

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**Integrated Impact Number:** 794      **SRP Section Number:** 9.5.7

#### **Suggested Changes to the SRP Section:**

Revise reference citations for IEEE-387 and Regulatory Guide 1.9.

The SRP Acceptance Criteria cites Regulatory Guide 1.9 and IEEE Standard 387 as related to the design of the diesel engine lubrication system. The standard in effect in July 1981 when the SRP was issued was IEEE Std 387-77. IEEE Std 387 was revised in 1984. Regulatory Guide 1.9, Rev. 3, which was issued in July 1993, endorses guidelines set forth in IEEE Std 387-1984 with certain provisions. In addition, the title of Regulatory Guide 1.9 was revised.

Consider revising reference citations for IEEE-387 and Regulatory Guide 1.9.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22691/C&S: IEEE 387

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**Integrated Impact Number:** 795      **SRP Section Number:** 15.5.1 - 15.5.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and References to reflect the staff's acceptance of ODYNA and REDYA transient analyses computer codes.

SRP Section 15.5.1 - 15.5.2 cites various analytical models acceptable to the staff for transient analyses (References 4 through 7). In the staff's FSER for the ABWR, the staff discussed its acceptance of the ODYNA and REDYA computer codes for ABWR transient analyses. The staff concluded that these codes are acceptable for design analysis of the ABWR.

Consider revising the specific acceptance criteria and references to include the ODYNA and REDYA computer codes as being acceptable to the staff.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24305/FINAL SER ABWR CH 15

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**Integrated Impact Number:** 796      **SRP Section Number:** 15.5.1 - 15.5.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and References to reflect the staff approved analysis methods and computer codes for the ABB-CE plants.

SRP Section 15.5.1 - 15.5.2 cites various analytical models acceptable to the staff for transient analyses (References 5 through 8). In the staff's FSER for the System 80+ design certification review, the staff provides a listing of ABB-CE topical reports describing analytical methods and the associated NRC approval letters in Table 15.3 and the approved computer codes for transient and accident analyses in Table 15.4. A number of the topical reports and the approval letters have dates later than August 1973.

Consider revising the specific acceptance criteria and references to appropriately include currently approved analytical methods and computer codes applicable to ABB-CE plants.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24306/FINAL SER CE80 CH 15

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**Integrated Impact Number:** 797      **SRP Section Number:** 9.5.7

#### **Suggested Changes to the SRP Section:**

Add a review procedure to address lubricating oil inventory.

In the FSER for the ABWR, the staff included a finding that the emergency diesel engine lubricating system is designed with sufficient inventory to support full load diesel operation for 7 days. The diesel generator LCOs in the Standard Technical Specifications, NUREGs-1430 through 1434, require a minimum volume of lubrication oil. The Bases state that the minimum volume requirements in the technical specification are to ensure the ability to support operation of the diesel for 7 days.

Consider adding a review procedure to verify the diesel engine lubrication system is designed with sufficient volume to support the 7 day operating specification.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24313/FINAL SER ABWR CH 9; 24319/NUREG 1430

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**Integrated Impact Number:** 798      **SRP Section Number:** 9.5.8

#### **Suggested Changes to the SRP Section:**

Revise reference citations for IEEE-387 and Regulatory Guide 1.9.

The SRP Acceptance Criteria cites Regulatory Guide 1.9 and IEEE Standard 387 as related to the design of the diesel engine combustion air system. The standard in effect in July 1981 when the SRP was issued was IEEE Std 387-77. IEEE Std 387 was revised in 1984. Regulatory Guide 1.9, Rev. 3, which was issued in July 1993, endorses guidelines set forth in IEEE Std 387-1984 with certain provisions. In addition, the title of Regulatory Guide 1.9 was revised.

Consider revising reference citations for IEEE-387 and Regulatory Guide 1.9.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22693/C&S: IEEE 387

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**Integrated Impact Number:** 799      **SRP Section Number:** 5.2.3

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for austenitic stainless steel to address staff positions more restrictive than RG 1.44.

RG 1.44 is cited in SRP 5.2.3 for control of the use of sensitized stainless steel.

NRC Generic Letter 88-01 regarded implementation of new staff positions covering technical areas related to intergranular stress corrosion cracking in BWR austenitic stainless steel piping. GL 88-01 also transmitted Revision 2 to NUREG-0313, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." The resolution to New Generic Issue 119.4, "BWR Piping Materials," notes that updating of RG 1.44 to reflect the staff's findings in NUREG-0313, Rev. 2 is required.

Consider revising Review Procedures for austenitic stainless steel to address staff positions more restrictive than RG 1.44 as outlined in NUREG 0313, Rev. 2. In addition, consideration should be given to a revision of RG 1.44 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-6.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

1855/NRC GENERIC LETTER 88-01; 2009/RG 1.44; 15761/NUREG 0933; 23414/NUREG 0313

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**Integrated Impact Number: 800      SRP Section Number: 5.2.3**

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address staff positions that are more restrictive than RG 1.31.

SRP Section 5.2.3 cites RG 1.31 as criteria for control of delta ferrite content in stainless steel weld metal. Review Procedures for review of methods for controlling the amount of delta ferrite in stainless steel weld deposits also cite RG 1.31. RG 1.31 specifies ferrite content limits between 5 and 20% for weld metal.

Generic Letter 88-01 transmitted NUREG-0313, Rev. 2 and provides positions related to welding which are applicable to BWRs based upon the recommendations of NUREG-0313, Rev. 2.

In the CE System 80+ FSER, in conjunction with reviews of control rod drive structural and ESF materials, the staff compared the ferrite content limits for austenitic stainless steel castings and weld metal against industry guidelines and NUREG-0313, Rev. 2. NUREG-0313, Rev. 2 identifies a lower ferrite content limit of 7.5% which is more restrictive than the lower limit specified in RG 1.31.

In the CE System 80+ FSER, the staff also accepted the applicant's commitment to limit the maximum ferrite content for austenitic stainless steel weld metal to 15% and concluded that the lower limits specified will provide reasonable assurance that components of these materials will maintain adequate fracture toughness for their 60 year life.

Consider revising Review Procedures for austenitic stainless steel to address staff positions outlined in NUREG-0313, Rev. 2, the CE System 80+ FSER, and industry guidelines. In addition, consideration should be given to revision of RG 1.31 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-7.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1855/NRC GENERIC LETTER 88-01; 2004/RG 1.31; 23354/FINAL SER ABWR CH 5; 23366/FINAL SER CE80 CH 5; 23414/NUREG 0313; 24344/FINAL SER CE80 CH 4; 24345/FINAL SER CE80 CH 6

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**Integrated Impact Number:** 804      **SRP Section Number:** 5.2.3

#### **Suggested Changes to the SRP Section:**

ASTM A-262, pertaining to detecting susceptibility to intergranular attack in stainless steel is referenced in Section VI of the SRP Section 5.2.3. RGs 1.37 and 1.44 are cited in SRP Section 5.2.3 for control of stainless steel cracking. ASTM A-262-68 is endorsed by RG 1.37 and ASTM A-262-70 is endorsed by RG 1.44, with exceptions. The current version of this standard is ASTM A262-1993. PNL has performed a detailed side-by-side comparison between cited and current versions of this standard.

Consider revising the reference to ASTM A-262 to specify the current 1993 version pending NRC review of the side-by-side comparison. Consideration should also be given to revising RGs 1.44 and 1.37 as future work items. The future work recommendations will be tracked with IPD 7.0 Forms number 4.5.1-1 (RG 1.44) and 4.5.1-4 (RG 1.37).

#### **Potential Impacts/Documents supporting the Suggested Changes:**

2009/RG 1.44; 20798/RG 1.37; 23448/C&S: ASTM A262

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**Integrated Impact Number:** 805      **SRP Section Number:** 5.2.3

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for review of the adequacy of austenitic stainless materials in BWRs to resist intergranular stress corrosion cracking (IGSCC).

GSI A-42, pertaining to the high incidence of cracks discovered in heat-affected zones of welds in primary piping in BWRs, was resolved via issuance of NUREG-0313. The original issuance of NUREG-0313 superseded Branch Technical Position (BTP) MTEB 5-7 as indicated in SRP Section 5.2.3. NRC Generic Letter 88-01 provides staff positions for minimizing the probability of cracking in BWR austenitic stainless steel and associated welds. The letter provides positions relating to acceptable base material properties, weld material ferrite content, and fabrication practices. GL 88-01 also transmitted Revision 2 to NUREG-0313, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping."

In the ABWR FSER, the staff noted that the controls to be used during all stages of welding to prevent contamination and sensitization that could cause stress corrosion cracking in austenitic stainless steel conform with the requirements of the appropriate RGs and NUREG 0313, Revision 2. In the EPRI Evolutionary Plant FSER, the staff recommended that licensees and



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applicants follow Rev. 2 of NUREG-0313 to prevent IGSCC in stainless steel.

Consideration should be given to revising Review Procedures to identify current guidance for review of austenitic stainless steel in BWRs to resist IGSCC based upon Generic Letter 88-01, and NUREG-0313, Rev. 2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1855/NRC GENERIC LETTER 88-01; 11074/NUREG 0933; 23354/FINAL SER ABWR CH 5; 23355/FINAL SER EPRI CH 5; 23414/NUREG 0313

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**Integrated Impact Number:** 807      **SRP Section Number:** 5.2.3

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address staff positions on grinding that are more restrictive than RG 1.37.

The SRP currently cites RG 1.37 which includes controls of surface preparation by manual grinding. In the EPRI Evolutionary Plant FSER, the staff reviewed a list of grinding controls applicable to BWRs that are more restrictive than RG 1.37. The staff found the controls to be adequate and noted they would be required for PWRs as well as BWRs.

In the CE System 80+ FSER, the staff discussed the above EPRI controls but did not explicitly take the position that they were applicable to the design (a PWR). The staff accepted the applicant's proposed grinding controls for austenitic stainless steel and noted the applicant's intent to avoid fabrication processes which would severely cold work the surface of austenitic stainless steel components.

Consideration should be given to revising Review Procedures for manual grinding of austenitic stainless reactor coolant pressure boundary materials based upon the above information. In addition, consideration should be given to revision of RG 1.37 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-5.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20798/RG 1.37; 23472/FINAL SER EPRI CH 1; 24335/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 808      **SRP Section Number:** 5.2.3

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of the acceptability of Nickel-Chromium-Iron alloys (e.g., Inconel) as reactor coolant pressure boundary materials.

The EPRI Evolutionary Plant FSER provides a staff position that alloys such as Inconel 600 are susceptible to primary water stress corrosion cracking and should generally not be used in applications where stress corrosion cracking is a concern. The staff also identified alternative alloys considered acceptable with respect to stress corrosion cracking in conjunction with these positions.

In the CE80+ FSER, the staff notes that the applicant plans to use Inconel 690 in lieu of 600 for reactor coolant pressure boundary components. The staff views the Inconel 690 alloy as the preferred nickel base alloy in the primary coolant loops because of its improved corrosion resistance compared to Inconel 600.

Consideration should be given to developing Review Procedures for review of the acceptability of Nickel-Chromium-Iron alloys as reactor coolant pressure boundary materials based upon the staff positions discussed above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23421/FINAL SER EPRI CH 1; 23422/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 809      **SRP Section Number:** 9.1.5

#### **Suggested Changes to the SRP Section:**

Modify acceptance criteria and review procedures related to the use of NUREG- 0612.

Generic Letter 85-11 discussed the implementation of guidelines presented in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and referred to Phase I and Phase II. Based on a review of Generic Letter 81-07, Phase I (which is associated with a six month report) deals with NUREG 0612, Section 5.1.1, which provides general requirements for overhead handling systems. Phase II (which is associated with a nine month report) deals with NUREG 0612, Sections 5.1.2, 5.1.3, and 5.1.5, which provide specific requirements for overhead handling systems in the vicinity of fuel storage pools, in containment, and in plant areas containing equipment required for reactor shutdown, core decay heat removal, or spent fuel pool cooling.

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Generic Letter 85-11 states that based on the improvements in heavy loads handling obtained from implementation of Phase I, further action per Phase II is not required to reduce the risks associated with the handling of heavy loads. Phase II implementation is not required and any associated license conditions may be deleted in a license amendment request.

Consider modifying acceptance criteria and review procedures to emphasize Phase I guidelines and indicate the staff position on Phase II guidelines.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21442/NRC GENERIC LETTER 85-11

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**Integrated Impact Number:** 810      **SRP Section Number:** 9.1.5

#### **Suggested Changes to the SRP Section:**

Modify review procedures related to the use of NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants."

Review Procedure III.4 discusses the use of NUREG-0554 in the review of the design of cranes for single failure considerations. In Generic Letter 83-42, the staff discussed concerns that NUREG-0554 may be deficient in assuring single failure proof cranes. The concerns related specifically to assuring that a single failure in the electrical power/control system, such as an electrical failure in one of the three phase power leads, will not cause a load drop.

Consider modifying review procedures to reflect the staff concern on the use of NUREG-0554.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21443/NRC GENERIC LETTER 83-42

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**Integrated Impact Number:** 811      **SRP Section Number:** 5.3.1

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to reflect requirements of 10 CFR 50 Appendix G and to use Reg. Guide 1.99 as guidance relative to the effects of irradiation on reactor vessel material fracture toughness.

SRP 5.3.1 includes GDC 31 and Appendix G to 10 CFR 50 as Acceptance Criteria. GDC 31 requires that the Reactor Coolant Pressure Boundary design reflect consideration of service

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temperatures and other conditions of the boundary material, including the effects of irradiation on material properties. SRP 5.3.1 provides Acceptance Criteria and Review Procedures to assess an applicant's compliance with 10 CFR 50 Appendix G with respect to the initial fracture toughness of reactor vessel material. However, Appendix G, as amended in 1983, 1986 and 1988, requires assessment of fracture toughness of reactor vessel beltline material throughout the life of the vessel. Regulatory Guide 1.99 describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. The CE80+ FSER states that the radiation-induced shifts in reference temperatures for these materials can be predicted with reasonable accuracy and conservatism using the methodology of RG 1.99, Revision 2.

SRP Section 5.3.1 cites Regulatory Guide 1.65, Positions C.1 and C.2 in Acceptance Criteria II.7. The Regulatory Guide positions make reference to portions of 10 CFR 50 Appendix G that are no longer appropriate as a result of amendments to Appendix G. Regulatory Guide 1.65 should be revised to cite the appropriate sections of Appendix G as amended.

Consider revising Acceptance Criteria and Review Procedures to reflect Appendix G requirements for reactor vessel fracture toughness over the expected life of the vessel. In addition, consider adding Reg. Guide 1.99 as guidance for calculating the effects of neutron radiation on the reactor vessel material fracture toughness. Changes to Regulatory Guide 1.65 will be tracked by IPD 7.0 form 5.3.1-2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

349/RG 1.65; 401/CODE OF FED. REGS 10CFR50; 428/RG 1.99; 2919/CODE OF FED. REGS 10CFR50; 23447/FINAL SER CE80 CH 5

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**Integrated Impact Number: 812      SRP Section Number: 5.3.1**

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for review of the adequacy of austenitic stainless materials in BWRs to resist intergranular stress corrosion cracking (IGSCC).

Generic Issue A-42 addressed the issue of pipe cracking in the heat-affected zones of welds in primary system piping (including vessel safe ends) in BWRs. This issue was resolved with the initial issuance of NUREG-0313. NRC Generic Letter 88-01 and its supplement provide staff positions for minimizing the probability of cracking in BWR austenitic stainless steel and associated welds. The letter provides positions relating to acceptable base material properties, weld material ferrite content, and fabrication practices. GL 88-01 also transmitted Revision 2 to NUREG-0313, "Technical Report on Material Selection and Process Guidelines for BWR

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Coolant Pressure Boundary Piping."

In the ABWR FSER, the staff noted that the controls to be used during all stages of welding to prevent contamination and sensitization that could cause stress corrosion cracking in austenitic stainless steel conform with the requirements of the appropriate RGs and NUREG 0313, Revision 2. In the EPRI Evolutionary Plant FSER, the staff recommended that licensees and applicants follow Rev. 2 of NUREG-0313 to prevent IGSCC in stainless steel.

Consideration should be given to revising Review Procedures to identify current guidance for review of austenitic stainless steel in BWRs to resist GSCC based upon Generic Letter 88-01, and NUREG-0313, Rev. 2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11076/NUREG 0933; 23368/NRC GENERIC LETTER 88-01; 23409/FINAL SER ABWR CH 5; 23410/NUREG 0313; 23411/FINAL SER EPRI CH 5

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**Integrated Impact Number: 813      SRP Section Number: 5.3.1**

#### **Suggested Changes to the SRP Section:**

Provide Acceptance Criteria and Review Procedures for the reactor vessel material surveillance program beyond a 40-year design lifetime.

General Design Criterion 32, cited as Acceptance Criteria in SRP Section 5.3.1, requires an appropriate materials surveillance program. SRP 5.3.1 reviews the surveillance program for reactor vessel materials. 10 CFR 50 Appendix H and ASTM E-185 are cited as providing the requirements for the surveillance program; however, they are based on a 40-year design life (32 effective full-power years). The ABWR FSER and the CE80+ FSER acknowledge the fact that these reactors are designed for a 60-year life and require that the reactor vessel material surveillance programs account for the 20-year increase in expected life of the vessel.

Consider providing Acceptance Criteria and Review Procedures for the reactor vessel material surveillance program beyond a 40-year design lifetime. In addition, consider future work to revise 10 CFR 50 Appendix H to accommodate a 60-year design life. The future work will be tracked by IPD-7.0 Form Number 5.3.1-1

#### **Potential Impacts/Documents supporting the Suggested Changes:**

404/CODE OF FED. REGS 10CFR50; 2920/CODE OF FED. REGS 10CFR50; 23451/FINAL SER CE80 CH 5; 23452/FINAL SER ABWR CH 5

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**Integrated Impact Number:** 814      **SRP Section Number:** 5.3.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for the reactor vessel materials cleanliness and cleanliness controls.

The SRP currently cites RG 1.37 with regard to cleanliness and cleanliness controls. RG 1.37 cites ANSI N45.2.1 in describing the staff's position with regard to cleanliness and cleanliness controls. In the CE 80+ FSER, the staff indicated that ANSI/ASME NQA-2 supersedes ANSI N45.2.1, cited in RG 1.37, and indicated that they had reviewed ANSI/ASME NQA-2-1983 and found it acceptable. The staff requested that this be reflected in the applicants SAR.

Consideration should be given to revising existing Review Procedures related to cleanliness and cleanliness controls to cite ANSI/ASME NQA-2-1983 in addition to RG 1.37. In addition, consideration should be given to revision of RG 1.37 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-3.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

Per 11/94 conversation with Quality Assurance and Maintenance Branch staff, N45.2.1 requirements are being incorporated into NQA-1 and NQA-2. RG 1.28, Revision 3 endorsed NQA-1. NRC has a program to revise the endorsement based on the results of an evaluation of the graded QA program. Also NQA is going through a review of both standards.

Per 11-10-94 telecon with Office of Research staff, two draft regulatory guides were prepared to endorse NQA-1 and NQA-2, respectively, through their 1993 addenda. Both regulatory guides were put on hold due to NRC/NEI work on the graded QA program. In the interim, NQA-1 and NQA-2 were consolidated into a new NQA-1.

#### **PNL COMMENT:**

This is a placeholder integrated impact and will not be processed further at this time.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

445/RG 1.37; 23432/FINAL SER CE80 CH 4; 23433/C&S: ANSI N45.2.1; 23434/C&S: ASME NQA-2

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**Integrated Impact Number:** 815      **SRP Section Number:** 5.3.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address staff positions that are more restrictive than RG 1.31.

SRP Section 5.3.1 uses RG 1.31 as criteria for control of delta ferrite content in stainless steel weld metal. RG 1.31 specifies ferrite content limits between 5 and 20% for weld metal.

Staff evaluation in Section 5.3.1 of the CE System 80+ FSER indicates that in Section 5.2.3 of this FSER, the staff compared the ferrite content limits for austenitic stainless steel castings and weld metal against industry guidelines and NUREG-0313, Rev. 2 (Generic Letter 88-01, Attachment A). NUREG-0313, Rev. 2 identifies a lower ferrite content limit of 7.5% which is more restrictive than the lower limit specified in RG 1.31.

Consider revising Review Procedures for austenitic stainless steel to address staff positions outlined in NUREG-0313, Rev. 2 and industry guidelines. In addition, consideration should be given to revision of RG 1.31 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-7.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1200/RG 1.31; 23367/FINAL SER CE80 CH 5; 23368/NRC GENERIC LETTER 88-01; 24598/FINAL SER ABWR CH 5

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**Integrated Impact Number:** 816      **SRP Section Number:** 5.3.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for austenitic stainless steel to address staff positions more restrictive than RG 1.44.

RG 1.44 is cited in SRP 5.3.1 for control of the use of sensitized stainless steel.

NRC Generic Letter 88-01 regarded implementation of new staff positions covering technical areas related to intergranular stress corrosion cracking in BWR austenitic stainless steel piping. GL 88-01 also transmitted Revision 2 to NUREG-0313, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." The resolution to New Generic Issue 119.4, "BWR Piping Materials," notes that updating of RG 1.44 to reflect the staff's findings in NUREG-0313, Rev. 2 is required.

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Consider revising Review Procedures for austenitic stainless steel to address staff positions more restrictive than RG 1.44 as outlined in NUREG 0313, Rev. 2. In addition, consideration should be given to a revision of RG 1.44 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-6.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

448/RG 1.44; 15762/NUREG 0933; 23368/NRC GENERIC LETTER 88-01

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**Integrated Impact Number: 819      SRP Section Number: 5.3.1**

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to incorporate 10 CFR 50.60 and NUREG-0744.

At present, the SRP Acceptance Criteria include Appendices G and H of 10 CFR 50 with respect to reactor vessel fracture toughness and surveillance program. The regulatory requirement to comply with Appendices G and H is contained in 10 CFR 50.60 which is not currently identified in SRP 5.3.1.

Appendix G to 10 CFR Part 50 requires that the Charpy upper shelf energy throughout the life of the vessel be no less than 50 ft-lb unless it is demonstrated that lower values will provide margins of safety against failure equivalent to those provided by Appendix G of the ASME code. USI A-11 was initiated to address the staff's concern that some vessels were projected to have beltline materials with Charpy upper shelf energy less than 50 ft-lb. NUREG-0744, developed as resolution of USI A-11, provides a method for evaluating reactor vessel materials when their Charpy upper shelf energy is predicted to fall below 50 ft-lb.

Consider revising Acceptance Criteria to include 10 CFR 50.60 in connection with 10 CFR 50 Appendices G and H. Also consider revising Review Procedures to reference the methodology in NUREG-0744, Rev. 1 for evaluation of vessel materials Charpy upper shelf energy less than 50 ft-lb.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

307/CODE OF FED. REGS 10CFR50; 401/CODE OF FED. REGS 10CFR50; 404/CODE OF FED. REGS 10CFR50; 1540/NUREG 0933; 2402/NRC GENERIC LETTER 82-26; 21615/NRC GENERIC LETTER 92-01

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**Integrated Impact Number:** 820      **SRP Section Number:** 5.3.1

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria, Review Procedures and Evaluation Findings for determining PWR reactor vessel acceptability under pressurized thermal shock (PTS) conditions.

SRP 5.3.1 includes Review Procedures for calculations of reactor vessel material nil-ductility transition reference temperatures.

Unresolved Safety Issue A-49 was initiated to assess the significance of simultaneous occurrence of a severe overcooling transient and a repressurization of a PWR primary system. The NRC resolved the pressurized thermal shock issue for operating reactors with the publication of 10 CFR 50.61 that establishes a screening criterion for the reference temperatures for nil-ductility transition. Licensees with plants predicted to exceed the criterion before the expiration of the license were required to submit plans for the reduction of neutron flux. Plants that will exceed the screening criteria are required to submit safety analyses for modifications to prevent failure of the reactor vessel as a result of postulated PTS events after the screening limit has been exceeded. Regulatory Guide 1.154 outlines recommended methods to be used in performing these analyses. The Regulatory Guide specifies the expected format and content of plant-specific PTS safety analysis reports and describes acceptance criteria that the NRC staff will use in evaluating licensee analyses and proposed corrective measures.

In the CE80+ FSER, the staff assessed the likelihood that the CE80+ reactor vessel will remain below the PTS screening criteria. The staff concluded that CE's approach to calculating reference temperatures has enough conservatism to provide adequate assurance of fracture toughness in a PTS event for its design life.

Consider adding 10 CFR 50.61 and Regulatory Guide 1.154 to Acceptance Criteria and developing Review Procedures and Evaluation Findings related to PWR reactor vessel acceptability under PTS conditions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

463/RG 1.154; 6065/NUREG 0933; 21032/CODE OF FED. REGS 10CFR50; 21615/NRC GENERIC LETTER 92-01; 23463/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 821      **SRP Section Number:** 13.6

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and develop Review Procedures for review of the physical security organization, including applicable requirements from Appendix B to 10 CFR Part 73, and guidance from NUREG-0674 and NUREG-0908.

SRP Section 13.6 (last revised in July 1981) lists Section 73.55(b) of 10 CFR 73 and Appendix B to 10 CFR 73 as specific acceptance criteria II.a and II.h. NUREG-0674 is listed as Acceptance Criterion II.4. No specific Review Procedures are provided for these criteria.

10 CFR 73.55(b), established in 1977 and revised in 1979, 1981, 1988 and 1992, defines requirements to be met by the licensee physical security organization. These requirements include: demonstration of the ability of the physical security personnel to perform their assigned duties; provisions for the development, revision, implementation, and enforcement of security procedures; and training, qualification and equipping of security personnel.

Appendix B to 10 CFR 73, established in 1978 and revised in 1981, 1988, and 1992, provides general criteria for security personnel. These general criteria establish requirements for the selection, training, equipping, testing, and qualification of individuals who will be responsible for protecting nuclear facilities. NUREG-0674, issued in 1980, identifies basic criteria for use in evaluating the responsiveness of guard training plans to the requirements of Appendix B, 10 CFR 73.

Reg. Guide 5.20, issued in 1974, provides criteria for a program for training, equipping and qualifying guards and watchmen.

NUREG-0908, published in 1982, was developed to assist the NRC reviewers in their review of new or revised security plans by outlining specific criteria for the elements of such plans. Section 3 of NUREG-0908 outlines criteria for the security organization, including: establishment of the security organization; security organization management; qualification for employment in security; training of plant personnel; and, security personnel equipment. In the EPRI Evolutionary Plant FSER, the staff documented their use of NUREG-0908 in review of security-related design criteria.

Consider revising Acceptance Criteria and developing Review Procedures for review of the physical security organization. These revisions should reference applicable requirements from Appendix B to 10 CFR Part 73 and guidance from NUREG-0908, as discussed above. Further evaluation is required to determine if NUREG-0674 should be retained as an Acceptance Criterion for SRP Section 13.6.

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In addition, consideration should be given to future work to revise Reg. Guide 5.20 so it can be referenced as guidance in SRP Section 13.6 in lieu of the outdated NUREG-0674 and NUREG-0908. This future work will be tracked on IPD-7.0 Form 13.6-1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11770/CODE OF FED. REGS 10CFR73; 11837/CODE OF FED. REGS 10CFR73; 12208/RG 5.20; 24044/FINAL SER EPRI CH 9; 24045/NUREG 0908; 24058/NUREG 0674

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**Integrated Impact Number:** 823      **SRP Section Number:** 5.3.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to address staff positions on grinding that are more restrictive than RG 1.37.

The SRP currently cites RG 1.37 which includes controls of surface preparation by manual grinding. In the EPRI Evolutionary Plant FSER, the staff reviewed a list of grinding controls applicable to BWRs that are more restrictive than RG 1.37. The staff found the controls to be adequate and noted they would be required for PWRs as well as BWRs.

In the CE System 80+ FSER, the staff discussed the above EPRI controls but did not explicitly take the position that they were applicable to the design (a PWR). The staff accepted the applicant's proposed grinding controls for austenitic stainless steel and noted the applicant's intent to avoid fabrication processes which would severely cold work the surface of austenitic stainless steel components.

Consideration should be given to revising Review Procedures for manual grinding of austenitic stainless steel reactor vessel materials based upon the above information. In addition, consideration should be given to revision of RG 1.37 as future work. Future work will be tracked by IPD-7.0 Form Number 4.5.1-5.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

445/RG 1.37; 23473/FINAL SER EPRI CH 1; 25476/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 825      **SRP Section Number:** 5.3.1

#### **Suggested Changes to the SRP Section:**

ASTM E-185, "Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear

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Power Reactor Vessels," is cited as acceptance criteria for material surveillance requirements in SRP Section 5.3.1. Since the SRP citation is not date specific, the version of this standard in effect at the time of issuance of SRP Section 5.3.1 (July 1981) would be applied as acceptance criteria for reviews of reactor vessel materials. The latest version of this standard is ASTM E-185-94. PNL is not currently performing a detailed side-by-side comparison between cited and latest versions of this standard.

Consider revising the reference to ASTM-185 to specify the latest (1994) version.

PNL COMMENT: This is a placeholder integrated impact and will not be processed further. The impact was edited to state that a side-by-side comparison is not currently planned for this standard and the latest version is 1994.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23575/C&S: ASTM E185

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**Integrated Impact Number:** 827      **SRP Section Number:** 13.5.2.1

#### **Suggested Changes to the SRP Section:**

Update current discussions of review applicability to address the review of DC and COL applications.

The standard plant licensing process of 10 CFR 52 has provisions for determining that certain aspects of a design may be determined to be outside the scope of design certification (DC) and consequently must be addressed by a combined license (COL) applicant referencing the certified design. The ABWR and CE 80+ FSERs indicate that the staff has determined that development of detailed procedures and associated training materials may be beyond the scope of design certification and the responsibility of a COL applicant referencing the certified design. SRP Section 13.5.2 currently describes differences in the review of plant procedures for the cases of applications for construction permits or operating licenses.

Consideration should be given to revising SRP 13.5.2 to indicate that development of detailed procedures and associated training material may be beyond the scope of design certification and in such a case, would be the responsibility of a COL applicant referencing the certified design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24231/FINAL SER CE80 CH 13; 24232/FINAL SER ABWR CH 13

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**Integrated Impact Number:** 830      **SRP Section Number:** 13.5.2.1

#### **Suggested Changes to the SRP Section:**

10 CFR 50, Appendix B, criteria V and VI, establish requirements for the development, approval, and control of procedures for all activities affecting quality. 10 CFR 50, Appendix B, is identified in the regulatory basis for Regulatory Guide 1.33 currently cited in SRP Section 13.5.2. These requirements are not currently identified in SRP Section 13.5.2 Acceptance Criteria.

Consideration should be given to adding citations of 10 CFR 50, Appendix B, criterion V and criterion VI, to SRP Acceptance Criteria.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

19940/CODE OF FED. REGS 10CFR50; 24233/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 831      **SRP Section Number:** 13.6

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and develop Review Procedures for review of physical barriers, vital areas, isolation zones and lighting. This would include applicable requirements from 10 CFR 73.2 and applicable guidance from 10 CFR 8.5(b), 10 CFR 8.5(c), Reg. Guide 5.44, Reg. Guide 5.65, Reg. Guide 5.68, NUREG-0908, and NRC Review Guideline 17.

SRP Section 13.6 (last revised in July 1981) lists Section 73.55(c) of 10 CFR 73 as specific acceptance criterion II.b. No specific Review Procedures are provided for this criterion.

10 CFR 73.55(c) sets requirements for: (1) location of vital equipment within a vital area, and at least two physical barriers; (2) separation of protected area and vital area barriers; (3) isolation zones adjacent to physical barriers at the perimeter of protected areas; (4) detection of penetration or attempted penetration of a protected area or adjacent isolation zone; (5) illumination of isolation zones and exterior areas within a protected area; (6) bullet resistant barriers around the control room; and (7) vehicle control measures.

10 CFR 8.5 provides NRC General Counsel interpretations of 10 CFR 73.55. Sections 8.5(b) and 8.5(c) interpret 73.55(c)(5) requirements for illumination of the protected area and isolation zones.

10 CFR 73.2 defines terms including "physical barriers," "vital area" and "vital equipment." The

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73.2 definition of "physical barriers" establishes detailed requirements for barriers to qualify as physical barriers under Part 73.

Reg. Guide 5.44, revised in 1980, describes six types of perimeter intrusion alarm systems and sets forth criteria for their performance and use as a means acceptable to the NRC staff for meeting specified portions of the Commission's regulations. Reg. Guide 5.65, issued in 1986, provides guidance for openings in vital area barriers, separation of protected area and vital area barriers at water sources, and designation of spent fuel pools as time-dependent vital areas.

Reg. Guide 5.68, Revision 0, issued August 1994 describes methods acceptable to the NRC staff by which the licensee can meet the requirements of the amended 10 CFR 73.1(a)(1) and 73.55(c)(7), (8), (9), and (10) regarding protection against land vehicle bombs.

NUREG-0908, published in 1982, was developed to assist the NRC reviewers in their review of new or revised security plans by outlining specific criteria for the elements of such plans. Section 4 outlines criteria and guidelines for physical barriers, including: protected area barriers; vital area/island barriers; security posts and structures; keys, locks and combinations; and testing and maintenance. Section 5.6 outlines criteria and guidelines for vital area/island compartmentalization. Section 6.1 outlines criteria and guidelines for illumination. In the EPRI Evolutionary Plant FSER, the staff documented their use of NUREG-0908 in review of security-related design criteria.

In the EPRI Evolutionary Plant FSER, the CE 80+ FSER and the ABWR FSER, the staff documented their application of NRC Review Guideline 17 (January 23, 1978, memorandum from R. Clark to safeguards licensing staff) in their reviews of vital area and vital equipment designations. In the CE 80+ FSER, the staff documented their position, based upon Review Guideline 17, that all seismic Category I equipment should be designated as vital equipment. In the ABWR FSER, the staff discussed their positions on vital equipment outside of vital areas and on effectiveness of ventilation barriers.

Consider revising Acceptance Criteria and developing Review Procedures for review of physical barriers, vital areas, isolation zones, and lighting, to incorporate the requirements and to reference applicable guidance, as discussed above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11719/CODE OF FED. REGS 10CFR08; 11759/CODE OF FED. REGS 10CFR73;  
11774/CODE OF FED. REGS 10CFR73; 12244/RG 5.44; 12349/RG 5.65; 22063/FINAL  
SER EPRI CH 3; 24044/FINAL SER EPRI CH 9; 24046/NUREG 0908; 24047/FINAL SER  
ABWR CH 13; 24048/FINAL SER CE80 CH 13; 24049/FINAL SER EPRI CH 9;  
24357/CODE OF FED. REGS 10CFR73; 24362/RG 5.68

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**Integrated Impact Number:** 832      **SRP Section Number:** 13.6

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and develop Review Procedures for review of access requirements. This would include requirements from 10 CFR 73.70(d) and interpretations from 10 CFR 8.5(d) and 10 CFR 8.5(e). Applicable guidance from Reg. Guide 5.12, Reg. Guide 5.65, NUREG-0908, Generic Letter (GL) 83-21 and GL 87-08 should also be referenced.

SRP Section 13.6 (last revised in July 1981) lists Section 73.55(d) of 10 CFR 73 as specific acceptance criterion II.c. No specific Review Procedures are provided for this criterion.

10 CFR 73.55(d), established in 1977 and revised in 1986 and 1988, sets requirements for: (1) controlling all points of personnel and vehicle access; (2) searching hand-carried packages; (3) checking and searching all packages and materials for delivery into the protected area; (4) searching and controlling all vehicles entering the protected area; (5) a numbered picture badge identification system; (6) escorts for individuals not authorized for unescorted access; (7) an access authorization system; (8) positive access control for reactor containment; and, (9) control of keys, locks, combinations and related access control devices.

10 CFR 8.5 provides NRC General Counsel interpretations of 10 CFR 73.55. Sections 8.5(d) and 8.5(e) interpret 73.55(d)(1) requirements for searches to detect explosives, firearms and incendiary devices.

10 CFR 73.70(d) requires a log indicating name, badge number, time of entry, and time of exit of all individuals granted access to a vital area. The ABWR FSER documents the staff's application of the record-keeping requirement of Section 73.70(d) in their review of access control methods.

NUREG-0908, published in 1982, was developed to assist the NRC reviewers in their review of new or revised security plans by outlining specific criteria for the elements of such plans. Section 5 outlines criteria and guidelines for controlling access, including: access authorization, picture badge systems, searches, access/entry, escorts, vital area/island compartmentalization, and records. In the EPRI Evolutionary Plant FSER, the staff documented their use of NUREG-0908 in review of security-related design criteria.

Reg. Guide 5.12, listed in SRP Section 13.6 as Acceptance Criterion II.7, provides criteria acceptable to the Regulatory Staff for the selection and use of commercially available locks in the protection of facilities. Reg. Guide 5.65, issued in 1986, provides guidance for control of access to vital areas under routine and emergency conditions and for suspending security

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measures.

Generic Letter 83-21 clarifies 73.55(d)(1) by stating that on-duty U.S., state, and local law enforcement officers need not leave their firearms and ammunition outside protected areas, provided that they have positive identification of their status. Generic Letter 87-08 provides staff positions on and interpretations of access requirements.

In the EPRI Evolutionary Plant FSER, the staff stated their position that protection against insider sabotage will be provided by vital area access controls, and that access controls to vital areas be located only at vital area boundaries and not between redundant divisions of vital components. In the ABWR FSER, the staff stated their position that compliance with 73.55(d)(7)(ii) should include consideration of an emergency requiring evacuation of the control room in the control building to the remote shutdown panel in the reactor building.

Consider revising Acceptance Criteria and developing Review Procedures for review of access requirements to incorporate requirements and to reference applicable guidance, as discussed above. (Also see Integrated Impact #838 regarding personnel access authorization program reviews.)

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11724/CODE OF FED. REGS 10CFR08; 11807/CODE OF FED. REGS 10CFR73; 12198/RG 5.12; 12352/RG 5.65; 12390/NRC GENERIC LETTER 87-08; 12449/NRC GENERIC LETTER 83-21; 24044/FINAL SER EPRI CH 9; 24050/CODE OF FED. REGS 10CFR73; 24051/FINAL SER ABWR CH 13; 24052/NUREG 0908; 24053/FINAL SER EPRI CH 6

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**Integrated Impact Number:** 833      **SRP Section Number:** 13.6

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and develop Review Procedures for review of detection aids, including applicable guidance from NUREG-0908 and Reg. Guides 5.12, 5.44 and 5.65.

SRP Section 13.6 (last revised in July 1981) lists Section 73.55(e) of 10 CFR 73 as specific acceptance criterion II.d. Reg. Guides 5.44 and 5.12 are listed as Acceptance Criteria II.3 and II.7, respectively. No specific Review Procedures are provided for these criteria.

10 CFR 73.55(e), established in 1977 and revised in 1988, sets requirements for: (1) alarms and alarm stations; (2) tamper resistance and self-checking of alarm devices; and, (3) alarming of emergency exits.



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NUREG-0908, published in 1982, was developed to assist the NRC reviewers in their review of new or revised security plans by outlining specific criteria for the elements of such plans. Section 6 outlines criteria and guidelines for detection aids, including: surveillance, alarm/intrusion, false and nuisance alarm rates, tamper indication and self-test capabilities, compensatory measures, central alarm station and secondary alarm station operation, security patrols, and records. In the EPRI Evolutionary Plant FSER, the staff documented their use of NUREG-0908 in review of security-related design criteria.

Reg. Guide 5.12 includes guidance regarding connection of electric locks with the alarm system.

Reg. Guide 5.44 describes six types of perimeter intrusion alarm systems and sets forth criteria for their performance and use.

Reg. Guide 5.65, issued in 1986, includes guidance for intrusion alarms on vital area barriers at water sources and for alarms on unattended vital area and emergency ingress/egress doors.

Consider revising Acceptance Criteria and developing Review Procedures for review of detection aids. These revisions should reference applicable guidance from NUREG-0908 and from Reg. Guides 5.12, 5.44 and 5.65, as discussed above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11816/CODE OF FED. REGS 10CFR73; 12198/RG 5.12; 12244/RG 5.44; 12349/RG 5.65; 12352/RG 5.65; 24044/FINAL SER EPRI CH 9; 24054/NUREG 0908

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**Integrated Impact Number:** 834      **SRP Section Number:** 13.6

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of communication systems.

SRP Section 13.6 (last revised in July 1981) lists Section 73.55(f) of 10 CFR 73 as specific acceptance criterion II.e. No specific Review Procedures are provided for this criterion.

10 CFR 73.55(f), established in 1977, sets requirements for: (1) communication capability between security personnel and alarm stations; (2) telephone communication capability between alarm stations and law enforcement authorities; (3) radio or microwave communication capability between alarm stations and law enforcement authorities; and, (4) power supplies for communication equipment.

NUREG-0908, published in 1982, was developed to assist the NRC reviewers in their review of

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new or revised security plans by outlining specific criteria for the elements of such plans. Section 7 outlines criteria and guidelines for general communication among elements of the security force and with law enforcement authorities. In the EPRI Evolutionary Plant FSER, the staff documented their use of NUREG-0908 in review of security-related design criteria.

In the CE80+ FSER, the staff stated their position that licensees/applicants should address transmissibility of radio signals as well as equipment shielding to prevent radios from interfering with plant equipment.

Consider developing Review Procedures for review of communication systems.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11817/CODE OF FED. REGS 10CFR73; 24044/FINAL SER EPRI CH 9; 24055/NUREG 0908; 24056/FINAL SER CE80 CH 13

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**Integrated Impact Number:** 835      **SRP Section Number:** 13.6

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and develop Review Procedures for review of testing and maintenance programs and program reviews, including requirements from 10 CFR 26.80, 10 CFR 50.54(p) and 10 CFR 73.56(g) and applicable guidance from NUREG-0908 and Reg. Guides 5.44 and 5.65.

SRP Section 13.6 (last revised in July 1981) lists Section 73.55(g) of 10 CFR 73 as specific acceptance criterion II.f. Reg. Guide 5.44 is listed as Acceptance Criterion II.3. No specific Review Procedures are provided for these criteria.

10 CFR 73.55(g), established in 1977 and revised in 1992, sets requirements for: (1) maintenance of and compensatory measures for loss of security equipment; (2) testing frequency for intrusion alarms; (3) testing frequencies for communication equipment; and, (4) periodic review of the security program.

10 CFR 50.54(p)(3), established in 1986, sets requirements for: review and audit of safeguards contingency procedures and practices, an audit of the security system testing and maintenance program, and a test of the safeguards systems along with commitments established for response by local law enforcement authorities.

10 CFR 26.80, established in 1989, sets forth requirements for periodic audits of fitness-for-duty programs.

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10 CFR 73.56(g), established in 1991, sets requirements for periodic audits of personnel access authorization programs.

NUREG-0908, published in 1982, was developed to assist the NRC reviewers in their review of new or revised security plans by outlining specific criteria for the elements of such plans. Section 8 outlines criteria and guidelines for evaluation and audit of the physical security program. In the EPRI Evolutionary Plant FSER, the staff documented their use of NUREG-0908 in review of security-related design criteria.

Reg. Guide 5.44 provides guidance on frequencies and test methods for assuring operability of intrusion alarm systems. Reg. Guide 5.65 provides guidance for periodic review of security and contingency plans.

Consider revising Acceptance Criteria and developing Review Procedures for review of testing and maintenance programs and program reviews. These revisions should include requirements from 10 CFR 26.80, 10 CFR 50.54(p) and 10 CFR 73.56(g) and should reference applicable guidance from NUREG-0908 and Reg. Guides 5.44 and 5.65, as discussed above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11754/CODE OF FED. REGS 10CFR50; 11818/CODE OF FED. REGS 10CFR73; 12253/RG 5.44; 12360/RG 5.65; 24044/FINAL SER EPRI CH 9; 24057/NUREG 0908; 24125/CODE OF FED. REGS 10CFR73; 24152/CODE OF FED. REGS 10CFR26

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**Integrated Impact Number: 836      SRP Section Number: 13.6**

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for review of response capabilities and law enforcement liaisons.

SRP Section 13.6 (last revised in July 1981) lists Section 73.55(h) of 10 CFR 73 as specific acceptance criterion II.g. No specific Review Procedures are provided for this criterion.

10 CFR 73.55(h), established in 1977 and revised in 1978, 1979, and 1992, sets requirements for: (1) a safeguards contingency plan; (2) liaison with local law enforcement authorities; (3) number of onsite security personnel; (4) responses to abnormal presence or activity; (5) instructions to response personnel; and (6) capability of observing the perimeter of the protected area. (Safeguard contingency plan review is covered by Integrated Impact 837.)

Consider developing Review Procedures for review of response capabilities and law enforcement liaisons.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

11823/CODE OF FED. REGS 10CFR73

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**Integrated Impact Number:** 837      **SRP Section Number:** 13.6

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and develop Review Procedures for review of licensee safeguards contingency plans, including requirements from 10 CFR 50.54(p) and applicable guidance from Reg. Guide 5.54, Reg. Guide 5.65 and Generic Letter (GL) 89-07.

SRP Section 13.6 (last revised in July 1981) lists Section 73.55(h) of 10 CFR 73 and Appendix C to 10 CFR 73 as specific acceptance criteria II.g and II.i. No specific Review Procedures are provided for these criteria.

10 CFR 73.55(h)(1), established in 1977 and revised in 1992, sets requirements for establishment of, maintenance of and compliance with, a safeguards contingency plan, in accordance with the criteria of Appendix C to Part 73. Appendix C, established in 1978 and revised in 1992, provides criteria to be met by licensee's safeguards contingency plans.

10 CFR 50.54(p), established in 1986 and revised in 1987 and 1988, provides requirements for safeguards contingency plans and procedures.

Reg. Guide 5.54, issued in 1979, provides detailed guidance on the purpose, format and content of safeguards contingency plans.

Reg. Guide 5.65, Position C.5.1, states that the licensee should specify in the physical security or contingency plan the individual, by title, responsible for relaxing security requirements if necessary during emergencies. The plan should also specify a chain of responsibility for suspension of safeguards requirements in the event that the first designated individual is unavailable.

Generic Letter 89-07 stated that the Commission had concluded it would be prudent to have power reactor licensees include in their safeguards contingency plans short-term actions to protect against attempted radiological sabotage involving a land vehicle bomb if such a threat were to materialize. Supplement 1 to Generic Letter 89-07 provides clarification of the original letter. (Note: Final Rule on "Protection Against Malevolent Use of Vehicles at Nuclear Power Plants" was published in the Federal Register on August 1, 1994.)

Consider revising Acceptance Criteria and developing Review Procedures for review of licensee

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safeguards contingency plans. These revisions should include requirements from 10 CFR 50.54(p) and should reference applicable guidance from Reg. Guide 5.54, Reg. Guide 5.65 and Generic Letter (GL) 89-07 as discussed above.

In addition, consider future work to update Reg. Guide 5.54. This work will be tracked on IPD 7.0 Form 13.6-2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11753/CODE OF FED. REGS 10CFR50; 11822/CODE OF FED. REGS 10CFR73; 12352/RG 5.65; 12432/NRC GENERIC LETTER 89-07; 12433/NRC GENERIC LETTER 89-07 Sup 1; 22326/CODE OF FED. REGS 10CFR73; 22327/RG 5.54

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**Integrated Impact Number:** 838      **SRP Section Number:** 13.6

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of personnel access and fitness for duty programs, incorporating requirements from 10 CFR 26, 10 CFR 73.56 and 10 CFR 73.57 and including applicable guidance from Appendix A to 10 CFR 26 and Reg. Guide 5.66.

SRP Section 13.6 (last revised in July 1981) lists ANSI N18.17, Paragraph 4.3, "Employee Screening" as Acceptance Criterion II.5. No specific Review Procedures are provided for this criterion.

ANSI N18.17 was replaced by ANS 3.3. In the CE80+ FSER, the staff indicated that the applicant should reference 10 CFR 73.56 and Reg. Guide 5.66 in lieu of ANSI N18.17.

10 CFR 73.57, established in 1987 and revised in 1988, 1990, 1992 and 1994, sets forth requirements for criminal history checks of individuals granted unescorted access to a nuclear power facility or access to safeguards information.

10 CFR 73.56, established in 1991, sets requirements for establishment and implementation of a personnel access authorization program as part of the site Physical Security Plan. Reg. Guide 5.66, issued in 1991, provides guidance for meeting the requirements of 10 CFR 73.56. NUMARC 89-01, "Industry Guidelines for Nuclear Power Plant Access Authorization Programs," is provided as an Appendix to this Reg. Guide.

10 CFR 26, established in 1989 as the Fitness-for-Duty rule and revised in 1992 and 1993, prescribes requirements and standards for the establishment and maintenance of certain aspects of fitness-for-duty programs and procedures by the licensed nuclear power industry. Appendix

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A to Part 26 provides guidelines for drug and alcohol testing programs.

Consider developing Acceptance Criteria and Review Procedures for review of personnel access and fitness for duty programs. These revisions would incorporate requirements from 10 CFR 26, 10 CFR 73.56 and 10 CFR 73.57 and should reference applicable guidance from Appendix A to 10 CFR 26 and Reg. Guide 5.66, as discussed above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22064/FINAL SER CE80 CH 13; 22067/RG 5.66; 22328/NRC POLICY STATEMENT 53 FR 7534; 22442/NRC GENERIC LETTER 91-16; 24034/C&S: ANSI N18.17; 24124/CODE OF FED. REGS 10CFR73; 24126/CODE OF FED. REGS 10CFR73; 24151/CODE OF FED. REGS 10CFR26

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**Integrated Impact Number:** 839      **SRP Section Number:** 13.6

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of controls on safeguards information, incorporating requirements from 10 CFR 73.21 and referencing 10 CFR 50.34(e) and 10 CFR 50.54(v).

10 CFR 50.34(e) and 10 CFR 50.54(v) require protection of physical security plans, safeguards contingency plans, guard qualification and training plans as safeguards information in accordance with 10 CFR 73.21.

10 CFR 73.21, revised in 1981 and 1989, sets requirements for the protection of safeguards information (SI), including: material to be controlled as SI, access to SI, protection while in use or storage, preparation and marking of documents, reproduction and destruction of matter containing SI, external transmission of documents and material, use of automatic data processing systems, and removal from SI category.

Consider developing Acceptance Criteria and Review Procedures for review of controls on safeguards information, incorporating requirements from 10 CFR 73.21 and including reference to 10 CFR 50.34(e) and 10 CFR 50.54(v).

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11752/CODE OF FED. REGS 10CFR50; 24157/CODE OF FED. REGS 10CFR73

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**Integrated Impact Number:** 840      **SRP Section Number:** 13.6

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of sabotage vulnerabilities of evolutionary reactors.

Generic Issue A-29 was established to consider alternatives to the basic design of nuclear power plants with the emphasis primarily on reduction of the vulnerability of reactors to industrial sabotage. The issue was resolved with no new requirements for current reactors, based upon the success of 10 CFR 73.55.

In its "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," August 8, 1985, the Commission stated: "The issues of both insider and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized in the design and in the operating procedures developed for new plants."

In Appendix B to Chapter 1 of the EPRI Evolutionary Plant FSER, the staff discussed their conclusion that the evolutionary requirements document adequately addressed Issue A-29 with respect to external sabotage, but had not adequately addressed insider sabotage. To address this issue, the requirements document was revised to require plant designers to analyze the vulnerability of their designs to insider sabotage before finalizing their designs. In Chapter 6 of the EPRI Evolutionary Plant FSER, the staff describes their consideration of access controls as providing a measure of protection against insider sabotage.

In the CE80+ FSER, the staff discussed the results of the vendor's vulnerability analysis and stated their intent to review the site-specific vulnerability analysis and final design during its review of the Combined Operating License applicants' site-specific security, contingency, and guard training plans.

In the ABWR FSER, the staff discussed design features that provide a measure of protection against insider sabotage. The staff also confirmed that GE had established an action item for the applicants referencing the ABWR standard design to conduct a sabotage vulnerability analysis.

Consider developing Acceptance Criteria and Review Procedures for review of sabotage vulnerabilities of evolutionary reactors. Alternatively, consider future work to develop a new SRP Section for review of sabotage vulnerability analyses in connection with severe accident analyses. This action will be tracked by IPD-7.0 Form 13.6-3.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22062/FINAL SER EPRI CH 9; 24053/FINAL SER EPRI CH 6; 24158/NUREG 0933;  
24159/FINAL SER ABWR CH 20; 24161/FINAL SER CE80 CH 13; 24162/FINAL SER  
EPRI CH 1; 24173/NRC POLICY STATEMENT 50 FR 32138; 25428/NUREG 1267

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**Integrated Impact Number:** 841      **SRP Section Number:** 13.6

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria discussion regarding implementation of the physical security program prior to fuel loading.

SRP 13.6 Acceptance Criteria states that implementation of the physical security program should be accomplished 1 to 2 months before fuel loading.

In the ABWR FSER, the staff stated their position that they require that at least 60 days before loading fuel, the COL applicant confirm that security systems and programs described in its physical security plan, safeguards contingency plan, and guard qualification and training plan have achieved operational status and are available for NRC inspection. Operational status means that the security systems and programs are functioning in entirety as they would when the reactor is operating and will remain so.

Consider revising the Acceptance Criteria discussion regarding implementation of the physical security program prior to fuel loading to require implementation at least 60 days before fuel loading.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22330/FINAL SER ABWR CH 13

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**Integrated Impact Number:** 842      **SRP Section Number:** 15.7.4

#### **Suggested Changes to the SRP Section:**

Revise acceptance criteria and review procedures to incorporate the application of revised source term data, as presented in draft NUREG-1465.

In Section 15.4.2.6.1 of the System 80+ FSER, the staff used the assumptions of Regulatory Guide 1.25, as modified by revised source term considerations, in its review of the radiological consequences of a fuel handling accident. Specifically, the gap activity was taken from Table



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3.12 of draft NUREG-1465 (and listed in Table 15.A-1 of the FSER), and assumptions as to the chemical form of iodine and fission product release timing differ from that presented in Regulatory Guide 1.25.

Consider revising the acceptance criteria and review procedures to incorporate the changes in analysis assumptions presented in Appendix 15A of the System 80+ with respect to gap release fractions, iodine chemical form, and fission product release timing.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24333/FINAL SER CE80 CH 15

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**Integrated Impact Number:** 843      **SRP Section Number:** 3.9.1

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to assure that the appropriate number of natural convection cooldown events are included in the transients used in the design and fatigue analysis of Code Class 1 and CS components.

Generic Issue 79, "Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown," established to address the potential for unanalyzed reactor vessel thermal stresses that could occur during natural convection cooldown (NCC) of PWR reactors, was resolved with the publication of Generic Letter 92-02, "Resolution of Generic Issue 79, 'Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown'." The staff's evaluation to resolve this issue included extensive analyses of an actual NCC event that occurred at St. Lucie. As discussed in GL 92-02, no new requirements were established for operating reactors based, in part, on the expected evaluation that would take place following an actual NCC event. In the DSER for CE 80+, the staff requested the applicant to submit additional information regarding the applicability of the St. Lucie and System 80 analyses to the System 80+ design, and to verify that the number of natural convection cooldown events assumed in the analyses are applicable to a 60-year plant design life. The CE 80+ FSER documents the staff's review of the applicants' response. SRP Section 3.9.1 Areas of Review includes transients which are used in the design and fatigue analysis of all Code Class 1 and CS components, and supports and reactor internals.

Consideration should be given to modifying the Review Procedures to assure that the appropriate number of natural convection cooldown events are included in the transients used in design and fatigue analyses for evolutionary PWRs.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22820/DRAFT SER CE80 CH 20; 22831/NUREG 0933; 22832/NRC GENERIC LETTER 92-02; 24168/FINAL SER CE80 CH 20

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**Integrated Impact Number:** 844      **SRP Section Number:** 6.2.1.1.A

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to address inadvertent actuation of the post-accident inerting system.

10CFR50.34(f)(2)(ix) requires that certain specific applicants for CPs include in their design a hydrogen control system that can accommodate a 100% fuel-clad metal water reaction. 10CFR50.34(f)(2)(ix)(D) and 10CFR50.34(f)(3)(v)(B)(1) both require that if the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of this system must be accommodated during plant operation. 10CFR50.52, subsections 52.47(a)(1)(ii) and 52.79(a)(3)(b), require that design certification and combined license applicants meet "technically relevant" portions of 10CFR50.34. If an applicant's proposed design utilizes a post-accident inerting system, the 10CFR50.34 requirement concerning such systems should be applied.

Consider revising the Acceptance Criteria and Review Procedures subsections to discuss that the containment must be designed to accommodate inadvertent actuation of the hydrogen control post-accident inerting system if such a system is incorporated in the plant design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22990/CODE OF FED. REGS 10CFR50; 24472/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 845      **SRP Section Number:** 5.2.3

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to identify further acceptable alternatives to compliance with Regulatory Guide 1.50. SRP Section 5.2.3 Acceptance Criteria (specific criterion 3.b(1)) currently indicates that the preheat controls described in Westinghouse Topical Report WCAP-8577 are an acceptable alternate to compliance with those of Regulatory Guide 1.50.

In the CE System 80+ FSER, the staff indicated that the applicant took exception to the recommendations in Position C.2 of RG 1.50 for controls imposed on preheat temperatures for

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welding ferritic steels. These controls normally provide reasonable assurance that components made from low-alloy steels will not crack during fabrication and minimize the possibility of subsequent cracking from hydrogen in the weldment. The applicant indicated that its basis for taking exception to Position C.2 was Westinghouse Topical Report WCAP-8678, "Effect of Preheat and Post Weld Heat Treat on Hydrogen-Induced Cracking in Pressure Vessel Steels," September 1975. The staff evaluated and accepted this report. This report presents three acceptable alternatives, and the applicant's position is that a particular alternative will be specified based upon various factors such as the configuration or the capabilities of the fabrication facility. The staff concluded that this approach should provide adequate assurance that low-alloy steel weldments will not develop cracking due to hydrogen.

Consideration should be given to revising Acceptance Criteria to identify topical report WCAP-8678 as a further acceptable alternative to compliance with Regulatory Guide 1.50.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24343/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 846      **SRP Section Number:** 5.2.3

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to identify acceptable alternatives to compliance with Regulatory Guide 1.44 controls to verify non-sensitization of austenitic stainless steel materials and weldments. SRP Section 5.2.3 Acceptance Criteria (specific criterion 4.a) and Review Procedures (III.4.a(3)) currently indicate that the controls to verify non-sensitization of austenitic stainless steels are described/covered in Regulatory Guide 1.44.

In the CE System 80+ FSER, the staff indicated that the applicant proposes to allow the continued reference to American Society for Testing and Materials (ASTM) A-708 for the purpose of maintaining the qualifications of older weld procedures. ASTM A-262, Practice A or E are the methods explicitly recommended in RG 1.44, and have been used by the applicant since the mid-1970s for verifying non-sensitization of austenitic stainless steel materials and weldments. The applicant qualified weld procedures developed prior to the mid-1970s and used A-708 in their qualifications. These weld procedures are still in use. There have been no intergranular stress corrosion cracking (IGSCC) failures involving stainless steel weldments in the applicant's nuclear steam supply system (NSSS) units. In addition, the staff has allowed in SRP 4.5.1, paragraph III.2, ASTM A-708 as an acceptable alternative test for ASTM A-262, Practice A or E.

Consideration should be given to revising Acceptance Criteria and Review Procedures to

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identify acceptable alternatives to compliance with Regulatory Guide 1.44 controls to verify non-sensitization of austenitic stainless steel materials and weldments.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23449/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 847      **SRP Section Number:** 9.2.6

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to add General Design Criteria 60 as an acceptance criterion and include Regulatory Guide 1.143 as guidance.

General Design Criteria 60 requires that sufficient holdup capacity be provided for retention of liquid effluents containing radioactive materials. Regulatory Guide 1.143 contains staff positions related to radioactive waste management systems, structures and components. Certain positions in C.1.2 apply to tanks containing radioactive materials in liquids and address the condensate storage tanks specifically. These positions provide guidance in meeting the requirements of General Design Criteria 60.

Consider citing General Design Criteria 60 as acceptance criterion and Regulatory Guide 1.143 as guidance in the Acceptance Criteria and developing a Review Procedure that verifies that the condensate storage facility is consistent with that guidance.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

270/RG 1.143; 24347/FINAL SER CE80 CH 9; 24348/FINAL SER ABWR CH 11;  
24365/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 848      **SRP Section Number:** 15.7.5

#### **Suggested Changes to the SRP Section:**

Revise acceptance criteria and review procedures to incorporate the application of revised source term data, as presented in draft NUREG-1465.

In Section 15.4.2.6.1 of the System 80+ FSER, the staff used the assumptions of Regulatory Guide 1.25, as modified by revised source term considerations, in its review of the radiological consequences of a fuel handling accident. Specifically, the gap activity was taken from Table 3.12 of draft NUREG-1465 (and listed in Table 15.A-1 of the FSER), and assumptions as to the

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chemical form of iodine differ from that presented in Regulatory Guide 1.25. In Section 15.4.2.6.3 of the System 80+ FSER for a postulated spent fuel cask drop accident, the staff did not address radiological consequences because the System 80+ design precludes cask lifts in excess of 30 feet. It is reasonable to assume that if radiological consequences had been performed, the calculation would have used assumptions consistent with those used by the staff in fuel handling accident calculations.

Consider revising the acceptance criteria and review procedures to incorporate the changes in analysis assumptions presented in Appendix 15A of the System 80+ FSER with respect to gap release fractions and iodine chemical form, to replace the current citations to Regulatory Positions C.1.d, e, and f of RG 1.25.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24355/FINAL SER CE80 CH 15

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**Integrated Impact Number:** 849      **SRP Section Number:** 9.3.3

#### **Suggested Changes to the SRP Section:**

Add a discussion of drain requirements with regard to the removal of fire suppression water to SRP Section 9.3.3.

Subsection I, Areas of Review, assigns the review for fire protection to another branch for evaluation. However, certain issues identified in RG 1.120, Rev. 1, C.4.a.9 and C.6, "Guidelines for Specific Plant Areas" need to be specifically addressed in SRP 9.3.3. These issues deal with the drainage of fire suppression water from areas such as the cable spreading room, switchgear rooms, diesel generator areas, and safety-related pump rooms. Fire suppression water could cause flooding and damage to safety-related equipment.

Consider modifying SRP 9.3.3, subsection II.2 (GDC 4) of Acceptance Criteria, subsection III.3 of Review Procedures, and subsection IV.1 of Evaluation Findings to include a discussion of drain requirements with regard to the removal of fire suppression water. Plant Systems Branch staff should be consulted on whether the RG 1.120 citations should be replaced by BTP CMEB 9.5-1 citations.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

2546/RG 1.120

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**Integrated Impact Number:** 851      **SRP Section Number:** 6.2.1.1.B

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to address inadvertent actuation of the post-accident inerting system.

10CFR50.34(f)(2)(ix) requires that certain specific applicants for CPs include in their design a hydrogen control system that can accommodate a 100% fuel-clad metal water reaction. 10CFR50.34(f)(2)(ix)(D) and 10CFR50.34(f)(3)(v)(B)(1) both require that if the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of this system must be accommodated during plant operation. 10CFR50.52, subsections 52.47(a)(1)(ii) and 52.79(a)(3)(b), require that design certification and combined license applicants meet "technically relevant" portions of 10CFR50.34. If an applicant's proposed design utilizes a post-accident inerting system, the 10CFR50.34 requirement concerning such systems should be applied.

Consider revising the Acceptance Criteria and Review Procedures subsections to discuss that the containment must be designed to accommodate inadvertent actuation of the hydrogen control post-accident inerting system if such a system is incorporated in the plant design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22979/CODE OF FED. REGS 10CFR50; 24479/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 852      **SRP Section Number:** 5.2.3

#### **Suggested Changes to the SRP Section:**

Add Review Procedures for review of the acceptability of cast austenitic stainless steel materials and weldments.

In the CE System 80+ FSER, the staff indicated that the applicant initially proposed to fabricate limited portions of the RCPB using cast austenitic stainless steels. The staff indicated concerns relating to the thermal aging properties and the inspectability of such materials using ultrasonic techniques. In response to the staff's concerns, the applicant redesigned affected portions of the RCPB using wrought materials to the extent feasible. Where cast austenitic stainless steel was determined to be the best material selection for a specific application, the applicant applied additional controls involving restrictions on the maximum permissible ferrite content of the casting and the upper ferrite content limit for austenitic stainless steel weld metal to values more stringent than standard limits. The staff accepted the applicant's redesign efforts and added

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controls while indicating that the lower ferrite content limits for austenitic stainless steel castings and weld metal will provide reasonable assurance that components of these materials will maintain adequate fracture toughness for their 60-year life.

Consideration should be given to adding Review Procedures for review of the acceptability cast austenitic stainless steel materials based upon the above information.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23366/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 853      **SRP Section Number:** 6.2.1.1.C

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to address inadvertent actuation of the post-accident inerting system.

10CFR50.34(f)(2)(ix) requires that certain specific applicants for Construction Permits (CPs) include in their design a hydrogen control system that can accommodate a 100% fuel-clad metal water reaction. 10CFR50.34(f)(2)(ix)(D) and 10CFR50.34(f)(3)(v)(B)(1) both require that if the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of this system must be accommodated during plant operation. 10CFR50.52, subsections 52.47(a)(1)(ii) and 52.79(a)(3)(b), require that design certification and combined license applicants meet "technically relevant" portions of 10CFR50.34. If an applicant's proposed design utilizes a post-accident inerting system, the 10CFR50.34 requirement concerning such systems should be applied.

Consider revising the Acceptance Criteria and Review Procedures subsections to discuss that the containment must be designed to accommodate inadvertent actuation of the hydrogen control post-accident inerting system if such a system is incorporated in the plant design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22994/CODE OF FED. REGS 10CFR50; 24480/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 854      **SRP Section Number:** 9.3.2

#### **Suggested Changes to the SRP Section:**

Revise current Acceptance Criteria and associated Review Procedures related to TMI Item

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#### **III.D.1.1.**

Item III.D.1.1 addresses provisions for leakage control and detection in the design of systems outside containment that contain (or might contain) radioactive materials following an accident. This issue was evaluated under Task III.D.1.1 as described in NUREG-0660. This issue was identified as III.D.1.1 and was approved for implementation in NUREG-0737 as an item applicable to all operating license applicants. A license application requirement related to this issue was incorporated in the CFR as 10 CFR 50.34(f)(2)(xxvi). Based upon a review of the ABWR design, documented in the ABWR FSER, the staff found that the leakage control program for the systems outside the containment for the ABWR design would include periodic leak testing and leak reduction measures for the listed systems including the post-accident sampling systems. The staff concluded that the ABWR design adequately addresses the requirements of this TMI item.

Consideration should be given to updating the Acceptance Criteria and other related portions of SRP Section 9.3.2 to reflect the 10 CFR 50.34(f)(2)(xxvi) requirement and the positions indicated in NUREG-0737 item III.D.1.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24373/CODE OF FED. REGS 10CFR50; 24374/FINAL SER ABWR CH 20; 24375/NUREG 0737

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**Integrated Impact Number: 855      SRP Section Number: 8.3.1**

#### **Suggested Changes to the SRP Section:**

Revise SRP Section 8.3.1 to properly reflect the latest revision of Regulatory Guide 1.9.

The staff issued Revision 3 of RG 1.9, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants," in July 1993. The RG incorporates guidance that was previously included in RG 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems At Nuclear Power Plants." RG 1.108 was subsequently withdrawn August 5, 1993. In addition, RG 1.9 endorses IEEE 387-1984, which is the current version of the IEEE standard. Revision 3 of the RG contains more technical information and has different paragraph numbers than the previous revision. Therefore, some references in SRP Section 8.3.1 to RG 1.9 are not correct, and some additional material should be referenced.

The resolution of Generic Issue B-56, "Diesel Reliability," included the issuance of Regulatory Guide 1.160 as well as Revision 3 to RG 1.9. The guidance relevant to monitoring EDG



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reliability in RG 1.160 in response to the Maintenance Rule, 10 CFR 50.65, also needs to be addressed in SRP Section 8.3.1.

Consider revising SRP Section 8.3.1, Subsection II, Acceptance Criteria, and Subsection III, Review Procedures, to properly reflect the changes in Revision 3 of RG 1.9 , and to incorporate applicable guidance of RG 1.160.

(Note: RG 1.108 is addressed in ROC 859)

#### **Potential Impacts/Documents supporting the Suggested Changes:**

9948/RG 1.9; 26020/RG 1.160

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**Integrated Impact Number: 859      SRP Section Number: 8.3.1**

#### **Suggested Changes to the SRP Section:**

Delete RG 1.108 from Section 8.3.1.

RG 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems At Nuclear Power Plants," was withdrawn August 5, 1993. The guidance that was contained in RG 1.108 was updated and incorporated into Revision 3 of RG 1.9, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants."

Consider deleting references to RG 1.108 from SRP Section 8.3.1. This should be performed in conjunction with incorporating revisions to RG 1.9.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

10214/RG 1.108

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**Integrated Impact Number: 862      SRP Section Number: 8.3.1**

#### **Suggested Changes to the SRP Section:**

Add references to RG 1.153 to SRP Section 8.3.1.

This Integrated Impact also recommends further research/regulatory action per IPD-7.0 form number 8.3.1-1.

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The staff published RG 1.153, "Criteria for Power, Instrumentation, and Control," in December 1985. The RG endorses, with some modification and supplements, IEEE Std 603-1980 as a method acceptable to the NRC staff for complying with Commission's regulations with regard to the design, reliability, qualification, and testability of the power (including electric, pneumatic, and hydraulic power), instrumentation, and control portions of safety systems. The RG identifies General Design Criteria 2, 4, 10, 12, 13, 15, 17, 18, 20, 21, 22, 23, 24, 25, 29, 34, 37, and 54 as being applicable to the power, instrumentation, and control portions of safety systems. The RG is not referenced in SRP Chapter 8.

RG 1.153 endorses IEEE 603-1980, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," rather than the current version which is IEEE 603-1991.

Consider adding references to RG 1.153 to SRP Section 8.3.1, Subsection II, Acceptance Criteria, and Subsection III, Review Procedures.

A detailed comparison between the 1980 and 1991 versions of IEEE Std 603 would be needed to support a change in RG 1.153 to endorse the current version. An IPD 7.0 form 8.3.1-1 has been initiated.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

10219/RG 1.153

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**Integrated Impact Number: 863      SRP Section Number: 8.3.1**

#### **Suggested Changes to the SRP Section:**

Add references to RG 1.155 to SRP Section 8.3.1.

10 CFR 50.63, " Loss of All Alternating Current Power," was issued in June 1988 and addresses the subject of station blackout. The NRC staff subsequently published RG 1.155 to provide guidance for complying with 10 CFR 50.63. The RG, in relevant part, establishes target reliability levels for emergency ac electric power sources, and guidance for a reliability program for such sources.

Consider adding references to RG 1.155 to SRP Section 8.3.1, Subsection II, Acceptance Criteria, and Subsection III, Review Procedures.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

10220/RG 1.155

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**Integrated Impact Number:** 864      **SRP Section Number:** 8.3.1

#### **Suggested Changes to the SRP Section:**

Add references to 10 CFR 50.63 to SRP Section 8.3.1.

NRC published 10 CFR 50.63, "Loss of all alternating current power," in June 1988. The station blackout rule, 10 CFR 50.63, requires each light-water- cooled nuclear power plant licensed to operate to be able to withstand for a specified duration, and recover, from a station blackout as defined in 10 CFR 50.2. The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The rule requires in part that emergency onsite ac source reliability be addressed to limit the potential for a station blackout. SRP Chapter 8 does not include 10 CFR 50.63 as acceptance criteria.

Consider adding references to 10 CFR 50.63 to SRP Section 8.3.1 in Areas of Review, Acceptance Criteria, Review Procedures and Evaluation Findings.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

9825/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 866      **SRP Section Number:** 8.3.1

#### **Suggested Changes to the SRP Section:**

Revise SRP Section 8.3.1 to address approved NRC policies stated in SECY-91-078.

SECY-91-078, NUREG-1242, and the EPRI Evolutionary Plant FSER Section 11, provide the NRC staff evaluation of the EPRI Evolutionary Plant Utility Requirements Document as it relates to the Electric Power Systems. Additional guidance not presently in SRP Section 8.3.1 is provided regarding NRC policy regarding alternate power source for non-safety loads and connection of offsite circuits directly to safety buses (SECY-91-078).

NUREG-1242, subsection 4.2.1, discusses a policy stated in SECY-91-078 that evolutionary plant design should include an alternate power source to non- safety loads and at least one offsite

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circuit to each redundant safety division that is supplied directly from one of the offsite power sources with no intervening non-safety buses. Applicable regulations for these two items are included in the System 80+ FSER and ABWR FSER.

Consider revising SRP Section 8.3.1, subsections II and III to present the NRC policy and supporting bases concerning an alternate power source for non-safety loads and direct offsite circuits to safety buses as stated in SECY-91-078.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24376/FINAL SER CE80 CH 8; 24377/FINAL SER ABWR CH 8; 24378/NUREG 1242

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**Integrated Impact Number:** 872      **SRP Section Number:** 8.3.2

#### **Suggested Changes to the SRP Section:**

Update RG 1.128 to address the current version of IEEE Std 484.

RG 1.128, Rev 1, October 1978, "Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants," is listed in Table 8-1 of SRP Chapter 8.0 as being applicable to SRP Section 8.3.2. The SRP section makes no mention of RG 1.128 or of IEEE Std 484-1975, "IEEE Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations" which is endorsed by the Reg Guide.

The implementation of IEEE Std 484-1975 could have a definite effect on the overall design of the d-c power system in the area of instrumentation and alarms and, in which case would constitute new design requirements.

Consider modifying the sections of ACCEPTANCE CRITERIA, REVIEW PROCEDURES and EVALUATION FINDINGS in SRP 8.3.2 to incorporate the guidance contained in IEEE Std 484-1975 as modified by Section C, "Regulatory Position," of RG 1.128.

Consider modifying the REFERENCE section of SRP 8.3.2 to include reference to RG 1.128.

A Research/Regulatory Action Needs Form (IPD 7.0), 8.3.2-5 has been initiated to address updating RG 1.128 to endorse the latest version of IEEE 484

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

12134/RG 1.128

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**Integrated Impact Number:** 874      **SRP Section Number:** 8.3.2

#### **Suggested Changes to the SRP Section:**

Add acceptance criteria and review procedures for station blackout rule compliance review.

10 CFR 50.63 addresses postulated station blackout events. 10 CFR 50.63 (a) (2) requires that the reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. Regulatory Guide 1.155, Position C.3, provides guidance for complying with 10 CFR 50.63 requirements.

Consider adding 10 CFR 50.63 and RG 1.155 provisions in the acceptance criteria and review procedure subsections.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11868/CODE OF FED. REGS 10CFR50; 12103/RG 1.155

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**Integrated Impact Number:** 875      **SRP Section Number:** 8.3.2

#### **Suggested Changes to the SRP Section:**

Incorporate the results from resolution of Generic Issue A-30, "Adequacy of Safety-Related DC Power Supplies."

Generic Letter 91-06 was issued to resolve Generic Issue A-30. The generic letter requested licensees to respond to 9 questions that were developed to facilitate staff determination of licensee implementation of existing recommendations. Many of these recommendations appear to be already incorporated in the Standard Review Plan or in standard technical specifications. A detailed evaluation should be performed as part of the detailed analyses/draft revision task to identify those portions which need to be added to SRP Section 8.3.2. For these portions, the detailed information presented in NUREG/CR-5414 should be considered for incorporation or reference. It is noted that IEEE 946 has been issued which supports many of the same recommendations contained in NUREG/CR-5414. A Research/Regulatory Action Needs Form (IPD 7.0) 8.3.2-6 has been initiated for potential endorsement of IEEE 946.

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In addition, the resolution of A-30 referenced the staff positions presented in NRC Bulletin 79-27. The bulletin required an evaluation of buses supplying power to instrumentation and control systems. The evaluation included a review of the alarms and indications for loss of power to the buses and a review of emergency procedures to achieve a cold shutdown condition upon loss of power to the buses. These staff position should be added to SRP Section 8.3.2, with a review interface to SRP Section 7.5, "Information Systems Important to Safety."

Consider including staff positions reflected in the resolution of Generic Issue A-30, and at a minimum, incorporate the provisions of Bulletin 79-27 into review procedures.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

12258/NRC BULLETIN 79-27; 12293/NRC GENERIC LETTER 91-06; 24394/NUREG CR-5414; 24395/C&S: IEEE 946-85

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**Integrated Impact Number:** 876      **SRP Section Number:** 8.3.2

#### **Suggested Changes to the SRP Section:**

Add a review procedure related to single-failure protection review for loss of an ac or dc bus.

In the EPRI Evolutionary Plant FSER, the staff stated that in the review of individual applications for FDA/DC, the staff will require that the plant designers perform an analysis of their ac and dc distribution systems to ensure that loss of any ac or dc bus does not result in a plant transient and simultaneous loss of single-failure protection in any safety-related system. The discussion indicated that this was related to concerns raised with respect to Generic Issue A-30 on dc power supply adequacy.

In the System 80+ FSER, the staff asked the vendor to perform a failure modes and effects analysis to ensure that failure of any ac or dc bus will not result in a plant transient and simultaneously cause the loss of single-failure protection in any safety-related system.

Consider adding a review procedure related to loss of an ac or dc bus.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24396/FINAL SER EPRI CH 11; 24397/FINAL SER CE80 CH 8

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**Integrated Impact Number:** 877      **SRP Section Number:** 8.3.2

#### **Suggested Changes to the SRP Section:**

Incorporate RG 1.153, Rev 0, December 1985 into SRP 8.3.2.

RG 1.153, Rev 0, December 1985, "Criteria for Power, Instrumentation, and Control Portions of Safety Systems," endorses IEEE Std 603-1980, "Criteria for Safety Systems for Nuclear Power Generating Stations." IEEE Std 603 incorporates the requirements and recommendations for both protection systems and safety-related systems, whereas IEEE Std 279-1971 is limited to protection systems only. Compliance with the provisions of IEEE Std 603 is considered to satisfy the provisions of IEEE Std 279-1971.

Consider modifying AREAS OF REVIEW to incorporate the "Safety System Criteria" of IEEE Std 603-1980 into the review areas of the primary and coordinating review branches.

Consider modifying the sections of ACCEPTANCE CRITERIA, REVIEW PROCEDURES and EVALUATION FINDINGS in SRP 8.3.2 to incorporate the guidance contained in IEEE Std 603-1980.

Consider modifying the REFERENCE section of SRP 8.3.2 to include RG 1.153.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

12101/RG 1.153

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**Integrated Impact Number:** 878      **SRP Section Number:** 5.4.6

#### **Suggested Changes to the SRP Section:**

Modify existing Review Procedures associated with TMI Task Action Plan Item II.K.3.15 involving spurious isolation of RCIC for BWR plants to incorporate additional guidance.

The RCIC system uses differential pressure sensors on elbow taps in the steam lines to their turbine drives to detect and isolate pipe breaks in the systems. TMI item II.K.3.15 requested that the applicant or licensee modify the pipe-break-detection circuitry so that pressure spikes resulting from RCIC system initiation will not cause inadvertent system isolation. Generic Letter 83-02 was subsequently issued that provides staff guidance for plants using a time delay relay in the pipe-break-detection circuitry to address this issue. The letter describes the minimum and maximum expected response times for this time delay function.

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The ABWR FSER subsections 5.4.6 and 20.4.64 discuss a bypass line that has been added to the steam supply line around the RCIC turbine steam inlet valve in the ABWR RCIC system design. This bypass is included in the RCIC system to improve system reliability, provide a smoother turbine start and reduce the possibility of an overspeed trip. Consequently, the ABWR does not incorporate a time delay relay in the pipe-break-detection circuitry to address the issue of spurious RCIC isolation. The staff indicated in the ABWR FSER that this approach to meeting the intent of II.K.3.15 was acceptable but requested that the RCIC bypass startup system be tested during the plant startup to ensure RCIC initiation will not cause an inadvertent isolation.

The current SRP Review Procedures address TMI item II.K.3.15 and cites NUREG-0737 as guidance. The SRP does not currently identify the information provided in Generic Letter 83-02 or approaches to compliance with TMI item II.K.3.15 that do not involve modification of the pipe-break-detection circuitry.

Consideration should be given to augmenting the current Review Procedure for TMI Task Action Plan Item II.K.3.15 to address the additional guidance provided in Generic Letter 83-02. For system designs addressing this issue by other means, consideration should also be given to providing a review of the proposed design and verification that a system test program will confirm that the proposed approach satisfies the intent of II.K.3.15 in preventing spurious isolation of the RCIC system.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

5748/NRC GENERIC LETTER 83-02; 24353/FINAL SER ABWR CH 20; 24354/FINAL SER ABWR CH 5

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**Integrated Impact Number:** 879      **SRP Section Number:** 5.4.6

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure to review the leak reduction, detection, and measurement program as it applies to the RCIC system.

The RCIC system is specifically addressed in NUREG-0737, Action Plan Item III.D.1.1 as one of the systems required to be included in a leakage reduction program. This program is to include leak reduction measures on systems that could carry radioactive fluid outside the containment, measurement of actual leakage rates and a preventive maintenance program to include periodic integrated leak testing. TMI item III.D.1.1 is listed in specific criteria II.7 of this SRP Section, but the current Review Procedures do not include review of the implementation of the leakage reduction program in relation to the RCIC system.



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Consideration should be given to adding a new Review Procedure to address review of the leak reduction program and that portion of the design relate to implementation of that program.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24356/FINAL SER ABWR CH 20; 24358/NUREG 0737

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**Integrated Impact Number: 880      SRP Section Number: 5.4.6**

#### **Suggested Changes to the SRP Section:**

Add a Review Interface and a Review Procedure to verify proper design of the suppression pool suction strainers with respect to RCIC system operation.

The RCIC system secondary water supply is from the suppression pool after the water in the condensate storage tank is depleted. The current Review Procedures do not include verification of the suction strainer design. SRP Section 6.2.2 provides review guidance for suction strainer design with respect to containment heat removal and the ECCS system.

In the ABWR FSER the staff found the design of the RCIC suction strainers to be acceptable. Their design is sized according to Regulatory Guide 1.82, but with conservatism in the mass of debris deposited on the strainers.

Consideration should be given to adding a Review Interface and a new Review Procedure to address verification of the design of the suppression pool suction strainers in SRP Section 6.2.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24351/FINAL SER ABWR CH 6; 24424/RG 1.82

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**Integrated Impact Number: 881      SRP Section Number: 5.4.1.1**

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures list of materials for fabrication of reactor coolant pump flywheels to include an additional grade of material.

Material selected for use in the fabrication of a flywheel is to comply with specified properties and production processes. SRP Section 5.4.1.1, Review Procedures lists the following materials previously evaluated as suitable for pump flywheels: ASME SA-533-B, Class 1; SA-508, Class 2; and SA-516, Grade 65.

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In Section 5.4.1 of the ABB-CE 80+ FSER, the staff states it believes SA-508, Class 3 should provide adequate properties to ensure a 60-year life of the reactor coolant pump flywheel. The FSER outlines the similarities in processing and properties of SA-508, Class 2, presently listed in the SRP, and SA-508, Class 3.

Regulatory Guide 1.14 lists examples of material that past evaluations have shown to be suitable for flywheel application, including SA-508, Class 3.

Consideration should be given to revising Review Procedure III.1 by adding SA-508, Class 3 material to the list of suitable materials.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

14902/RG 1.14; 24432/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 884      **SRP Section Number:** 6.2.1.1.C

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures for containment capability to withstand hydrogen burn or pressurization from the post-accident inerting system.

10 CFR 50.34(f)(3)(v)(A)(1) states that containment integrity will be maintained by meeting applicable ASME code requirements during an accident that releases hydrogen generated from a 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. 10 CFR 52, subsections 52.47(a)(1)(ii) and 52.79(a)(3)(b), require that design certification and combined license applicants meet "technically relevant" portions of 10 CFR 50.34. The requirement to design containment to withstand hydrogen burning or initiation of the post-accident inerting system appears to be technically relevant.

Consider revising the Acceptance Criteria and Review Procedures sections to discuss that containment must be designed to withstand hydrogen burning or actuation of the post-accident inerting system, if installed, during an accident that releases hydrogen.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24459/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 886      **SRP Section Number:** 6.2.1.1.A

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures for containment capability to withstand hydrogen burn or pressurization from the post-accident inerting system.

10 CFR 50.34(f)(3)(v)(A)(1) states that containment integrity will be maintained by meeting applicable ASME code requirements during an accident that releases hydrogen generated from a 100 % fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. 10 CFR 52, subsections 52.47(a)(1)(ii) and 52.79(a)(3)(b), require that design certification and combined license applicants meet "technically relevant" portions of 10 CFR 50.34. The requirement to design containment to withstand hydrogen burning or initiation of the post-accident inerting system appears to be technically relevant.

Consider revising the Acceptance Criteria and Review Procedures sections to discuss that containment must be designed to withstand hydrogen burning or actuation of the post-accident inerting system, if installed, during an accident that releases hydrogen.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24462/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 887      **SRP Section Number:** 6.2.1.1.B

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures for containment capability to withstand hydrogen burn or pressure from the post-accident inerting system.

10 CFR 50.34(f)(3)(v)(A)(1) states that containment integrity will be maintained by meeting applicable ASME code requirements during an accident that releases hydrogen generated from a 100 % fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. 10 CFR 52, subsections 52.47(a)(1)(ii) and 52.79(a)(3)(b), require that design certification and combined license applicants meet "technically relevant" portions of 10 CFR 50.34. The requirement to design containment to withstand hydrogen burning or initiation of the post-accident inerting system appears to be technically relevant.

Consider revising the Acceptance Criteria and Review Procedures sections to discuss that

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containment must be designed to withstand hydrogen burning or actuation of the post-accident inerting system, if installed, during an accident that releases hydrogen.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24463/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number: 888      SRP Section Number: 9.5.1**

#### **Suggested Changes to the SRP Section:**

ASTM E84-1976 is cited in Branch Technical Position (BTP) CMEB 9.5-1.

The latest version of this standard is ASTM E84-1994, "Standard Test Method for Surface Burning Characteristics of Building Materials."

Consideration should be given to performing a detailed side-by-side comparison to allow SRP reviewers to use the latest version of the standard.

#### **INSPECTION PROGRAM BRANCH COMMENT:**

No comparison needed. Per 11/94 conversation with Plant Systems Branch staff, the new version of this fire standard can be cited without the performance of a side by side comparison.

#### **PNL COMMENT:**

Part A has been revised to identify the 1994 version as the latest standard.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23904/C&S: ASTM E84

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**Integrated Impact Number: 890      SRP Section Number: 17.1**

#### **Suggested Changes to the SRP Section:**

Update SRP Section 17.1 by removing RG 1.88 as a review criterion.

RG 1.28, Rev. 3, August 1985, developed to endorse ANSI/ASME NQA-1-1983, provides guidance acceptable to the NRC staff for complying with the provisions of Appendix B with regard to establishing and implementing the requisite quality assurance program for the design

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and construction of nuclear power plants.

RG 1.88 was withdrawn in a June 17, 1991, letter by E. Beckjord. The ANSI Standard endorsed by RG 1.88 was incorporated into NQA-1.

Consider replacing RG 1.88 with RG 1.28 as a review criterion in SRP 17.1, 17.4, and 17.5 acceptance criteria and in the References Subsection.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18509/RG 1.88

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**Integrated Impact Number:** 891      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Update SRP Section 17.1 by removing RG 1.64 as a review criterion.

RG 1.28, Rev. 3, August 1985, developed to endorse ANSI/ASME NQA-1-1983, provides guidance acceptable to the NRC staff for complying with the provisions of Appendix B with regard to establishing and implementing the requisite quality assurance program for the design and construction of nuclear power plants.

RG 1.64 was withdrawn by a June 17, 1991, letter by E. Beckjord. The ANSI Standard endorsed by RG 1.64 was incorporated into NQA-1.

Consider replacing RG 1.64 with RG 1.28 as a review criterion in SRP Section 17.1 for acceptance criterion 3F2 and in the Reference subsection.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18518/RG 1.64

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**Integrated Impact Number:** 892      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Update SRP Section 17.1 by removing RG 1.74 as a review criterion.

RG 1.28, Rev. 3, August 1985 developed to endorse ANSI/ASME NQA-1-1983, provides guidance acceptable to the NRC staff for complying with the provisions of Appendix B with

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regard to establishing and implementing the requisite quality assurance program for the design and construction of nuclear power plants.

RG 1.74 was withdrawn in a September 1, 1989 letter by E. Beckjord. The ANSI Standard endorsed by RG 1.74 was incorporated into NQA-1.

Consider replacing RG 1.74 with RG 1.28 in the References subsection.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18569/RG 1.74

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**Integrated Impact Number:** 893      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Update SRP Section 17.1 by removing RG 1.146 as a review criterion.

RG 1.28, Rev. 3, August, 1985, developed to endorse ANSI/ASME NQA-1-1983, provides guidance acceptable to the NRC staff for complying with the provisions of Appendix B with regard to establishing and implementing the requisite quality assurance program for the design and construction of nuclear power plants.

RG 1.146 was withdrawn in a June 17, 1991 letter by E. Beckjord. The ANSI Standard endorsed by RG 1.146 was incorporated into NQA-1.

Consider replacing RG 1.146 with RG 1.28 for acceptance criterion 18B3 and in the References subsection.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18570/RG 1.146

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**Integrated Impact Number:** 894      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Update SRP Section 17.1 by removing RG 1.144 as a review criterion.

RG 1.28, Rev. 3, August 1985, developed to endorse ANSI/ASME NQA-1-1983, provides guidance acceptable to the NRC staff for complying with the provisions of Appendix B with

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regard to establishing and implementing the requisite quality assurance program for the design and construction of nuclear power plants.

RG 1.144 was withdrawn in a June 17, 1991 letter by E. Beckjord. The ANSI Standard endorsed by RG 1.144 was incorporated into NQA-1.

Consider replacing RG 1.144 with RG 1.28 for acceptance criterion 18B3 and in the References subsection.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18571/RG 1.144

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**Integrated Impact Number:** 896      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Update SRP Section 17.1 by removing RG 1.123 as a review criterion.

RG 1.28, Rev. 3, August 1985, developed to endorse ANSI/ASME NQA-1-1983, provides guidance acceptable to the NRC staff for complying with the provisions of Appendix B with regard to establishing and implementing the requisite quality assurance program for the design and construction of nuclear power plants.

RG 1.123 was withdrawn in a June 17, 1991 letter by E. Beckjord. The ANSI Standard endorsed by RG 1.123 was incorporated into NQA-1.

Consider replacing RG 1.123 with RG 1.28 for acceptance criterion 4B2 and in the References subsection.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18577/RG 1.123

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**Integrated Impact Number:** 897      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Update SRP Section 17.1 to incorporate RG 1.28 as a primary review criterion.

RG 1.28, Rev 3, August 1985 developed to endorse ANSI/ASME NQA-1-1983, and the

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ANSI/ASME NQA-1a-1983 Addenda, provides guidance acceptable to the NRC staff for complying with the provisions of Appendix B with regard to establishing and implementing the requisite quality assurance program for the design and construction of nuclear power plants.

The above referenced Potential Impacts address the regulatory positions and the implementation schedule of RG 1.28. The positions and the schedule will have an effect on the review process of SRP 17.1 as they contain the additions and modifications to the ANSI/ASME standards as prescribed by the staff.

Consider revising SRP Section 17.1 to incorporate RG 1.28 directly into the main body of Section 17.1 as a review criterion.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18578/RG 1.28; 18579/RG 1.28; 18580/RG 1.28; 18581/RG 1.28; 18582/RG 1.28;  
18583/RG 1.28; 18584/RG 1.28

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**Integrated Impact Number:** 898      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Update SRP Section 17.1 by removing RG 1.58 as a review criterion.

RG 1.28, Rev 3, August 1985 developed to endorse ANSI/ASME NQA-1-1983, provides guidance acceptable to the NRC staff for complying with the provisions of Appendix B with regard to establishing and implementing the requisite quality assurance program for the design and construction of nuclear power plants.

RG 1.58 was withdrawn in a June 17, 1991 letter by E. Beckjord. The ANSI standard endorsed by RG 1.58 was incorporated into NQA-1.

Consider replacing RG 1.58 with RG 1.28 as a review criterion in SRP 17.1 for acceptance criterion 2D and in the References subsection.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18587/RG 1.58

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**Integrated Impact Number:** 901      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Update SRP Section 17.1 to address the unique requirements of optical disk storage systems.

Appendix B of 10 CFR 50, under criterion XVII, "Quality Assurance Records," establishes requirements for a record keeping system. The purpose of Generic Letter 88-18 is to inform addressees that the staff approves the use of optical disk document imaging systems for the storage and retrieval of record copies of quality assurance records when appropriate quality assurance controls are applied.

The generic letter lists appropriate quality controls for an optical disk document imaging system which could have an effect on the QA program review criteria of SRP Section 17.1.

Consider revising SRP Section 17.1 to address the unique requirements of optical disk storage systems and the provisions of Generic Letter 88-18.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18610/NRC GENERIC LETTER 88-18

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**Integrated Impact Number:** 902      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Consider specifying the current version of ANSI/ANS 3.1 in SRP 17.1 and revise acceptance criterion 1C2.

RG 1.8 endorses ANSI/ANS 3.1-1981, "Selection, Qualification and Training of Personnel for Nuclear Power Plants" for selected plant personnel. This ANSI standard now has a 1993 revision date and the revised standard could have provisions which may affect the review criteria of SRP Section 17.1.

A comparison between the 1981 and the 1993 versions of the standard would be needed to support endorsement of the current version in SRP 17.1.

IPD 7.0 form (17.3-2) has been initiated for the future revision of RG 1.8 to incorporate the current version of ANSI/ANS 3.1.

In any case, SRP Section 17.1 Acceptance Criterion 1C2 needs to be revised to indicate that the

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current revision of RG 1.8 endorses ANSI N18.1-1971 for the QA Manager position.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

19809/RG 1.8

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**Integrated Impact Number:** 903      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Regulatory Guide 1.155, Regulatory Position C.3.5, provides quality assurance guidance for station blackout equipment. Existing quality assurance requirements of Appendix B and Appendix R of Part 50 continue to apply. For nonsafety-related equipment that are used to meet the requirements of 10 CFR 50.63 and that are not already covered by existing QA requirements in Appendix B or R, the guidance provided in Appendices A and B of RG 1.155 apply as specified in position C.3.5.

Consider adding acceptance criteria (for item 2A1 on the scope of the QA program) on the quality assurance guidance for station blackout equipment.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18505/RG 1.155

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**Integrated Impact Number:** 904      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Add acceptance criteria for quality assurance guidance for radwaste management systems.

Regulatory Guide 1.143, Regulatory Position C.6, provides quality assurance guidance for radwaste management systems. The specified quality assurance program is limited based on the limited impact of these systems on safety, and ensures that all design, construction, and testing provisions are met to ensure that these systems will perform their intended functions.

Consider adding acceptance criteria (for item 2A1 on the scope of the QA program) on the quality assurance guidance for radwaste management systems.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

18507/RG 1.143

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**Integrated Impact Number:** 905      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Add acceptance criteria for quality assurance guidance for accident monitoring instrumentation.

Regulatory Guide 1.97 in Table 1 provides design and qualification criteria for instrumentation to assess plant and environs conditions during and following an accident. Paragraph 5 of Table 1 specifies the quality assurance guidance for this type of instrumentation.

Consider adding acceptance criteria (for item 2A1 on the scope of the QA program) on the quality assurance guidance for accident monitoring instrumentation.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18510/RG 1.97

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**Integrated Impact Number:** 906      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Augment acceptance criteria for the dedication of commercial grade equipment for use in safety-related applications.

SRP Section 17.1 Acceptance Criterion 7B4 deals with commercial items. Generic Letter 91-05 provides staff positions with respect to certain aspects of licensee commercial grade procurement and dedication programs. For example, staff positions for critical characteristics and like-for-like replacements are discussed in the generic letter and its enclosure.

In addition, GL 91-05 clarified certain portions of Generic Letter 89-02 which discussed elements of an effective procurement and dedication program to improve the detection of counterfeit or fraudulently marketed products. Generic Letter 89-02 contains additional staff positions, and addresses other aspects of Appendix B, Criterion VII beyond Acceptance Criterion 7B4.

Consider augmenting acceptance criteria with applicable provisions of Generic Letters 91-05 and 89-02.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

18608/NRC GENERIC LETTER 91-05; 18609/NRC GENERIC LETTER 89-02

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**Integrated Impact Number:** 907      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Add acceptance criteria for quality assurance guidance for ATWS equipment.

Generic Letter 85-06 provides quality assurance guidance for non-safety related equipment encompassed by the ATWS rule, 10 CFR 50.62.

In the FSER for the System 80+ design, the staff states in Section 17.1.3, paragraph 14, that a graded approach that bases the QA requirements on the specific functions and their importance to safety should be applied to items specified in Section 3.3.1.4 of ANSI/ANS-51.1 (including the safety parameter display system or its equivalent), fire protection, non-safety-related ATWS items specified in 10 CFR 50.62, and non-safety-related items specified in 10 CFR 50.65.

In the FSER for the EPRI Utility Requirements Document for Evolutionary Plants, the staff states in Chapter 1, Section 7, "Quality Assurance" paragraph, that pertinent quality assurance provisions should be applied to activities and items that have some importance to safety even though they are not safety related (for example, see the new maintenance rule, 10 CFR 50.65(b)(2)); and that the staff will assure that applicants for FDA/DC or a combined license have acceptably described a quality assurance program for these activities and items.

In Section 17.1.3, paragraph 15, of the System 80+ FSER, the staff states that the quality assurance guidelines given in GL 85-06 should be followed.

Consider adding acceptance criteria (for item 2A1 on the scope of the QA program) on the quality assurance guidance for non-safety related ATWS equipment.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18611/NRC GENERIC LETTER 85-06

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**Integrated Impact Number:** 908      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

1. Provide discussion on the relationship between Appendix A, Criterion 1 and Appendix B

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requirements.

Several documents reflect certain aspects of the relationship between GDC 1 and Appendix B.

In Generic Letter 84-01 which dealt with the NRC use of the terms "important to safety" and "safety related," the staff states that applicants are responsible for developing and implementing quality assurance programs for plant design and construction or for plant operation which meet the more general requirements of GDC 1 for plant equipment "important to safety," and the, more prescriptive requirements of Appendix B for "safety-related" plant equipment.

2. This staff position should be incorporated into the current discussion of SRP Section 17.1, Acceptance Criterion 2B3. In addition, review if the Commission's Memorandum and Order, CLI-84-9, dated June 1984, is needed as it provides further clarification to GL 84-01.

3. Any applicable NRC positions in the review of CLI-84-9 should be incorporated into SRP Section 17.1.

In the FSER for the EPRI Utility Requirements Document for Evolutionary Plants, the staff states in Chapter 1, Section 9, that pertinent QA provisions should be applied to software, facilities, structures, systems, and components that have some safety importance or have some importance to safety even though they are not safety related; and that this is in accordance with the requirements of GDC 1 of Appendix A to 10 CFR Part 50; and that the staff will assure that the applications for FDA/DC or a combined license have acceptably described a QA Program for these activities and items, in the staff's review of individual applications for FDA/DC or a combined license.

In addition, a footnote in SRP Section 17.1, page 17.1-8 refers to rulemaking related to equipment "important to safety." The NUREG-0933 discussion for TMI Action Plan item I.f.1 on an expanded QA list appears to indicate that rulemaking to expand the QA list to equipment "important to safety" will not be pursued. Alternatively, the footnote may be associated with issuance of 10 CFR 50.34 (f)(3)(ii). Further review is necessary to determine whether the rulemaking referred to in the footnote corresponds to 10 CFR 50.34(f)(3)(ii).

4. The footnote should be updated based on a review of the documents referred to in the NUREG-0933 discussion. It is also noted that 10 CFR 50.34 (f)(3)(ii) requires treatment of equipment "important to safety" in the QA list. In the reviews of the ABWR (FSER 20.5.42) and System 80+ (FSER 20.4, Issue I.F.1) designs, this requirement was handled through use of the staff review of quality group classifications per SRP Section 3.2.2. Additional review of the FSER discussions should be performed and applicable staff positions incorporated into SRP Section 17.1.

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5. In any case, 50.34 (f)(3)(ii) needs to be included in the acceptance criteria for SRP Section 17.1.

Additionally, citations to the requirements stated in 10 CFR 50.34 (f)(3)(iii) on various aspects of a quality assurance program should be added to SRP Section 17.1. It appears that these requirements have already been reflected in SRP Section 17.1 acceptance criteria; however, the SRP section does not cite 50.34 (f)(3)(iii).

6. Consider additional discussion related to GDC 1, equipment "important to safety," and 10 CFR 50.34 (f)(3)(ii). Further evaluation will be required at the draft revision stage to further clarify and focus the issues discussed above.

7. Add citations to 50.34 (f)(3)(iii).

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18612/NRC GENERIC LETTER 84-01

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**Integrated Impact Number:** 909      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Add acceptance criteria related to fire protection program audits.

Generic Letter 82-21 provides staff positions on an acceptable audit program in the fire protection area. These positions would provide additional guidance beyond BTP CMEB 9.5-1 and Appendix B which are cited in acceptance criterion 2A1.d.

Consider augmenting acceptance criterion 2A1.d with the provisions of GL 82- 21.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18613/NRC GENERIC LETTER 82-21

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**Integrated Impact Number:** 910      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Add acceptance criteria for code verification activities.

Generic Letter 83-11 stated the staff position that users of safety analyses computer codes in

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support of licensing actions should demonstrate their proficiency in the use of the codes by submitting code verification performed by themselves, and not others. These code verifications would compare results to experimental data, plant operational data, or other benchmarked analyses.

Consider augmenting SRP Section 17.1, Acceptance Criterion 3E4, with the staff position expressed in GL 83-11.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

19833/NRC GENERIC LETTER 83-11

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**Integrated Impact Number: 911      SRP Section Number: 17.1**

#### **Suggested Changes to the SRP Section:**

Modify Areas of Review discussion on standard designs and previously approved quality assurance topical reports.

SRP Section 17.1, Revision 2, provides a discussion on the extent of staff review performed in cases where the applicant is referencing a previously approved quality assurance program topical report or a standard design (fourth paragraph, Areas of Review subsection). This paragraph needs to reflect the provisions of 10 CFR 52.63 on finality of standard design certifications processed under Part 52.

Changes to design certification by the NRC are not permitted except where modification is necessary to secure compliance with the Commission's regulations applicable and in effect at the time the certification was issued, or to assure adequate protection of the public health and safety or the common defense and security. The current SRP section discussion, in contrast, deals with regulatory positions in effect at the time of docketing of the application and to positions determined to be significant to safety.

Consider adding discussions to reflect 10 CFR 52.63 in Areas of Review.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24474/CODE OF FED. REGS 10CFR52

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**Integrated Impact Number:** 912      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Modify Acceptance Criteria to reflect the requirements to implement and report changes to the quality assurance program.

10 CFR 50.55 (f) requires that construction permit holders implement a quality assurance program that meet the requirements of Part 50 Appendix B, and to report changes to the QA program to the NRC, including the need for prior NRC approval for changes that reduce prior commitments. SRP Section 17.1, Revision 1, Acceptance Criterion 2B2 needs to be changed to reflect 50.55 (f) requirements. In addition, Acceptance Criterion 2B4 needs to include 50.55 (f) in the listing of regulations that must be covered by procedures.

Consider reflecting 10 CFR 50.55 (f) requirements in acceptance criteria.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24475/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 913      **SRP Section Number:** 5.4.6

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure that addresses the adequacy of RCIC pump minimum flow.

NRC Bulletin 88-04 identified a concern for potential damage to safety-related centrifugal pumps from operation and testing in the minimum flow mode. The Bulletin requested licensees to evaluate the adequacy of the minimum flow bypass lines and verify the minimum flow rate with the pump supplier. Minimum flow rates should be sufficient to ensure that there will be no pump damage from low flow operation. Also, the Bulletin states that some pump manufacturers are advising that their pumps should have minimum flow capacities of 25% to more than 50% of best efficiency flow for extended operation to protect against hydraulic instability or impeller recirculation problems.

Consideration should be given to adding a Review Procedure that addresses RCIC pump minimum flow capacity in relation to the pump manufacturers' recommendations.



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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24476/NRC BULLETIN 88-04; 24477/FINAL SER ABWR CH 5

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**Integrated Impact Number:** 916      **SRP Section Number:** 17.3

#### **Suggested Changes to the SRP Section:**

Add references for guidance related to fire protection quality assurance requirements.

Generic Letter 86-10, Section D, discusses quality assurance requirements applicable to fire protection systems. Generic Letter 82-21 provides staff positions on an acceptable audit program in the fire protection area. These positions would provide additional guidance beyond BTP CMEB 9.5-1 which is listed as a reference under VI.B.1. That reference is invoked in Acceptance Criterion II.A.7.c as providing programmatic QA guidance for specific items and activities that are important to safety.

Consider adding references to GL 86-10 and GL 82-21 to augment the current reference to BTP CMEB 9.5-1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20001/NRC GENERIC LETTER 82-21; 20049/NRC GENERIC LETTER 86-10

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**Integrated Impact Number:** 917      **SRP Section Number:** 17.3

#### **Suggested Changes to the SRP Section:**

Add a reference to provide additional guidance on the dedication of commercial grade equipment.

Generic Letter 91-05 provides staff positions with respect to certain aspects of licensee commercial grade procurement and dedication programs. For example, staff positions for critical characteristics and like-for-like replacements are discussed in the generic letter and its enclosure. In addition, GL 91-5 clarified certain portions of Generic Letter 89-02 which discussed elements of an effective procurement and dedication program to improve the detection of counterfeit or fraudulently marketed products.

GL 89-02 is currently listed as Reference VI.A.7 in SRP Section 17.3. Acceptance Criterion II.A.7.b invokes the documents listed as references in VI.A.

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Consider adding a reference to GL 91-5 to augment the current reference to GL 89 02.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20089/NRC GENERIC LETTER 91-05

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**Integrated Impact Number:** 918      **SRP Section Number:** 17.3

#### **Suggested Changes to the SRP Section:**

Add a discussion related to 50.34 (f) requirements.

10 CFR 50.34 (f)(3)(ii) and (iii) establish requirements related to the quality assurance list and various considerations for the quality assurance program. SRP Section 17.3, Revision 0, Acceptance Criterion II.A.7.a lists various applicable regulations for the quality assurance program.

Consider adding citations of 50.34 (f)(3)(ii) and (iii) to Acceptance Criterion II.A.7.a.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24481/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 920      **SRP Section Number:** 17.3

#### **Suggested Changes to the SRP Section:**

Add acceptance criteria discussion on 10 CFR 50.120.

10 CFR 50.120 provides requirements on implementation and maintenance of a training program for nuclear power plant personnel for the operations phase. As indicated in 50.120 (b)(2), implementation is subject to the quality assurance program. Discussion of 10 CFR 50.120 needs to be included in Acceptance Criterion II.A.7.a which itemizes regulatory requirements related to the overall QA program.

Consider adding 10 CFR 50.120 to Acceptance Criterion II.A.7.a.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24483/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 921      **SRP Section Number:** 4.4

#### **Suggested Changes to the SRP Section:**

Modify Acceptance Criteria and Review Procedures regarding TMI Action Plan Requirement II.F.2 and inadequate core cooling instrumentation.

TMI Action Plan item II.F.2 of NUREG 0737 required operating license holders and applicants provide instrumentation that provides unambiguous, easy-to-interpret, indication of inadequate core cooling (ICC). This item also requires development of procedures to be used with the proposed ICC instrumentation. 10 CFR 50.34(f)(2)(xviii) establishes an equivalent requirement applicable to Construction Permit and Manufacturing License applicants and, via 10 CFR 52.47, Design Certification applicants.

Enclosure 3 of NUREG 0737 describes staff positions with regard to ICC instrumentation and includes positions regarding the incorporation of this instrumentation in plant training and procedures. Following the publication of NUREG 0737, Generic Letters were issued that describe positions relevant to the implementation of II.F.2. Generic Letter 82-28 describes positions related to PWRs and Generic Letter 84-23 describes positions related to BWRs. In addition, and related to ICC instrumentation, Generic Letter 92-04 was issued that describes concerns with noncondensable gases that may become dissolved in the reference leg of BWR water level instrumentation and can lead to a false high level indication after a rapid depressurization event. Generic Letter 92-04 contains positions related to the resolution of these concerns.

The SRP currently cites TMI Action Plan item II.F.2 in Acceptance Criteria (specific criteria) item 9, however, does not reflect the additional guidance related to the issues identified above.

Consideration should be given to modifying the SRP Section to reflect additional guidance regarding instrumentation to provide indication of inadequate core cooling instrumentation.

As documented in 6/9/94 memorandum from J. Wermiel to A. Gody the instrumentation and controls branch (HICB) disapproved a similar potential impact for inclusion in Chapter 7.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24488/NRC GENERIC LETTER 92-04; 24489/NRC GENERIC LETTER 84-23;  
24490/FINAL SER ABWR CH 20; 24491/NRC GENERIC LETTER 82-28

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**Integrated Impact Number:** 924      **SRP Section Number:** 4.4

#### **Suggested Changes to the SRP Section:**

Develop review procedures to address core protection algorithms and logic functionality.

In their review of the CE System 80+ thermal-hydraulic design under SRP Section 4.4, the staff requested that ABB-CE 1) describe the software design and algorithms for the core protection calculator and the control element assembly calculator system, 2) identify any differences between the system proposed for the CE 80+ and those previously approved by the NRC and evaluate the impacts on performance and safety functions, and 3) discuss the verification program for implementation of the software during the certification process. This system is designed to trip the reactor on low DNBR and or high local power density to ensure protection of the fuel design limits. In response to the staff's request, ABB-CE provided a list of documents that establish the basis for the design of the software and for controlling subsequent changes to the algorithms and data. The NRC accepted ABB-CE response based on previous NRC SERs issued for the ABB-CE documentation.

The staff further states in the FSER that the software implementation will be verified by testing on a plant specific basis. The testing will verify software modification implementation with regard to calculation systems, software-hardware integration, and confirmation of integrated system performance when matched against the predictions of design analyses.

Consider developing review procedures to address the evaluation and testing of core protection software systems to ensure that algorithms, logic, and data are functional and consistent with the thermal-hydraulic analyses.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24502/FINAL SER CE80 CH 4; 24503/FINAL SER CE80 CH 4

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**Integrated Impact Number:** 926      **SRP Section Number:** 6.2.1.1.C

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures subsections to incorporate the review of

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ABWR containments.

The ABWR containment was reviewed by the NRC as described in the ABWR FSER, NUREG-1503, July 1994. Sections 3.6.2.2, 6.2.1.2, 6.2.1.3, 6.2.1.5.1, 6.2.1.6, 6.2.1.8, 15.4.4.1, and 20.1.8 of the ABWR FSER established specific acceptance criteria and review procedures for the ABWR containment. Many of the acceptance criteria and review procedures already established in SRP 6.2.1.1.C are specific to Mark I, II, and III BWR containments and were applied to the ABWR containment as applicable. Certain criteria and review procedures were also established for the ABWR containment that are not specifically described in SRP Section 6.2.1.1.C.

Consider revising the Acceptance Criteria and Review Procedures subsections to add applicable references appropriate to the review of ABWR containments.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24492/FINAL SER ABWR CH 6; 24493/FINAL SER ABWR CH 20; 24494/FINAL SER ABWR CH 6; 24495/FINAL SER ABWR CH 3; 24496/FINAL SER ABWR CH 6; 24497/FINAL SER ABWR CH 15; 24498/FINAL SER ABWR CH 6; 24499/FINAL SER ABWR CH 6; 24500/FINAL SER ABWR CH 6

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**Integrated Impact Number:** 927      **SRP Section Number:** 15.3.3 - 15.3.4

#### **Suggested Changes to the SRP Section:**

Update acceptance criteria to reflect staff approved analytical methods.

The last paragraph of Acceptance Criteria states that the equations, sensitivity studies, and models described in References 7 through 11 are acceptable. Reference 9 applies to CE plants and refers to the August 1973 CESSAR. In the System 80+ FSER, more recent analytical methods used in non- LOCA analyses are listed in Table 15.3, including CESEC-III, TORC, and HERMITE, and the CE-1 correlation, with references to the associated NRC approval letters.

Consider updating the acceptance criteria and related reference to reflect staff approved analytical methods used to analyze the reactor coolant pump shaft seizure event.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24505/FINAL SER CE80 CH 15

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**Integrated Impact Number:** 928      **SRP Section Number:** 15.3.3 - 15.3.4

#### **Suggested Changes to the SRP Section:**

Incorporate applicable provisions of the revised source term.

SRP Section 15.3.3-15.3.4 includes the determination of the radiological consequences of a postulated rotor seizure or broken shaft event. Acceptance Criteria refer to Part 100 dose limits. Review Procedures include a determination of the fraction of fuel rod failures for use in radiological dose calculations. Evaluation Finding (f) discusses the use of the computer code SARA. In Section 15.4.2.3 of the System 80+ FSER, the staff discusses the use of draft NUREG-1465 with regard to gap fractions and relevant isotopes (noble gases, iodines, cesiums, and rubidiums) and chemical species of iodines in the fuel rod gap.

Consider incorporating applicable provisions from draft NUREG-1465.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24506/FINAL SER CE80 CH 15

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**Integrated Impact Number:** 931      **SRP Section Number:** 17.2

#### **Suggested Changes to the SRP Section:**

Modify Acceptance Criteria to reflect the requirement to implement and report changes to the quality assurance program.

10 CFR 50.54(a)(1) - (a)(3) requires that operating license holders implement a quality assurance program that meets the requirements of Part 50 Appendix B, and to report changes to the QA program to the NRC, including the need for prior NRC approval for changes that reduce prior commitments. SRP Section 17.1, Revision 2, Acceptance Criterion 2B2 addresses changes to the QA program and notification to the NRC. This acceptance criterion will be changed for SRP Section 17.1 to incorporate the requirements of 10 CFR 50.55(f) which applies to construction permit holders.

SRP Section 17.2 which relies heavily on SRP Section 17.1 should state a difference to SRP Section 17.1 for this acceptance criterion, replacing 50.54(a) for 50.55(f).

Consider adding acceptance criteria to address 10 CFR 50.54(a).

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

18652/CODE OF FED. REGS 10CFR50; 19868/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 932      **SRP Section Number:** 17.2

#### **Suggested Changes to the SRP Section:**

Add acceptance criteria discussion to address Part 21 reporting requirements.

SRP Section 17.1, Revision 2, Acceptance Criterion 2B3, states that the applicant should commit to conduct activities under 10 CFR 50.55(e) in accordance with the QA program. The corresponding regulation for the operations phase, vice design and construction, is 10 CFR Part 21. SRP Section 17.2 relies on SRP Section 17.1 acceptance criteria unless otherwise stated.

Consider adding an acceptance criterion to SRP Section 17.2 noting the difference to SRP Section 17.1 by replacing the reference to 50.55(e) with a reference to Part 21.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

19842/CODE OF FED. REGS 10CFR21

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**Integrated Impact Number:** 934      **SRP Section Number:** 17.2

#### **Suggested Changes to the SRP Section:**

Add acceptance criteria discussion on 10 CFR 50.120.

10 CFR 50.120 provides requirements on implementation and maintenance of a training program for nuclear power plant personnel for the operations phase. SRP Section 17.1, Revision 2, Acceptance Criterion 2B3 include a discussion of regulations which must be met with respect to the quality assurance program. For SRP Section 17.2, a difference to SRP Section 17.1 needs to be noted to state the applicability of 50.120 to the operations phase.

Consider adding an acceptance criterion to SRP Section 17.2 noting the difference to SRP Section 17.1 by adding a reference to 50.120 in acceptance criterion 2B3.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24504/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 935      **SRP Section Number:** 3.9.4

#### **Suggested Changes to the SRP Section:**

Add a review procedure to address the guidance contained in Generic Letter 86-01 regarding safety concerns associated with pipe breaks in the BWR scram system.

Generic Letters 81-20, 34, and 35 provided licensees with information in the form of an AEOD report (Generic Letter 81-20) and NUREG-0803 (Generic Letters 81-33, 34, and 35) that documented potential safety concerns associated with the BWR scram system. At the time these Generic Letters were written the staff was concerned that the basis for assuring the mechanical integrity of important reactor pressure boundary components within the scram system may have been inadequate in view of the potentially important safety risks associated with a scram system pipe break. The NUREG-0803 guidelines were developed to address the consequences of a staff postulated leak in the scram system. The potential for a leak in the scram system was based upon conservative assumptions which were later clarified and modified by Generic Letter 86-01.

In Generic Letter 86-01, the staff concluded, based upon the system classification and low stress threshold, the system satisfies BTP MEB 3-1, position B.2.c( 1) and that a through wall leakage crack thus need not be postulated. Since the SDV piping system fulfills the above criteria, breaks and through wall cracks in the SDV piping need not be postulated. In addition, the staff has concluded that, even if a staff-postulated through-wall flaw is initially present in the SDV system, it will grow negligibly and will not propagate into a break under the staff defined piping loads. Further, leakage from such a flaw will be small (less than or equal to about 5 gpm) and, therefore, a harsh environment over large areas of the reactor building which could affect redundant safety related mitigating equipment will not result. Thus, the potentially exposed safety-related equipment need not be qualified for operation in a harsh environment associated with an SDV break. Section 1.4 of the safety evaluation report enclosed with Generic Letter 86-01 reflects that the staff's approach to the issue of postulated SDV pipe failures described in the safety evaluation report supersedes the approach described in Generic Letters 81-20, 34, and 35 and NUREG-0803.

In addition, the staff concluded that the revised BWROG Emergency Procedure Guidelines for secondary containment control (NEDO-24934), together with normal plant procedures and the proposed periodic visual verification of the scram system piping integrity (BWROG-8420), provide sufficient measures for detecting and mitigating the consequences of leakage which may occur in the SDV piping system. The design basis of the SDV piping system has considered transient forces resulting from the worst case control rod drive (CRD) system actuation. Although water hammer has been analytically postulated and hydraulic instabilities have been experienced in the CRD system, no events have been experienced of a severity significant enough to constitute a water hammer. Therefore, water hammer is not considered a contributing



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factor in potential SDV pipe breaks.

Consideration should be given to adding a review procedure that address the conclusions and clarifications of Generic Letter 86-01 regarding classification of the system and the staff positions which preclude the need to postulate a through wall leakage crack. Also consideration should be given to discussing the conclusions regarding the plant procedures and proposed periodic visual verification of the integrity of the scram system piping.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24508/NRC GENERIC LETTER 86-01; 24509/NRC GENERIC LETTER 81-20; 24510/NRC GENERIC LETTER 81-35; 24511/NRC GENERIC LETTER 81-33; 24512/NRC GENERIC LETTER 81-34

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**Integrated Impact Number:** 936      **SRP Section Number:** 15.3.3 - 15.3.4

#### **Suggested Changes to the SRP Section:**

Augment acceptance criteria related to TMI Action Item II.K.3.5.

Acceptance Criterion II.6 deals with tripping of the reactor coolant pumps and specifies that this should be done consistent with the resolution to Action Item II.K.3.5. Generic Letters 83- 10A through 83-10F provides the resolution of this TMI Action Plan item with detailed and extensive guidance for developing RCP trip setpoints and methods for RCP operation during all transients and accidents, including small break LOCAs and SGTRs. Generic Letters 85-12, 86-05 and 86-06 provide staff review findings for owners group submittals in response to Generic Letters 83-10A through 83-10F.

Consider adding references to the above discussed generic letters in Acceptance Criterion II.6.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24514/NRC GENERIC LETTER 83-10A

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**Integrated Impact Number:** 937      **SRP Section Number:** 13.4

#### **Suggested Changes to the SRP Section:**

Update Regulatory Guide 1.33 to include the latest version of ANSI Standard N18.7.

Regulatory Guide 1.33 and its associated standard, ANSI N18.7, are cited in acceptance Criteria

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II.1.a and II.2 for the scope of plant staff reviews and provisions for independent review. Regulatory Guide 1.33, Rev. 2, February 1978 endorses ANSI N18.7-1976/ANS-3.2, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," which now has a 1988 revision date. The revised standard could affect the criteria for administrative controls and quality assurance during the operational phase of plant life.

Consider updating Regulatory Guide 1.33 to encompass the latest revision of ANS N18.7 or successor standard.

IPD-7.0 Form 17.2-1 has been initiated for the future revision of Regulatory Guide 1.33 to endorse the current version of ANSI N18.7/ANS-3.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

7170/RG 1.33; 15880/RG 1.33; 15883/RG 1.33; 15885/RG 1.33

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**Integrated Impact Number:** 939      **SRP Section Number:** 15.3.1 - 15.3.2

#### **Suggested Changes to the SRP Section:**

Update the reference on approved analyses methods.

SRP Section 15.3.1 - 15.3.2, Revision 1, states in Acceptance Criteria that the equations, sensitivity studies, and models described in References 6 through 9 are acceptable. Reference 8 is the August 1973 System 80 SSAR. In the FSER for the System 80+ design, Section 15.1 lists the analytical methods used for the transient analyses, and the associated NRC approval letters, in Tables 15.3 and 15.4.

Consider updating Reference 8 to reflect more recent staff-approved analyses methods.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24523/FINAL SER CE80 CH 15

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**Integrated Impact Number:** 940      **SRP Section Number:** 15.3.1 - 15.3.2

#### **Suggested Changes to the SRP Section:**

Update the reference on approved analyses methods.

SRP Section 15.3.1 - 15.3.2, Revision 1, states in Acceptance Criteria that the equations,

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sensitivity studies, and models described in References 6 through 9 are acceptable. Reference 6 is the April 1973 SSAR for the BWR/6 design. In Section 15.1 of the ABWR FSER, the staff discusses its acceptance review of the ODYNA and REDYA computer codes used in the ABWR transient analyses.

Consider updating Reference 6 to reflect more recent staff-approved analyses methods.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24524/FINAL SER ABWR CH 15

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**Integrated Impact Number:** 941      **SRP Section Number:** 15.4.1

#### **Suggested Changes to the SRP Section:**

Modify the Review Procedures applicable to CE System 80+ applications to include the assumption of the loss of offsite power (LOOP), in addition to the limiting single failure event, for the analysis of this anticipated operational occurrence.

SRP Section 15.4.1 addresses the control rod withdrawal error at low power. CE was required to consider a LOOP in conjunction with a single failure for the performance of the analysis.

In the CE 80+ FSER, the staff took a position that GDC 17 requires that the LOOP not be considered as a single failure event, but should be assumed in the analysis for each transient and accident.

This position appears to have been applied only to the CE 80+ review. This position does not appear in the ABWR FSER, and no other regulatory correspondence could be located to support other applications of this position. Neither does it appear to be a current requirement for plants preparing cycle licensing submittals.

Consider modifying review procedures to reflect the staff position in the CE 80+ FSER that analysis of control rod withdrawal error at a low power include consideration of a LOOP in conjunction with a limiting single failure.

PIPB Comment:

SRXB confirmed that this staff position was new to the CE 80+ SER and not previously addressed. GDC 17 requires consideration of a LOOP in transient analysis and LOOP should be considered in all events. Lacking specific guidance in the SRP, GDC 17 prevails as an acceptance criterion and should be added to those sections where turbine trip results from the

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transient.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24526/FINAL SER CE80 CH 15

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**Integrated Impact Number:** 942      **SRP Section Number:** 9.5.1

#### **Suggested Changes to the SRP Section:**

IEEE Standard 383-1974 is cited in Branch Technical Position (BTP) CMEB 9.5-1.

The IEEE Standard was reaffirmed in 1992. The latest version is IEEE Std 383 - 1974 (R92).

Consider updating the BTP to cite the latest version of IEEE Std. 383.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23905/C&S: IEEE 383

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**Integrated Impact Number:** 943      **SRP Section Number:** 9.5.1

#### **Suggested Changes to the SRP Section:**

Revise SRP Section 9.5.1, Branch Technical Position 9.5-1 to incorporate staff positions from the ABWR and CE 80+ FSERs regarding seismic fire water supply system capacity.

Both the ABWR and CE 80+ FSERs require that volume-limited (i.e., tanks) primary fire water supply systems must have a passive reserve to supply the seismically designed portion of the fire suppression system for a period of 2 hours. The SRP section and associated BTP do not currently identify such a criteria.

Consider incorporating the staff position regarding passive reserve fire water supplies in BTP CMEB 9.5-1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24534/FINAL SER ABWR CH 9; 24535/FINAL SER CE80 CH 9

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**Integrated Impact Number:** 944      **SRP Section Number:** 13.4

#### **Suggested Changes to the SRP Section:**

Add acceptance criteria for post-trip review.

Generic Letter 83-28 discussed staff positions on the program, procedures, and data collection capability to assure that the causes for unscheduled reactor shutdowns, as well as the response of safety-related equipment, are fully understood prior to plant restart. Post trip review needs to include a determination for the need for independent assessment of an event. The enclosure to GL 83-28 provides a discussion for Item 1.1 Post Trip Review (Program Description and Procedure) and for Item 1.2 Post Trip Review - Data and Information Capability.

Consider adding acceptance criteria and a reference to reflect items 1.1 and 1.2 of GL 83-28.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

16077/NRC GENERIC LETTER 90-04; 24513/NRC GENERIC LETTER 83-28

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**Integrated Impact Number:** 945      **SRP Section Number:** 13.3

#### **Suggested Changes to the SRP Section:**

Add a discussion of Regulatory Guide 1.101, Revision 3, to Areas of Review, Acceptance Criteria, and Evaluation Findings.

Regulatory Guide 1.101, Rev. 3, provides staff positions on acceptable methods for complying with 10 CFR Part 50.47, "Emergency Plans," and 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities." These positions discuss Revision 1 of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," and NUMARC/NESP-007, Rev. 2, January 1992, "Methodology for Development of Emergency Action Levels." NUMARC/NESP-007, Rev. 2, provides an alternative method for developing emergency action levels to that presented in NUREG-0654, Rev. 1.

NUREG-0654, Rev. 1, is addressed in SRP Section 13.3. NUMARC/NESP-007, Rev. 2, is not addressed in SRP Section 13.3.

Consider modifying Areas of Review, Acceptance Criteria, and Evaluation Findings to include a discussion of Regulatory Guide 1.101, Rev. 3, and NUMARC/NESP-007, Rev. 2.

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(NOTE: This ROC has been changed and will not be processed further. NUREG-0654 Rev. 1 supersedes Regulatory Guide 1.101 and therefore RG 1.101 was eliminated from an earlier draft version of this section. )

#### **Potential Impacts/Documents supporting the Suggested Changes:**

10405/RG 1.101; 22137/RG 1.101

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**Integrated Impact Number:** 946      **SRP Section Number:** 13.3

#### **Suggested Changes to the SRP Section:**

Add a discussion of Supplement 1 to NUREG-0737 to Review Procedures.

Supplement 1, in part, provides additional clarification regarding NUREG-0737 Items: I.D.2, "Safety Parameters Display Systems;" III.A.1.2, "Upgrade of Emergency Support Facilities," and; III.A.2.2, "Meteorological Data." (I.D.2 is reviewed only to assure that a slave of the SPDS is located in the Technical Support Center and the Emergency Operations Facility.) These topics, which are addressed in subsection IV, Review Procedures, should be upgraded.

Consider modifying Review Procedures relative to the discussion of NUREG-0737, III.A.1.2, and III.A.2.2 to include reference to the guidance provided in supplement 1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

10747/NRC GENERIC LETTER 82-33

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**Integrated Impact Number:** 947      **SRP Section Number:** 13.3

#### **Suggested Changes to the SRP Section:**

Add a discussion of NUREG-0654, FEMA-REP-1, Rev. 1, Supplement 1, to SRP Section 13.3.

NUREG-0654, FEMA-REP-1, Rev. 1, Supplement 1 provides guidance for the development, review, and evaluation of utility offsite radiological emergency response planning and preparedness for those situations in which state or local governments decline to participate in emergency planning. Supplement 1 is intended to be used with Section 1 and Appendices 1-5 of NUREG-0654, FEMA-REP-1, Rev 1.

Consider modifying Areas of Review, Acceptance Criteria and Evaluation Findings to include discussion of the staff positions presented in NUREG-0654, FEMA-REP-1, Rev. 1, Supplement

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24521/NUREG 0654 SUP 1

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**Integrated Impact Number:** 948      **SRP Section Number:** 15.4.1

#### **Suggested Changes to the SRP Section:**

Modify the Review Procedures to assure that applicants presenting BWR submittals have considered the control rod assembly withdrawal during refueling or control blade removal error during refueling operations while assessing the impact of the uncontrolled control rod assembly withdrawal anticipated operational occurrence.

Reg Guide 1.70 Table 15-1, section 4.1 indicates that the analysis of the anticipated operational occurrence of an uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition should also address a control rod or temporary control device removal error during refueling operations. SRP Section 15.4.1 does not explicitly provide for review of control rod assembly withdrawal or control blade removal error during refueling operations.

In the Final SER for ABWRs, GE assessed the transient and determined that no analysis would be required because the refueling interlock in the rod control system precluded the transient from occurring.

Information Notice 83-35 describes two separate incidents in which plants loaded fuel assemblies in core locations in which the control rod had been removed. The notice also stated that Standard BWR Technical Specifications allows the defeating of the refueling interlock for removed control rods to allow for multiple control rod removal for control rod drive maintenance. Fuel movement with control rods removed is allowed by Technical Specifications at most facilities. This allows physical and neutronic configurations to be established that are identical to a control rod removal error.

Consider developing Review Procedures to verify that BWR applicants have assessed a control rod assembly withdrawal or control blade removal error during refueling operations.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24527/RG 1.70; 24585/FINAL SER ABWR CH 15; 24586/NRC NOTICE 83-35

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**Integrated Impact Number:** 949      **SRP Section Number:** 3.7.4

#### **Suggested Changes to the SRP Section:**

Revise specific criteria and Review Procedures to accommodate improved seismic instrumentation.

SRP 3.7.4 provides for review of the seismic instrumentation and of procedures that will be followed to inform the control room operator of the earthquake level shortly after an earthquake. Regulatory Guide 1.12 is cited as specific criteria for acceptability of seismic instrumentation.

As discussed in the ABWR and CE 80+ FSERs, design certification is predicated, in part, on the ability to ascertain the need to take action to shut down the plant and to evaluate the performance of structures, systems and components. In the ABWR and CE80+ FSERs, the NRC staff reviewed criteria developed by the Electric Power Research Institute (EPRI) for evaluating the need to shut down the plant following an earthquake. The staff's stated exceptions to EPRI guidelines establish information needs that should be addressed by the seismic instrumentation. In the ABWR FSER, the staff also stated that because of the continuous enhancement in the state of the art of seismic instrumentation, and the proposed revisions to Appendix A of 10 CFR 100 and to RG 1.12, conformity with instrumentation guidelines in existence at the time of an individual license application will be required.

Pending issuance of the proposed revisions to Appendix A of 10 CFR 100 and Regulatory Guide 1.12, consider revising specific criteria and Review Procedures for seismic instrumentation to accommodate improved technologies.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4321/RG 1.12; 24545/FINAL SER ABWR CH 3; 24547/FINAL SER CE80 CH 3

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**Integrated Impact Number:** 952      **SRP Section Number:** 13.2.1

#### **Suggested Changes to the SRP Section:**

Update SRP Section 13.2.1 to include the current review criteria addressing reactor operator training.

10 CFR Part 55, Section 55.45(a)(1) through (13) addresses the recommended content of operating tests.

10 CFR Part 55, Section 55.43 describes the required content of the written examination for



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senior operators.

The above listed PIs address topics of the Code that are referenced in SRP Section 13.2.1 and are from 10 CFR Part 55. 10 CFR Part 55 went through a major rewrite in 1987 which accounts for the outdated citations to Part 55 sections noted in the 1981 version of SRP Section 13.2.1. In addition, Generic Letters 82-18 and 84-14 contain guidance regarding requalification requirements.

Consider revising SRP Section 13.2.1 to reflect the review criteria and the requirements of the Code of Federal Regulations as currently written.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

6756/CODE OF FED. REGS 10CFR55; 16374/CODE OF FED. REGS 10CFR55;  
16375/CODE OF FED. REGS 10CFR55; 24452/NRC GENERIC LETTER 82-18; 24453/NRC  
GENERIC LETTER 84-14

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**Integrated Impact Number:** 953      **SRP Section Number:** 13.2.1

#### **Suggested Changes to the SRP Section:**

Update SRP Section 13.2.1 to address the requirements of 10 CFR Part 50, Section 50.120.

10 CFR Part 50, Section 50.120 states that each nuclear power plant licensee shall have a training program derived from a systems approach to training.

Implementation of the requirements of Section 50.120 could affect the contents of operator training programs as the current training programs may not cover the five elements listed in 10 CFR Part 55, Section 55.4, "Definitions." 10 CFR 50.120(b)(2) specifies requirements for periodic evaluation and revision of the training program; periodic review of effectiveness; and record keeping.

Consider updating SRP Section 13.2.1 to include review criteria necessary to determine the adequacy of operator training programs in complying with the requirements of 10 CFR Part 50, Section 50.120.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22423/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 954      **SRP Section Number:** 13.2.1

#### **Suggested Changes to the SRP Section:**

Add requirements for the use of a simulation facility in operating tests.

SRP Section 13.2.1, Revision 0, Acceptance Criteria include TMI Action Plan items I.A.3.1 and I.A.4.2 which deal with the use of simulators in operator and senior operator licensing examinations. As part of the resolution of TMI Action Plan item I.A.4.2, the NRC issued 10 CFR 55.45 which, in part, requires the use of a simulation facility in the operating test, and NUREG-1258 which provides the staff procedure for evaluating certified simulation facilities. In addition, 10 CFR 50.34 (f)(2)(i) requires that the simulator correctly models the control room and includes the capability to simulate small-break LOCA's.

Consider updating the acceptance criteria for TMI Action items I.A.3.1 and I.A.4.2 to include discussion of 10 CFR 55.45(b), 10 CFR 50.34 (f)(2)(i), and NUREG-1258.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

16333/CODE OF FED. REGS 10CFR50; 16337/CODE OF FED. REGS 10CFR50;  
16393/CODE OF FED. REGS 10CFR55; 24540/CODE OF FED. REGS 10CFR55

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**Integrated Impact Number:** 956      **SRP Section Number:** 13.2.2

#### **Suggested Changes to the SRP Section:**

Eliminate the apparent inconsistency between RG 1.8, ANSI/ANS 3.1, ANSI/ANS N18.1, and SRP 13.2.2.

This RG describes the method acceptable to the NRC staff for complying with the Commission's regulations with regard to the training and qualifications of nuclear power plants personnel. Section C, "Regulatory Position," describes the positions in ANSI/ANS-3.1-1981 and ANSI/ANS N18.1-1971 that are endorsed, in part, by the RG.

Review of the SRP Section 13.2.2. II, "Acceptance Criteria," discloses that the sections in these documents that are endorsed by the RG are inconsistent with those that are endorsed by the SRP. Following are a few examples of the discrepancies between the two documents.

#### **RG 1.8**

Takes exceptions to Section 5.1 of ANSI/ANS-3.1-1981.

Endorses certain portions of Section 4.3 and 4.5 of ANSI/ANS-3.1-1981.

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#### **SRP 13.2.2**

Endorses Section 5.1 entirely.

Does not include Sections 4.3 or 4.5 among those that are endorsed by the SRP.

It appears that the version of ANS 3.1 cited in SRP 13.2.2, Revision 0, is the 1978 version. As explained in RG 1.8, Revision 2, ANS 3.1-1978 was not endorsed; instead portions of ANS 3.1-1981 were endorsed with exceptions along with ANSI N18.1-1971. SRP 13.2.2 needs to be revised to reflect the staff positions stated in RG 1.8, Revision 2.

Additionally, as mentioned above, RG 1.8 endorses the issues of ANSI/ANS-3.1-1981 and ANSI/ANS N18.1-1971. Review of NUREG/CR-5973, Rev. 1 discloses that the latest issue of these documents is 1993.

A detailed comparison between the current issues of ANSI/ANS-3.1, ANSI/ANS 18.1, and RG 1.8 would be needed to determine to what extent they are acceptable to the staff, and update the SRP 13.2.2 accordingly.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

17471/RG 1.8

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**Integrated Impact Number:** 957      **SRP Section Number:** 13.2.2

#### **Suggested Changes to the SRP Section:**

Add a discussion in the Acceptance Criteria to reflect the requirements of 10 CFR 52.78 and 10 CFR 50.120.

10 CFR 52.78 requires that for the operating phase of the plant the applicant must demonstrate compliance with the requirements of the training programs established in §50.120 of 10 CFR Chapter I, which requires that each nuclear power plant applicant, by November 22, 1993, or 18 months prior to fuel load, whichever is later, and each nuclear power plant licensee, by November 22, 1993 shall establish, implement, and maintain a training program derived from a system approach to training as defined in 10 CFR 55.4. The training program must include several categories of non-licensed personnel as specified in 10 CFR 50.120.

Review of the pertinent sections of the SRP disclosed that 10 CFR 52.78 and §50.120 are not referenced or incorporated in the SRP.

Consider modifying Acceptance Criteria to include a discussion of compliance with 10 CFR 52.78 and 10 CFR 50.120.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22424/CODE OF FED. REGS 10CFR50; 22428/CODE OF FED. REGS 10CFR52

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**Integrated Impact Number:** 959      **SRP Section Number:** 13.2.2

#### **Suggested Changes to the SRP Section:**

Modify acceptance criteria to reflect issuance of a policy statement.

Acceptance Criterion II.6 (Note: there are two "II.6" items in SRP Section 13.2.2, Revision 0, due to a typographical error) cites TMI Action Plan Item I.A.1.1 for requirements related to a shift technical advisor. In October, 1985, the NRC issued the Policy Statement on Engineering Expertise on Shift which provided two options for meeting the requirement to have an STA available to the shift. The policy statement is addressed in Regulatory Guide 1.8, Revision 2, which is an acceptance criterion for SRP Section 13.2.2, Revision 0.

In addition, Acceptance Criterion II.6.C discusses absences from STA duties for a period of 30 days or longer. This needs to be revised to reflect Regulatory Position C.1.j which deals with performance of STA duties at least three shifts per quarter.

Consider modifying Acceptance Criterion II.6 to reflect the issuance of the policy statement on engineering expertise on shift.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

17535/NRC POLICY STATEMENT 50 FR 43621; 17537/NRC POLICY STATEMENT 50 FR 43621

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**Integrated Impact Number:** 960      **SRP Section Number:** 13.2.2

#### **Suggested Changes to the SRP Section:**

Add acceptance criteria for fitness for duty programs.

10 CFR 26.21 states requirements for fitness for duty training of personnel granted unescorted access to nuclear power plant protected areas, and to personnel required to physically report to the Technical Support Center or Emergency Operations Facility in accordance with licensee emergency plans and procedures. 10 CFR 26.22 covers training of supervisors and escorts.

Consider adding acceptance criteria and evaluation findings (similar to those for 10 CFR 19.12

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in SRP Section 13.2.2, Revision 0) related to 10 CFR Part 26 requirements.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24541/CODE OF FED. REGS 10CFR26

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**Integrated Impact Number:** 961      **SRP Section Number:** 13.5.1.1

#### **Suggested Changes to the SRP Section:**

Incorporate the staff policy with regard to shift manning requirements as stated in Generic Letters 82-02 and 82-12 and NRC Policy Statement 46 FR 23836.

In Generic Letter 82-02 the staff stated that they were revising Regulatory Guide 1.33 and NUREG-0737 (Item I.A.1.3) to reflect the policy that enough plant operating personnel should be employed to maintain adequate shift coverage without routine heavy use of overtime.

In Generic Letter 82-12 the policy statement was slightly revised and the revised version was attached to the letter. Generic Letter 82-12 restated Generic Letter 82-02 in that actions were underway to incorporate the Commission policy on working hours into Regulatory Guide 1.33 and into NUREG-0737 (Item I.A.1.3). In addition, Generic Letter 83-14 provided clarification to Generic Letter 82-12 on the term "key maintenance personnel."

NRC Policy Statement 46 FR 23836 conveys the same policy and information as do the generic letters.

Consider augmenting Acceptance Criterion II.A.5 on TMI Action Plan Item I.A.1.3 to reflect the staff policy governing required shift staffing in accordance with the provisions contained in Generic Letters 82-02, 82-12 and 83-14 and Policy Statement 46 FR 23836.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

7133/NRC GENERIC LETTER 82-12; 7213/NRC GENERIC LETTER 82-12; 7402/NRC GENERIC LETTER 82-02; 7404/NRC GENERIC LETTER 82-12; 18439/NRC POLICY STATEMENT 46 FR 23836; 18440/NRC POLICY STATEMENT 46 FR 23836; 18441/NRC POLICY STATEMENT 46 FR 23836; 18719/NRC GENERIC LETTER 82-02

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**Integrated Impact Number:** 962      **SRP Section Number:** 13.5.1.1

#### **Suggested Changes to the SRP Section:**

Incorporate the Code of Federal Regulations 10 CFR Part 26, "Fitness for Duty Programs," into SRP Section 13.5.1.

On June 7, 1989 (54 FR 24468), the NRC issued 10 CFR Part 26, "Fitness for Duty Programs," that requires each licensee authorized to construct or operate a nuclear power reactor to implement a fitness for duty program. Licensees and applicants are to develop and implement fitness for duty programs to provide reasonable assurance that all nuclear power plant personnel with access to vital areas at operating plants are fit for duty. The rule states that all licensed operators and senior operators will be subject to their facility licensees' fitness for duty requirements.

10 CFR 26.20 requires licensees to establish and implement written policies and procedures for a fitness for duty program. Minimum requirements are specified for the policy and procedures.

Generic Letter 91-16 describes Federal legislation that may affect fitness for duty issues.

Consider revising SRP Section 13.5.1 to reflect the Commission policy governing the required fitness for duty programs as outlined in 10 CFR Part 26. Specifically, consider augmenting Acceptance Criterion II.A which already includes staff positions on working hour limitations. (It is noted that fitness for duty deals with the use of illegal drugs and abuse of legal drugs, and also addresses other factors that could affect fitness for duty such as mental stress, fatigue and illness per 10 CFR 26.20.)

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18326/CODE OF FED. REGS 10CFR26; 18328/CODE OF FED. REGS 10CFR26;  
19735/CODE OF FED. REGS 10CFR26; 19736/CODE OF FED. REGS 10CFR26;  
19737/CODE OF FED. REGS 10CFR26; 19738/CODE OF FED. REGS 10CFR26;  
19821/NRC GENERIC LETTER 89-23; 22441/NRC GENERIC LETTER 91-16

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**Integrated Impact Number:** 963      **SRP Section Number:** 13.5.1.1

#### **Suggested Changes to the SRP Section:**

Add provisions for a vendor interface program in acceptance criteria.

Areas of Review Item I.A.3 addresses administrative procedures for the control of maintenance.

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Acceptance Criterion II.A lists staff positions related to administrative procedures. Generic Letter 83-28 stated staff positions with respect to the establishment and maintenance of a program to ensure that vendor information for safety related components is complete and appropriately referenced or incorporated in plant instructions and procedures (Item 2.1 for reactor trip system components and Item 2.2 for all safety related components). The staff positions in Generic Letter 83-28 were superseded by those in Generic Letter 90-03.

Consider adding Generic Letter 90-03 provisions on a vendor interface program in Acceptance Criterion II.A.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

16127/NRC GENERIC LETTER 83-28; 25909/NRC GENERIC LETTER 90-03

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**Integrated Impact Number:** 965      **SRP Section Number:** 5.4.12

#### **Suggested Changes to the SRP Section:**

Add new Acceptance Criteria regarding control room operability, backup power sources, capability to remove a steam bubble, seismic classification, and design for periodic inspection of the reactor coolant system high point vents.

Section 6.7.1 of the CE System 80+ FSER reviews the Reactor Coolant Gas Vent System (RCGSVS). Several specific areas are evaluated against specific GDCs or other regulatory documents as follows:

- The RCGVS will have the capability to be manually actuated, monitored, and controlled from the control room - GDC 19.

- The parallel isolation valves are powered from a normal and an emergency ac power source - GDC 17 and 34.

- The RCGVS can remove a steam bubble/depressurize the RCS to allow natural circulation cooldown - BTP RSB 5-1.

- The RCGVS is designed to seismic Category I requirements - Regulatory Guide 1.29.

- The RCGVS is designed to permit periodic inspection in accordance with ASME Section XI - GDC 36.

Section 20.4 (issue II.B.1) of the CE System 80+ FSER States that the staff has reviewed the

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RCGV system design and concludes, in Section 6.7.1 of the FSER, that it is acceptable because it meets the following design criteria: (1) the system must be operable from the control room (GDC 19), (2) the system must be testable (GDC 36), (3) the system must be capable of functioning following a LOOP (GDC 17), and (4) the system must be able to withstand an OBE (RG 1.29).

Consider modifying SRP Section 5.4.12 to add new Acceptance Criteria to incorporate the applicable GDCs and other regulatory documents discussed in the CE System 80+ FSER.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24576/FINAL SER CE80 CH 20; 24577/FINAL SER CE80 CH 6

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**Integrated Impact Number: 966      SRP Section Number: 9.5.3**

#### **Suggested Changes to the SRP Section:**

This impact will not be processed further based upon PRB comments on the related draft revision of SRP Section 9.5.3.

Add Review Procedures and a Review Interface for review of lighting system conformance to ALARA principles.

In the EPRI Evolutionary Plant FSER, in conjunction with the review of lighting systems, the staff discussed location of lighting fixtures and bulb life with respect to ALARA objectives. The staff determined that the EPRI proposed requirements in this regard meet the intent of Regulatory Guide 8.8. The staff subsequently applied the acceptable EPRI lighting system requirements, as supplemented by applicable staff positions, in the review of the CE System 80+ lighting system design.

Consideration should be given to adding Review Procedures and a Review Interface for review of lighting system conformance to ALARA principles.



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#### **Potential Impacts/Documents supporting the Suggested Changes:**

23094/FINAL SER CE80 CH 9; 23099/FINAL SER EPRI CH 11

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**Integrated Impact Number:** 971      **SRP Section Number:** 3.7.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria, Review Procedures and Evaluation Findings, applicable to evolutionary plants, for review of seismic system analysis.

As discussed in the ABWR and CE 80+ FSERs and in SECY 93-087, evolutionary plants may eliminate the Operating Basis Earthquake (OBE) analysis from the design basis for structures, systems and components (SSCs). In such cases, the OBE will serve as an inspection level earthquake above which the licensee would shut down the plant and inspect for potential damage to SSCs important to safety. Design certification based upon this single-earthquake design approach is predicated, in part, on the ability to ascertain the need to take action to shut down the plant in case of occurrence of the OBE, and to evaluate the performance of SSCs important to safety. In the ABWR and CE80+ FSERs, the NRC staff reviewed criteria developed by the Electric Power Research Institute (EPRI) for evaluating whether or not the plant needs to be shut down following an earthquake.

Consider revising Acceptance Criteria, Review Procedures and Evaluation Findings, applicable to evolutionary plants, for review of seismic system analysis.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23011/SECY 93-087; 23012/SECY 93-087; 24594/SECY 93-087; 24595/FINAL SER CE80 CH 1; 24596/FINAL SER ABWR CH 1

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**Integrated Impact Number:** 972      **SRP Section Number:** 5.3.2

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria, Review Procedures and Evaluation Findings for determining PWR reactor vessel acceptability under pressurized thermal shock (PTS) conditions.

Unresolved Safety Issue A-49 was initiated to assess the significance of simultaneous occurrence of a severe overcooling transient and a repressurization of a PWR primary system. The NRC resolved the pressurized thermal shock issue for operating reactors with the publication of 10 CFR 50.61 that establishes a screening criterion for the reference temperatures for nil-ductility

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transition. Licensees with plants predicted to exceed the screening criteria are required to submit safety analyses for modifications to prevent failure of the reactor vessel as a result of postulated PTS events after the screening limit has been exceeded. Regulatory Guide 1.154 outlines recommended methods to be used in performing these analyses. The Regulatory Guide specifies the expected format and content of plant-specific PTS safety analysis reports and describes acceptance criteria that the NRC staff will use in evaluating licensee analyses and proposed corrective measures.

In the CE 80+ FSER, the staff assessed the likelihood that the CE 80+ reactor vessel will remain below the PTS screening criteria. The staff concluded that CE's approach to calculating reference temperatures has enough conservatism to provide adequate assurance of fracture toughness in a PTS event for its design life.

Consider adding 10 CFR 50.61 and Regulatory Guide 1.154 to Acceptance Criteria and developing Review Procedures and Evaluation Findings related to PWR reactor vessel acceptability under PTS conditions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

7107/RG 1.154; 13451/CODE OF FED. REGS 10CFR50; 25075/RG 1.154

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**Integrated Impact Number: 973      SRP Section Number: 5.3.1**

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to allow for the use of alternatives to recommendations in Regulatory Guide 1.44 for verifying the non-sensitization of austenitic stainless steels.

SRP Section 5.3.1 lists Regulatory Guide 1.44 as providing acceptance criteria for avoiding sensitization of stainless steel. Regulatory Guide 1.44, Position C.3 states that non-sensitization of the material should be verified using ASTM A 262-1970, Practices A or E, or another method that can be demonstrated to show non-sensitization of austenitic stainless steel. ABB CE proposed the use of ASTM A 708 as an alternative to ASTM A 262. In the CE System 80+ FSER, the staff accepted the proposed use of ASTM A 708 based in part on SRP Section 4.5.1, paragraph III.2, which allows the use of ASTM 708 as an alternative to ASTM A 262 in evaluating the sensitization of austenitic stainless steel.

Consider revising the Acceptance Criteria and Review Procedures to allow the use of ASTM A 708 as an alternative to the recommendations of Regulatory Guide 1.44 to use ASTM A 262.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24181/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 974      **SRP Section Number:** 5.3.1

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria and Review Procedures to incorporate accepted alternatives to Regulatory Guide 1.50, Position C.2.

Regulatory Guide 1.50 provides the recommended procedures for control of preheat temperatures in welding of low-alloy steel to prevent cold cracking or hydrogen embrittlement of the weld or heat affected zones. As documented in the staff's FSER for the ABB CE System 80+ design, ABB CE proposed to use Westinghouse Topical Report, WCAP-8678, "Effect of Preheat and Post Weld Heat Treat on Hydrogen-Induced Cracking in Pressure Vessel Steels," as an alternative to Regulatory Guide 1.50, Position C.2. The staff has evaluated and accepted the WCAP and finds ABB CE's approach to be acceptable.

Consider revising the Acceptance Criteria and Review Procedures to include WCAP-8678 as an acceptable alternative to Regulatory Guide 1.50, Position C.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24597/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 975      **SRP Section Number:** 5.3.1

#### **Suggested Changes to the SRP Section:**

The SRP cites the following ASME Boiler and Pressure Vessel Code Article, Appendix and Paragraph:

Article NA-8000

Appendix IX-6000

Paragraph NB-2322a

In addition, ASTM A-262 1970 was added to SRP Section 5.3.1 under Integrated Impact No. 973.

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The latest version of the ASME B&PV Code is the 1992 Edition through the 1994 Addenda. The latest version does not contain the citations listed above. The latest version of ASTM A-262 is dated 1993. A side-by-side comparison has been performed for the ASTM standard but is not currently planned for the ASME citations.

Consider updating the standard citations in SRP Section 5.3.1 to cite the latest version of the ASME and ASTM standards. This is a placeholder Integrated Impact and will not be processed further.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24599/C&S: ASME III; 24600/C&S: ASTM A262-70

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**Integrated Impact Number: 976      SRP Section Number: 3.6.1**

#### **Suggested Changes to the SRP Section:**

SRP Section 3.6.1 cites IEEE 279. The citation is non-date-specific with regard to the applicable standard version.

Standard IEEE 603-1980 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of IEEE 279 to cite the IEEE 603-1980 version.

(NOTE: This ROC has been revised to reflect the NRC's approach addressing codes and standards which do not have dates specified.)

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23961/C&S: IEEE 279; 24620/RG 1.153

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**Integrated Impact Number: 978      SRP Section Number: 5.2.2**

#### **Suggested Changes to the SRP Section:**

SRP Section 5.2.2 cites IEEE 279 (no date specified) in Branch Technical Position RSB 5-2 as guidance for design of the low temperature overpressurization protection system. As stated in Regulatory Guide 1.153, the requirements and recommendations of IEEE 603-1980 incorporate the requirements and recommendations of IEEE 279-1971. Compliance with IEEE 603-1980 as supplemented by Regulatory Guide 1.153 is considered by the NRC staff to satisfy the provisions of IEEE 279-1971. The NRC staff is in the process of revising Regulatory Guide

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1.153 to update its endorsement from IEEE 603-1980 to IEEE 603-1991.

Pending revision of Regulatory Guide 1.153, supplement citations of IEEE 279- 1971 to read as follows: IEEE 279 (IEEE 603 as supplemented by Regulatory Guide 1.153) and include IEEE 603-1980 and Regulatory Guide 1.153 as references in subsection VI.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1250/RG 1.153; 23987/C&S: IEEE 279

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**Integrated Impact Number: 979      SRP Section Number: 9.2.2**

#### **Suggested Changes to the SRP Section:**

SRP Section 9.2.2 cites IEEE 279 (no date specified) as containing requirements for instrumentation related to auxiliary cooling water systems. As stated in Regulatory Guide 1.153, the requirements and recommendations of IEEE 603-1980 incorporate the requirements and recommendations of IEEE 279-1971. Compliance with IEEE 603-1980 as supplemented by Regulatory Guide 1.153 is considered by the NRC staff to satisfy the provisions of IEEE 279-1971. The NRC staff is in the process of revising Regulatory Guide 1.153 to update its endorsement from IEEE 603-1980 to IEEE 603-1991.

Pending revision of Regulatory Guide 1.153, supplement citations of IEEE 279-1971 to read as follows: IEEE 279 (IEEE 603 as supplemented by Regulatory Guide 1.153) and include IEEE 603-1980 and Regulatory Guide 1.153 as references in subsection VI.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24013/C&S: IEEE 279; 24623/RG 1.153

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**Integrated Impact Number: 985      SRP Section Number: 13.2.2**

#### **Suggested Changes to the SRP Section:**

Delete citations of NUREG-0660 TMI action plan item I.A.2.2.

NUREG-0660 TMI action plan item I.A.2.2 was developed to consider training programs including licensed and auxiliary operators, technicians, maintenance personnel and supervisors in light of the safety significance of duties performed.

As indicated in NUREG-0933, the Commission recognized that the industry had made progress

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in developing programs to improve nuclear utility training and personnel qualification. As a result, the Commission adopted a Policy Statement on Training and Qualifications (53 FR 46603) which made the training accreditation program managed by INPO the focus of training improvement in the industry. SRP Section 13.2.2 currently cites I.A.2.2 of NUREG-0660. It should also be noted the SRP Section 13.2.2 cites Regulatory Guide 1.8 and ANSI/ANS-3.1 that do provide staff positions and guidance related to this issue.

This item does not appear in NUREG-0737 Enclosures 1 or 2 (TMI items applicable to operating license holders and applicants) and was indicated not to be a licensing issue for construction permit or manufacturing license applicants in NUREG-0718. As indicated in NUREG-0933 this item was resolved with no requirements. It should be noted that this item was not addressed in the CE 80+ or ABWR evolutionary plant SERs.

Since there is no specific guidance associated with I.A.2.2 and there are no specific requirements for licensees or applicants to respond to I.A.2.2, consideration should be given to deleting the existing citations of NUREG-0660 TMI action plan item I.A.2.2 from the SRP. Regulatory Guide 1.8 and ANSI/ANS-3.1 currently cited in SRP address this issue.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25037/NUREG 0933; 25038/NRC POLICY STATEMENT 53 FR 46603

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**Integrated Impact Number:** 986      **SRP Section Number:** 13.1.2 - 13.1.3

#### **Suggested Changes to the SRP Section:**

Consider editorial changes related to TMI Action Plan item I.A.1.1 of NUREG 0737.

TMI Action Plan item I.A.1.1 of NUREG 0737 required each operating license applicant and licensee to provide an on-shift technical advisor to the shift supervisor. The Commission Policy Statement on Engineering Expertise on Shift (50 FR 43621) and Generic Letter 86-04 were subsequently issued that provide additional options for meeting the requirements of I.A.1.1 of NUREG 0737. The Policy Statement indicates that the Commission encourages licensees to move toward the dual- role (SRO/STA) position, with the eventual goal of the Shift Supervisor serving in the dual role.

TMI Action Plan item I.A.1.1 of NUREG 0737 is currently referenced in the Acceptance Criteria item II.C.3 of the SRP. The current discussion in Acceptance Criteria contains a reference to the Commission Policy statement related to this issue.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24637/NUREG 0737; 24639/NRC GENERIC LETTER 86-04; 24653/NRC POLICY STATEMENT 50 FR 43621; 24686/NUREG 0933

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**Integrated Impact Number:** 990      **SRP Section Number:** 8.3.1

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to cite 10 CFR 50.34(f)(2)(xx) related to TMI action plan item II.G.1 regarding pressurizer equipment power supplies.

TMI action plan item II.G.1 of NUREG-0737 addresses power supplies for pressurizer relief and block valves and pressurizer level indicators. This item is applicable to all operating license holders and applicants. 10 CFR 50.34(f)(2)(xx) establishes an equivalent requirement applicable to certain construction permit and manufacturing license applicants, and via 10 CFR 52.47, design certification applicants.

SRP Section 8.3.1 currently cites II.G.1 and identifies NUREG-0737 in regard to this item in Acceptance Criteria. No other guidance specifically related to this item was identified.

Consideration should be given to revising SRP Section 8.3.1 to cite the requirement 10 CFR 50.34(f)(2)(xx) related to the current citation of TMI action plan item II.G.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24752/NUREG 0737; 24926/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 996      **SRP Section Number:** 6.2.1.1.C

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to cite 10 CFR 50.34(f)(2)(xvii) related to TMI action plan item II.F.1 regarding accident monitoring instrumentation.

TMI action plan item II.F.1 of NUREG-0737 addresses instrumentation to monitor plant variables and effluents during and following an accident. II.F.1 primarily addresses instrumentation for monitoring containment parameters and post-accident plant effluent monitoring. This item is applicable to all operating license holders and applicants. 10 CFR 50.34(f)(2)(xvii) establishes an equivalent requirement applicable to certain construction permit and manufacturing license applicants, and via 10 CFR 52.47, design certification applicants. No

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other guidance specific to TMI action plan item II.F.1 was identified, however, Regulatory Guide 1.97 was revised after NUREG-0737 was promulgated and contains guidance related to the issues identified in II.F.1.

SRP Section 6.2.1.1.C currently cites II.F.1 and identifies NUREG-0737 with regard to this item. The SRP Section also identifies the guidance of Regulatory Guide 1.97 related to the issue of accident monitoring instrumentation.

Consideration should be given to citing 10 CFR 50.34(f)(2)(xvii) related to TMI action plan item II.F.1 in the SRP Section.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24746/NUREG 0737; 24918/CODE OF FED. REGS 10CFR50; 24932/RG 1.97

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**Integrated Impact Number: 997      SRP Section Number: 6.2.1.1.B**

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to cite 10 CFR 50.34(f)(2)(xvii) related to TMI action plan item II.F.1 regarding accident monitoring instrumentation.

TMI action plan item II.F.1 of NUREG-0737 addresses instrumentation to monitor plant variables and effluents during and following an accident. II.F.1 primarily addresses instrumentation for monitoring containment parameters and post-accident plant effluent monitoring. This item is applicable to all operating license holders and applicants. 10 CFR 50.34(f)(2)(xvii) establishes an equivalent requirement applicable to certain construction permit and manufacturing license applicants, and via 10 CFR 52.47, design certification applicants. No other guidance specific to TMI action plan item II.F.1 was identified, however, Regulatory Guide 1.97 was revised after NUREG-0737 was promulgated and contains guidance related to the issues identified in II.F.1.

SRP Section 6.2.1.1.B currently cites II.F.1 and identifies NUREG-0737 with regard to this item. The SRP Section also identifies the guidance of Regulatory Guide 1.97 related to the issue of accident monitoring instrumentation.

Consideration should be given to citing 10 CFR 50.34(f)(2)(xvii) related to TMI action plan item II.F.1 in the SRP Section.



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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24745/NUREG 0737; 24917/CODE OF FED. REGS 10CFR50; 24931/RG 1.97

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**Integrated Impact Number:** 998      **SRP Section Number:** 6.2.1.1.A

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to cite 10 CFR 50.34(f)(2)(xvii) related to TMI action plan item II.F.1 regarding accident monitoring instrumentation.

TMI action plan item II.F.1 of NUREG-0737 addresses instrumentation to monitor plant variables and effluents during and following an accident. II.F.1 primarily addresses instrumentation for monitoring containment parameters and post-accident plant effluent monitoring. This item is applicable to all operating license holders and applicants. 10 CFR 50.34(f)(2)(xvii) establishes an equivalent requirement applicable to certain construction permit and manufacturing license applicants, and via 10 CFR 52.47, design certification applicants. No other guidance specific to TMI action plan item II.F.1 was identified, however, Regulatory Guide 1.97 was revised after NUREG-0737 was promulgated and contains guidance related to the issues identified in II.F.1.

SRP Section 6.2.1.1.A currently cites II.F.1 and identifies NUREG-0737 with regard to this item. The SRP Section also identifies the guidance of Regulatory Guide 1.97 related to the issue of accident monitoring instrumentation.

Consideration should be given to citing 10 CFR 50.34(f)(2)(xvii) related to TMI action plan item II.F.1 in the SRP Section.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24744/NUREG 0737; 24916/CODE OF FED. REGS 10CFR50; 24930/RG 1.97

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**Integrated Impact Number:** 1003      **SRP Section Number:** 8.1

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to cite the requirement of 10 CFR 50.34(f)(2)(v) regarding the implementation of Regulatory Guide 1.47.

TMI action plan item I.D.3 of NUREG-0660 recommended that a study be undertaken to determine the need for all licensees and applicants not committed to Regulatory Guide 1.47 to

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install a bypass and inoperable status indication system or similar system such as that described in Regulatory Guide 1.47. 10 CFR 50.34(f)(2)(v), related to I.D.3, requires automatic indication of the bypassed and operable status of safety systems. This requirement is applicable to certain construction permit and manufacturing license applicants, and via 10 CFR 52.74, design certification applicants. NUREG-0718, which provides guidance related to implementation of 10 CFR 50.34(f) requirements, describes conformance with Regulatory Guide 1.47 as appropriate to address the issue described in TMI action plan item I.D.3 of NUREG-0660.

Reviewing applicant conformance with the guidance of Regulatory Guide 1.47 is currently addressed in the reviews described in the Chapter 7 and Chapter 8 sections of the SRP. SRP Section 8.1 current cites Regulatory Guide 1.47 as guidance applicable to the reviews of SRP Sections 8.2, 8.3.1, and 8.3.2.

Consideration should be given to citing the requirement of 10 CFR 50.34(f)(2)(v) in the Acceptance Criteria section of Table 8-1 of SRP Section 8.1 as criteria applicable to SRP Sections 8.2, 8.3.1, and 8.3.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24966/CODE OF FED. REGS 10CFR50; 24968/RG 1.47; 24970/NUREG 0933

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**Integrated Impact Number:** 1004      **SRP Section Number:** 15.0

#### **Suggested Changes to the SRP Section:**

Editorial change related to the citation of TMI action plan item I.C.1.

TMI action plan item I.C.1 required an analyses of transients and accidents, preparation of emergency procedure guidelines and upgrading of emergency operating procedures. The emergency operating procedures were required to be consistent with the actions necessary to cope with the transients and accidents analyzed. This item was included in NUREG-0737 as an item applicable to operating license holders and applicants and a clarification was provided in Enclosure 3. Additional criteria related to I.C.1 was issued in Section 7 of Supplement 1 to NUREG-0737.

SRP Section 15.0 currently cites I.C.1 in the phrase 'procedures developed under Action Plan item I.C.1'. SRP Section 15.0 is an introductory section and does not provide and specific guidance related to development of emergency operating procedures. The development of emergency operating procedures is addressed under SRP Section 13.5.2. Current criteria for factoring transient and accident analysis into emergency operating procedures is currently more involved than is specifically discussed under TMI action plan item I.C.1 which is a 'short-term'

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action. Recommend replacing 'procedures developed under Action Plan item I.C.1' with 'emergency operating procedures' which is clearer and more general identification of these procedures.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24696/NUREG 0737; 24697/NUREG 0737 SUPPLEMENT 1

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**Integrated Impact Number:** 1005      **SRP Section Number:** 13.1.1

#### **Suggested Changes to the SRP Section:**

Revise the current citation of TMI action plan item II.J.3.1 regarding an applicants management plan for design and construction activities.

NUREG-0660 Task II.J.3 addresses NRC staff and licensee actions to improve the qualification of licensees for operating nuclear power plants by requiring greater oversight of design, construction, and modification activities. Item II.J.3.1 of this task involves licensee submission of a description of the organization, training, and staffing proposed to accomplish these improvements.

As indicated in NUREG-0933, this item was incorporated into TMI action plan item I.B.1.1. I.B.1.1 was resolved in consideration of a number of NRC staff and industry efforts in the area of improving facility management and no requirements specific to II.J.3.1 or I.B.1.1 were established. Prior to this resolution, an application requirement, 10 CFR 50.34(f)(3)(vii), was established related to TMI action plan item II.J.3.1. This requirement is applicable to design certification applicants via 10 CFR 52.47. NUREG-0718 provides a clarification of this item and describes information to be included in an application regarding an applicants management plan for design and construction activities. SRP Section 13.1.1 currently cites TMI action plan item II.J.3.1 and the associated guidance of NUREG-0718, however, does not cite the associated application requirement of 10 CFR 50.34(f).

Consideration should be given to revising the Acceptance Criteria of SRP Section 13.1.1 to cite the requirement of 10 CFR 50.34(f)(3)(vii) related to the current citation of TMI action plan item II.J.3.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24963/CODE OF FED. REGS 10CFR50; 24964/NUREG 0933; 24965/NUREG 0933

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**Integrated Impact Number:** 1007      **SRP Section Number:** 6.4

#### **Suggested Changes to the SRP Section:**

Add citation of 10 CFR 50.34(f)(2)(xxviii) in connection with the existing citation of III.D.3.4.

TMI action plan item III.D.3.4, 'Control Room Habitability, ' is addressed in Revision Options Checklist 320 associated with SRP Section 6.4.

An application requirement related to this issue was incorporated in the 10 CFR 50.34(f) as 10 CFR 50.34(f)(2)(xxviii). This requirement is applicable to certain applicants for construction permits and manufacturing licenses, and via 10 CFR 52.47, this requirement is also applicable to design certification applicants.

Consideration should be given to incorporating a citation of the 10 CFR 50.34(f)(2)(xxviii) application requirement as part of the changes developed under Revision Options Checklist 320.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24836/NUREG 0737; 24959/NRC GENERIC LETTER 83-36; 24961/NUREG CR-5659;  
25086/CODE OF FED. REGS 10CFR50; 25088/NUREG 0933

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**Integrated Impact Number:** 1013      **SRP Section Number:** 9.3.4

#### **Suggested Changes to the SRP Section:**

Update the Acceptance Criteria of the SRP Section to reflect the requirement of 10 CFR 50.34(f)(2)(xxvi) and NUREG-0737 TMI action plan item III.D.1.1 related to leakage detection and control.

TMI action plan item III.D.1.1 addresses leakage detection and control in the design of systems outside containment that contain (or might contain) radioactive source term materials following an accident. The purpose of actions under III.D.1.1 is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. This issue was evaluated under Task III.D as described in NUREG-0660. This specific issue was identified as III.D.1.1 and was approved for implementation in NUREG-0737 as an item applicable to all operating license applicants. 10 CFR 50.34(f)(2)(xxvi) establishes an equivalent application requirement applicable to certain manufacturing license and construction permit applicants and, via 10 CFR 52, is also applicable to design certification applicants.

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The CVCS system is identified in NUREG-0737 as one systems that should address III.D.1.1 leakage detection and control criteria. TMI item III.D.1.1 is listed in specific criteria of SRP Section 9.3.4. Review Procedures describe a related review (does not specifically cite III.D.1.1, however).

Consideration should be given to revising Acceptance Criteria to cite the requirement of 10 CFR 50.34(f)(2)(xxvi). It should be noted that NUREG-0737 also describes a continuing program to maintain leakage as-low-as-practical is these systems. This aspect of III.D.1.1 is addressed in a Revision Options Checklist associated with SRP Section 13.5.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24833/NUREG 0737; 24950/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1015      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Update the Acceptance Criteria of the SRP Section to reflect the requirement of 10 CFR 50.34(f)(2)(xxvi) and NUREG-0737 TMI action plan item III.D.1.1 related to leakage detection and control.

TMI action plan item III.D.1.1 addresses leakage detection and control in the design of systems outside containment that contain (or might contain) radioactive source term materials following an accident. The purpose of actions under III.D.1.1 is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. This issue was evaluated under Task III.D as described in NUREG-0660. This specific issue was identified as III.D.1.1 and was approved for implementation in NUREG-0737 as an item applicable to all operating license applicants. 10 CFR 50.34(f)(2)(xxvi) establishes an equivalent application requirement applicable to certain manufacturing license and construction permit applicants and, via 10 CFR 52 , is also applicable to design certification applicants.

RHR systems are identified in NUREG-0737 as systems that should address III.D.1.1 leakage detection and control criteria. TMI item III.D.1.1 is cited in specific criteria and Review Procedures of SRP Section 5.4.7. However, the related requirement of 10 CFR 50.34(f) is not.

Consideration should be given to revising Acceptance Criteria to cite the requirement of 10 CFR 50.34(f)(2)(xxvi). It should be noted that NUREG-0737 also describes a continuing program to maintain leakage as-low-as-practical is these systems. This aspect of III.D.1.1 is addressed in a Revision Options Checklist associated with SRP Section 13.5.1.

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### **Integrated Impacts**

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24831/NUREG 0737; 24948/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1016      **SRP Section Number:** 5.4.6

#### **Suggested Changes to the SRP Section:**

Update the Acceptance Criteria and other portions of the SRP Section to reflect the requirement of 10 CFR 50.34(f)(2)(xxvi) and NUREG-0737 TMI action plan item III.D.1.1 related to leakage detection and control.

TMI action plan item III.D.1.1 addresses leakage detection and control in the design of systems outside containment that contain (or might contain) radioactive source term materials following an accident. The purpose of actions under III.D.1.1 is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. This issue was evaluated under Task III.D as described in NUREG-0660. This specific issue was identified as III.D.1.1 and was approved for implementation in NUREG-0737 as an item applicable to all operating license applicants. 10 CFR 50.34(f)(2)(xxvi) establishes an equivalent application requirement applicable to certain manufacturing license and construction permit applicants and, via 10 CFR 52, is also applicable to design certification applicants.

The RCIC system is identified in NUREG-0737 as one systems that should address III.D.1.1 leakage detection and control criteria. TMI item III.D.1.1 is listed in specific criteria II.7 of SRP Section 5.4.6, but the current Review Procedures do not include review of the implementation of the leakage reduction program in relation to the RCIC system.

Consideration should be given to revising Acceptance Criteria to cite the requirement of 10 CFR 50.34(f)(2)(xxvi). Consideration should also be given to revising Review Procedures to describe a design review related to leakage detection and control features of the system. It should be noted that NUREG-0737 also describes a continuing program to maintain leakage as-low-as-practical in these systems. This aspect of III.D.1.1 is addressed in a Revision Options Checklist associated with SRP Section 13.5.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24830/NUREG 0737; 24947/CODE OF FED. REGS 10CFR50

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### **Integrated Impacts**

**Integrated Impact Number:** 1017      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

Update the Acceptance Criteria and other portions of the SRP Section to reflect the requirement of 10 CFR 50.34(f)(2)(xxvi) and NUREG-0737 TMI action plan item III.D.1.1 related to leakage detection and control.

TMI action plan item III.D.1.1 addresses leakage detection and control in the design of systems outside containment that contain (or might contain) radioactive source term materials following an accident. The purpose of actions under III.D.1.1 is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. This issue was evaluated under Task III.D as described in NUREG-0660. This specific issue was identified as III.D.1.1 and was approved for implementation in NUREG-0737 as an item applicable to all operating license applicants. 10 CFR 50.34(f)(2)(xxvi) establishes an equivalent application requirement applicable to certain manufacturing license and construction permit applicants and, via 10 CFR 52, is also applicable to design certification applicants.

ECCS systems are identified in NUREG-0737 as systems that should address III.D.1.1 leakage detection and control criteria. SRP Section 6.3 currently cites III.D.1.1 in Acceptance Criteria (specific criteria) and Review Procedures.

Consideration should be given to revising Acceptance Criteria to cite the requirement of 10 CFR 50.34(f)(2)(xxvi). Specific criteria should be revised to cite III.D.1.1 of NUREG-0737 as opposed to NUREG-0694 which has been superseded (Refer to 'Further Commission Guidance for Power Reactor Operating Licenses,' 45 FR 85236 and 46 FR 15242). It should be noted that NUREG-0737 also describes a continuing program to maintain leakage as-low-as-practical in these systems. This aspect of III.D.1.1 is addressed in a Revision Options Checklist associated with SRP Section 13.5.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24829/NUREG 0737; 24946/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1018      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Delete current citation of NUREG 0718 TMI action plan item II.B.7.

The purpose of NUREG 0660 TMI action plan item II.B.7 was to establish interim hydrogen control measures for small volume (BWR Mark I and II) containment structures to address overpressurization as a consequence of hydrogen burning following a severe accident involving extensive reaction between fuel cladding and coolant. Long-term resolution of severe accident hydrogen control issues were addressed under action plan item II.B.8. SRP Section 5.4.7 currently cites TMI action plan item II.B.7 of NUREG 0718 in Areas of Review. The SRP does not currently identify any Acceptance Criteria or Review Procedures related to this item.

10 CFR 50.44 was amended to include requirements for inerting of BWR Mark I and II containments in response to this issue. Also related to this issue, additional hydrogen control requirements were added to 10 CFR 50.44 to address hydrogen control for plant designs with intermediate containment volumes (i.e., PWR ice condenser containments and BWR Mark III containments).

Changes to the SRP to reflect amendments to 10 CFR 50.44 are addressed in Revision Options Checklists associated with SRP Section 6.2.5. TMI action plan item II.B.8 resulted in the establishment of a number of requirements under 10 CFR 50.34(f) applicable to certain construction permit and manufacturing license applicants, and via 10 CFR 52.47, design certification applicants. Specifically, 10 CFR 50.34(f)(2)(ix) and 10 CFR 50.34(f)(3)(v) address severe accident hydrogen control. 10 CFR 50.34(f)(2)(ix), 10 CFR 50.34(f)(3)(v), and II.B.8 are the subject of other Revision Options Checklists.

Consideration should be given to deleting the current citation of TMI action plan item II.B.7 of NUREG 0718 in Areas of Review of SRP Section 5.4.7.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25032/NUREG 0933; 25033/NUREG 0933; 25039/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1021      **SRP Section Number:** 5.2.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria related to TMI action plan item II.D.3 regarding direct indication of relief and safety valve position.



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TMI action plan item II.D.3 of NUREG-0737 requires reactor coolant system relief and safety valves be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe. A clarification of II.D.3 is provided in Enclosure 3 of NUREG-0737. 10 CFR 50.34(f)(2)(xi) establishes an equivalent requirement applicable to certain construction permit and manufacturing license applicants, and via 10 CFR 52.47, design certification applicants. These requirements are applicable to all operating licensees and applicants. No additional guidance was identified that is directly related to II.D.3.

SRP Section 5.2.2 addresses TMI action plan item II.D.3 in Acceptance Criteria II.2 and identifies the guidance contained in NUREG-0737. SRP Section 5.2.2 does not currently cite the related requirement of 10 CFR 50.34(f)(2)(xi).

Consideration should be given to revising Acceptance Criteria associated with TMI action plan item II.D.3 to cite the related requirement of 10 CFR 50.34(f)(2)(xi) including a discussion of the applicability of this requirement.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24728/NUREG 0737; 24906/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1022      **SRP Section Number:** 13.2.1

#### **Suggested Changes to the SRP Section:**

Delete citations of NUREG-0737 TMI Action Plan item I.A.2.3.

TMI Action Plan item I.A.2.3 addresses the administration of training programs. NUREG-0660 describes this item as NRC staff action to develop criteria and procedures to be used in auditing training programs. NUREG-0660 indicated that pending accreditation of training institutions, the staff was to establish a requirement that addressed instructor qualifications.

NUREG-0737 established a requirement related to I.A.2.3 applicable to operating license applicants and holders. A clarification was provided in Enclosure 3 of NUREG-0737. The clarification describes requirements for training instructor qualifications. The clarification indicates that the position stated in NUREG-0737 was a short-term position pending the accreditation of training institutions. SRP Section 13.2.1 currently cites I.A.2.3 of NUREG-0737 in Acceptance Criteria.

Following the issuance of NUREG-0737, substantial efforts were pursued by the nuclear industry and the NRC to address improvements in training and qualifications. The industry

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establishment of INPO training program accreditation and the Commission Policy Statement of Training and Qualification of Nuclear Power Plant Personnel (53 FR 46603) specifically address the issue associated I.A.2.3 of NUREG-0660.

Since the position identified in NUREG-0737, Enclosure 3, I.A.2.3, was a interim position and the NRC position regarding the accreditation of training institutions is now addressed in the Commission Policy Statement on Training and Qualification of Nuclear Power Plant Personnel, Consideration should be given to deleting the citation of the position and guidance of I.A.2.3 in Nureg-0737 from the SRP.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24690/NUREG 0737; 24886/NUREG 0933

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**Integrated Impact Number:** 1023      **SRP Section Number:** 13.2.1

#### **Suggested Changes to the SRP Section:**

Delete citation of NUREG-0737 TMI Action Plan item I.A.3.3.

NUREG-0660 TMI Action Plan item I.A.3.3 calls for the NRC to develop a regulatory approach to: (1) provide assurance that applicants for RO and SRO licenses are psychologically fit, and (2) prohibit licensing of persons with histories of drug and alcohol abuse or criminal backgrounds.

Since the publication of NUREG-0660, the Fitness-for-Duty Rule, 10 CFR Part 26, was established. NUREG 1385, "Fitness-for-Duty in the Nuclear Power Industry: Responses to Implementation Questions," (transmitted via Generic Letter 89-23) contains guidance related to the implementation of the rule. The rule and guidance are significantly broader in scope than the actions described under NUREG-0660 item I.A.3.3 and are outside the scope of this Revision Options Checklist.

SRP Section 13.2.1 cites I.A.3.3 of NUREG-0737 in Evaluation Findings (no corresponding citations appear in Acceptance Criteria or Review Procedures. Contrary to this citation in the SRP, this item does not appear in NUREG-0737 Enclosures 1 or 2 (TMI items applicable to operating license holders and applicants). It should also be noted that this item was not addressed in the CE 80+ or ABWR evolutionary plant SERs.

Since there is not specific guidance associated with I.A.3.3 and there are not specific requirements for licensees or applicants to respond to I.A.3.3, consideration should be given to deleting the existing citation of I.A.3.3 from Evaluation Findings of the SRP.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25034/NUREG 0933; 25035/NRC GENERIC LETTER 89-23; 25040/CODE OF FED. REGS 10CFR26

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**Integrated Impact Number:** 1024      **SRP Section Number:** 13.2.1

#### **Suggested Changes to the SRP Section:**

TMI action plan item II.B.4 of NUREG-0737 requires operating license holders and applicants to develop a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged. A clarification for this item was provided in Enclosure 3 of NUREG-0737. No additional guidance specifically related to II.B.4 was identified.

SRP Section 13.2.1 cites this item and identifies NUREG-0737 as providing guidance related to this item. However, Evaluation Findings for construction permit application review refers to NUREG-0718 with regard to II.B.4. Contrary to this citation, NUREG-0718 does not identify II.B.4 as an item to be addressed by construction permit applicants and provides no positions with regard to this item. The current wording of the Evaluation Finding relates to a staff conformation that the applicant has committed to implement II.B.4 later in the licensing process. It would be appropriate at the construction permit stage to for an applicant to commit to future implementation of NUREG-0737 item II.B.4 which is applicable at the operating license stage of the process.

Consideration should be given to revising the citation of II.B.4 in Evaluation Findings for construction permit applicants to cite NUREG-0737.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24726/NUREG 0737

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**Integrated Impact Number:** 1025      **SRP Section Number:** 13.2.1

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to address current requirements related to plant simulators and requalification related to TMI action plan item I.A.4.2.

TMI action plan item I.A.4.2 of NUREG-0660 addressed a number of long-term actions to improve nuclear plant simulators and their use . NUREG-0718 item I.A.4.2, currently cited in

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SRP Section 13.2.1, addressed the provision of plant simulators for construction permit applicants. 10 CFR 50.34(f)(2)(I) establishes a requirement related to I.A.4.2 applicable to certain construction permit applicants. This requirement is also applicable to design certification applicants via 10 CFR 52.47. Since NUREG-0718 and 10 CFR 50.34(f)(2)(I) were promulgated, 10 CFR 55 was amended and regulatory requirements related to nuclear plant simulators were established under 10 CFR 55.45.

Regulatory Guide 1.149 (issued in 1987 as Revision 1) provides guidance related to the design to nuclear plant simulators and compliance with the associated regulatory requirements. Regulatory Guide 1.149 is currently cited in Acceptance Criteria of SRP Section 13.2.1.

Consideration should be given to updating Acceptance Criteria to cite the appropriate portions of 10 CFR 55.45 and 10 CFR 50.34(f)(2)(I) regarding to the provision and design of nuclear plant simulators. In view of these requirements and guidance provided in the SRP, references to NUREG-0660/NUREG-0718 action plan item I.A.4.2 are unnecessary.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24692/CODE OF FED. REGS 10CFR50; 24693/NUREG 1258; 24694/RG 1.149;  
24893/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1026      **SRP Section Number:** 13.1.1

#### **Suggested Changes to the SRP Section:**

Evaluate, and possibly delete, Acceptance Criteria associated with TMI action plan item I.B.1.2 of NUREG-0694.

TMI action plan item I.B.1.2 of NUREG-0660 required the NRC staff to evaluate the organization and management capabilities of near-term operating license applicants before license issuance. This item also identified a licensee action to establish a group that would perform independent reviews of plant operational activities and provide a capability for evaluation of operating experiences.

The staff developed draft criteria for the purpose performing the staff evaluation of near-term operating license applicant organization and management capabilities consistent with that aspect of NUREG-0660 item I.B.1.2. NUREG-0694 identifies this criteria as being contained in an NRC document titled 'Draft Criteria for Utility Management and Technical Competence,' dated February 25, 1980. NUREG-0694, item I.B.1.2, also required near-term operating license applicants to comply with the draft staff evaluation criteria and to establish the independent operation review organization also referred to in NUREG-0660 item I.B.1.2. NUREG-0694,

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item I.B.1.2, is currently cited in Acceptance Criteria (specific criteria) of SRP Section 13.1.1.

NUREG-0737 was subsequently issued and superseded NUREG-0694 (Refer to 'Further Commission Guidance for Power Reactor Operating Licenses,' 45 FR 85236 and 46 FR 15242). NUREG-0737, item I.B.1.2, was clarified in Enclosure 3 and established a requirement for operating license applicants to establish an onsite independent safety engineering group (ISEG) to perform independent reviews of plant operations. This aspect of item I.B.1.2 is the subject of a Revision Options Checklist associated with SRP Section 13.4 that specifically addresses the ISEG.

Operating license applicant compliance with the draft organization and management capability criteria referred to in NUREG-0660 and NUREG-0694 was not approved for implementation under NUREG-0737, item I.B.1.2. As indicated in NUREG-0933, the staff completed a number of evaluations of licensee organization and management capabilities per the draft criteria developed for that purpose and subsequently resolved that aspect of I.B.1.2 with no new requirements. As indicated in NUREG-0737 (item I.B.1.2) and NUREG-0933 (item I.B.1.1), long-term licensee organization and management improvements were considered under TMI action plan item I.B.1.1. As indicated in NUREG-0933 (reference I.B.1.1, Generic Issue 75, and HF6), NRC actions taken with regard improvement of licensee organization and management did not involve the evaluation criteria referred to in NUREG-0660 and NUREG-0694, item I.B.1.2. It should also be noted that, in addition to staff actions taken in connection with I.B.1.1, the NRC Inspection Manual contains a number of provisions related to assessing an applicant's organizational and management readiness to assume plant operations responsibilities during the plant construction and licensing process (e.g., Inspection Procedures 93806 and 94300).

There is no current NRC position or regulation that specifically requires an applicant to respond to draft evaluation criteria referred to in NUREG-0694 item I.B.1.2 that is cited in the SRP (it should be noted that I.B.1.2 was not an issue reviewed in either the ABWR or CE 80+ evolutionary plant FSERs). Since this criteria was not directly transmitted to licensees and long-term actions related to issue do not discuss this criteria, it isn't clear if this criteria is currently relevant to the NRC staff licensing review described in the SRP.

Consideration should be given to coordinating with the responsible plant review branch and determining if the draft evaluation criteria referred to in NUREG-0694 related to TMI action plan item I.B.1.2 is currently relevant to NRC staff licensing reviews. The SRP Section 13.1.1 citation related to I.B.1.2 of NUREG-0694 and related specific criteria should be retained or deleted accordingly.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24695/NUREG 0737

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**Integrated Impact Number:** 1027      **SRP Section Number:** 7.1

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to cite the requirement of 10 CFR 50.34(f)(2)(v) regarding the implementation of Regulatory Guide 1.47.

TMI action plan item I.D.3 of NUREG-0660 recommended that a study be undertaken to determine the need for all licensees and applicants not committed to Regulatory Guide 1.47 to install a bypass and inoperable status indication system or similar system such as that described in Regulatory Guide 1.47. 10 CFR 50.34(f)(2)(v), related to I.D.3, requires automatic indication of the bypassed and operable status of safety systems. This requirement is applicable to certain construction permit and manufacturing license applicants, and via 10 CFR 52.74, design certification applicants. NUREG-0718, which provides guidance related to implementation of 10 CFR 50.34(f) requirements, describes conformance with Regulatory Guide 1.47 as appropriate to address the issue described in TMI action plan item I.D.3 of NUREG-0660.

Reviewing applicant conformance with the guidance of Regulatory Guide 1.47 is currently addressed in the reviews described in the Chapter 7 and Chapter 8 sections of the SRP. SRP Section 7.1 current cites Regulatory Guide 1.47 as guidance applicable to the reviews of SRP Sections 7.1 through 7.5.

Consideration should be given to citing the requirement of 10 CFR 50.34(f)(2)(v) in the Acceptance Criteria section of Table 7-1 of SRP Section 7.1 as criteria applicable to SRP Sections 7.1 through 7.5.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24967/CODE OF FED. REGS 10CFR50; 24969/RG 1.47; 24971/NUREG 0933

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**Integrated Impact Number:** 1029      **SRP Section Number:** 15.2.8

#### **Suggested Changes to the SRP Section:**

Delete current citations of TMI Action Plan item II.K.2.1.

TMI Action Plan item II.K.2.1 of NUREG-0933, "Measures to Mitigate Small-Break

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Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents - Commission Orders on Babcock and Wilcox Plants - Upgrade Timeliness and Reliability of AFW System," relates to a short term action applicable to B&W licensees at the time of TMI accident. As indicated in NUREG-0660, these short term actions were transmitted via Commission Orders and were required to be completed prior to restart following the TMI accident. As indicated in NUREG-0933, actions related to this issue were completed by the appropriate licensees. Related actions applicable to other licensees and applicants are addressed under TMI Action Plan items II.E.1.1 and II.E.1.2 of NUREG-0737. The SRP currently cites II.E.1.1 and II.E.1.2 (refer to evaluations associated with these items) in addition to II.K.2.1.

Consideration should be given to deleting current citations related to TMI Action Plan item II.K.2.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25103/NUREG 0933

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**Integrated Impact Number:** 1033      **SRP Section Number:** 14.2

#### **Suggested Changes to the SRP Section:**

Add a review interface discussion related to TMI action plan item I.C.7 regarding vendor review of initial test procedures.

TMI Action plan item I.C.7 of NUREG-0660 addressed an action for near-term operating license applicants to obtain NSSS vendor review of low-power and power-ascension test and emergency procedures. The NSSS vendor review of low-power and power-ascension test procedures aspect of this issue was incorporated in NUREG-0694 under item I.C.7 and was included in NUREG-0737, Enclosure 2, as an item applicable to operating license applicants. No clarification was provided for this item in NUREG-0737, Enclosure 3. No other additional guidance specifically regarding the implementation of I.C.7 was identified. The aspect of this issue related to NSSS vendor review of emergency procedures is addressed under action plan items I.C.1 and I.C.9. Action plan items I.C.1 and I.C.9 are addressed in SRP Section 13.5.2.

The SRP does not currently cite I.C.7 of NUREG-0737. SRP Section 13.5.1 describes NSSS vendor involvement in the development of the plant initial test program. Recommended changes to SRP Section 13.5.1 to specifically cite I.C.7 is the subject of another Revision Options Checklist. Since SRP Section 13.5.1 describes an appropriate review of the issue identified in I.C.7, there does not appear to be a need to duplicate this information in SRP Section 14.2.

In the ABWR FSER, the staff verified that GE had identified TMI Item I.C.7 for COL applicant

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action and indicated that the COL applicant's proposed resolution of the issue would be reviewed.

Consideration should be given to including a review interface in SRP Section 14.2 to identify the criteria established for the initial test program procedures in SRP Section 13.5.1.

Delete "low-power and power-ascension" text from new paragraph 1 under Review Interfaces, Page 14.2-3. SRP Section 13.5.1 does not limit the NSSS vendor review to only these phases of the initial test program.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

5219/NUREG 0737; 23673/FINAL SER ABWR CH 20; 24701/NRC GENERIC LETTER 80-57; 24703/NUREG 0694

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**Integrated Impact Number:** 1034      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

Revise Areas of Review discussion associated with TMI action plan item I.C.6 regarding verification of correct performance of operating activities.

TMI action plan item I.C.6 of NUREG-0737 requires review and revision, as necessary, of procedures to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. Such a verification system may include automatic system status monitoring and human verification of operations and maintenance activities independent of the people performing the activity.

NUREG-0737 lists I.C.6 as an action item applicable to all operating license holders and applicants. NUREG-0737 provides a clarification of this item in Enclosure 3. No other guidance related specifically to I.C.6 was identified.

SRP Section 6.3 indicates in an Areas of Review discussion that this is a NUREG-0718 item and is applicable to construction permit applicants only. Contrary to this citation, NUREG-0718 does not indicate that this is a TMI action plan item to be addressed in an application for a construction permit.

Consideration should be given to replacing the current citation of NUREG-0718 with NUREG-0737 and eliminating the reference to construction permit applicants. This change would only involve a review interfaces discussion and does not affect the Acceptance Criteria or



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the review described in SRP Section 6.3.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24700/NUREG 0737

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**Integrated Impact Number:** 1035      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Revise Areas of Review discussion associated with TMI action plan item I.C.6 regarding verification of correct performance of operating activities.

TMI action plan item I.C.6 of NUREG-0737 requires review and revision, as necessary, of procedures to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. Such a verification system may include automatic system status monitoring and human verification of operations and maintenance activities independent of the people performing the activity.

NUREG-0737 lists I.C.6 as an action item applicable to all operating license holders and applicants. NUREG-0737 provides a clarification of this item in Enclosure 3. No other guidance related specifically to I.C.6 was identified.

SRP Section 5.4.7 indicates in an Areas of Review discussion that this is a NUREG-0718 item and is applicable to construction permit applicants only. Contrary to this citation, NUREG-0718 does not indicate that this is a TMI action plan item to be addressed in an application for a construction permit.

Consideration should be given to replacing the current citation of NUREG-0718 with NUREG-0737 and eliminating the reference to construction permit applicants. This change would only involve a review interfaces discussion and does not affect the Acceptance Criteria or the review described in SRP Section 5.4.7.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24699/NUREG 0737

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**Integrated Impact Number:** 1039      **SRP Section Number:** 15.2.6

#### **Suggested Changes to the SRP Section:**

Consider editorial revision related to TMI Action Plan item II.E.1.2 regarding design review of instrumentation and controls for the auxiliary feedwater system.

TMI Action Plan item II.E.1.2 of NUREG 0737 required operating license holders and applicants to implement upgrades to AFW instrumentation and controls. 10 CFR 50.34(f)(2)(xii) establishes equivalent requirements applicable to Construction Permit and Manufacturing License applicants and, via 10 CFR 52.47, Design Certification applicants. SRP Section 15.2.6 currently cites TMI Action Plan item II.E.1.2 in Acceptance Criteria.

SRP Sections 7.1, 7.3, and 7.5 address the appropriate design review of instrumentation and controls associated with auxiliary feedwater systems and cites criteria and guidance associated with II.E.1.2. It does not appear appropriate to duplicate the citation of 10 CFR 50.34(f)(2)(xii) in SRP Section 15.2.6 which is only indirectly related to the issues involved. Revising the SRP to address citation of 10 CFR 50.34(f)(2)(xii) is the subject of another Revision Options Checklists associated with SRP Section 7.1.

It may be appropriate to delete the citation of II.E.1.2 and describe any relationship between the review of SRP Section 15.2.6 and the instrumentation and control reviews of SRP Chapter 7 (including conformance with II.E.1.2) through a Review Interface. Note that a similar recommendation has been made in connection with II.E.1.1 and is the subject of another Revision Options Checklist.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24731/NUREG 0737; 24909/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1041      **SRP Section Number:** 8.3.1

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to cite 10 CFR 50.34(f)(2)(xiii) related to TMI action plan item II.E.3.1 regarding pressurizer heater power supplies.

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TMI action plan item II.E.3.1 of NUREG-0737 addresses emergency power supply for pressurizer heaters to ensure natural circulation cooling is maintained in the event of loss of offsite power. This item is applicable to all operating license holders and applicants. 10 CFR 50.34(f)(2)(xiii) establishes an equivalent requirement applicable to certain construction permit and manufacturing license applicants, and via 10 CFR 52.47, design certification applicants.

SRP Section 8.3.1 currently cites II.E.3.1 and identifies NUREG-0737 in regard to this item in Acceptance Criteria. No other guidance specifically related to this item was identified.

Consideration should be given to revising SRP Section 8.3.1 to cite the requirement 10 CFR 50.34(f)(2)(xiii) related to the current citation of TMI action plan item II.E.3.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24734/NUREG 0737; 24912/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1044      **SRP Section Number:** 15.2.8

#### **Suggested Changes to the SRP Section:**

Delete current citations of TMI Action Plan item II.K.2.8.

TMI Action Plan item II.K.2.8 of NUREG-0660, "Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents - Commission Orders on Babcock and Wilcox Plants -Continued Upgrading of AFW System," relates to an action applicable to B&W licensees at the time of TMI accident. This action was transmitted via Commission Orders to the affected B&W licensees.

As indicated in NUREG-0660, NUREG-0737, and NUREG-0933, short-term actions related to this issue were completed by B&W licensees and evaluated by the NRC staff prior to allowing restart of those plants. Long-term actions related to issue were then determined to be addressed under TMI Action Plan items II.E.1.1 and II.E.1.2 of NUREG-0737 is applicable to all operating licensees and applicants. The SRP currently cites II.E.1.1 and II.E.1.2 (refer to evaluations associated with these items in Revision Options Checklists 67 and 78) in addition to II.K.2.8.

Consideration should be given to deleting current citations related to TMI Action Plan item II.K.2.8.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25007/NUREG 0737; 25012/NUREG 0933

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**Integrated Impact Number:** 1045      **SRP Section Number:** 15.1.5

#### **Suggested Changes to the SRP Section:**

Delete current citations of TMI Action Plan item II.K.2.8.

TMI Action Plan item II.K.2.8 of NUREG 0660, 'Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents - Commission Orders on Babcock and Wilcox Plants -Continued Upgrading of AFW System,' relates to an action applicable to B&W licensees at the time of TMI accident. This action was transmitted via Commission Orders to the affected B&W licensees.

As indicated in NUREG-0660, NUREG-0737, and NUREG 0933, short-term actions related to this issue were completed by B&W licensees and evaluated by the NRC staff prior to allowing restart of those plants. Long-term actions related to issue were then determined to be addressed under TMI Action Plan items II.E.1.1 and II.E.1.2. II.E.1.1 and II.E.1.2 of NUREG 0737 is applicable to all operating licensees and applicants. The SRP currently cites II.E.1.1 and II.E.1.2 (refer to evaluations associated with these items in Revision Options Checklists 68 and 75) in addition to II.K.2.8.

Consideration should be given to deleting current citations related to TMI Action Plan item II.K.2.8.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25006/NUREG 0737; 25011/NUREG 0933

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**Integrated Impact Number:** 1046      **SRP Section Number:** 5.2.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria related to TMI action plan item II.D.1 regarding qualification of reactor coolant system relief and safety valves .

TMI action plan item II.D.1 of NUREG-0737 requires PWR and BWR licensees and applicants to conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents. A clarification of II.D.1 is

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provided in Enclosure 3 of NUREG-0737. 10 CFR 50.34(f)(2)(x) establishes an equivalent requirement applicable to certain construction permit and manufacturing license applicants, and via 10 CFR 52.47, design certification applicants.

SRP Section 5.2.2 addresses TMI action plan item II.D.1 in Acceptance Criteria II.2 and identifies the guidance contained in NUREG-0737. SRP Section 5.2.2 does not currently cite the related requirement of 10 CFR 50.34(f)(2)(x).

Consideration should be given to revising Acceptance Criteria associated with TMI action plan item II.D.1 to cite the related requirement of 10 CFR 50.34(f)(2)(x) including a discussion of the applicability of this requirement.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24727/NUREG 0737; 24905/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1047      **SRP Section Number:** 15.2.7

#### **Suggested Changes to the SRP Section:**

Consider editorial revision related to TMI Action Plan item II.E.1.2 regarding design review of instrumentation and controls for the auxiliary feedwater system.

TMI Action Plan item II.E.1.2 of NUREG 0737 required operating license holders and applicants to implement upgrades to AFW instrumentation and controls. 10 CFR 50.34(f)(2)(xii) establishes equivalent requirements applicable to Construction Permit and Manufacturing License applicants and, via 10 CFR 52.47, Design Certification applicants. SRP Section 15.2.7 currently cites TMI Action Plan item II.E.1.2 in Acceptance Criteria.

SRP Sections 7.1, 7.3, and 7.5 address the appropriate design review of instrumentation and controls associated with auxiliary feedwater systems and cites criteria and guidance associated with II.E.1.2. It does not appear appropriate to duplicate the citation of 10 CFR 50.34(f)(2)(xii) in SRP Section 15.2.7 which is only indirectly related to the issues involved. Revising the SRP to address citation of 10 CFR 50.34(f)(2)(xii) is the subject of another Revision Options Checklists associated with SRP Section 7.1.

It may be appropriate to delete the citation of II.E.1.2 and describe any relationship between the review of SRP Section 15.2.7 and the instrumentation and control reviews of SRP Chapter 7 (including conformance with II.E.1.2) through a Review Interface. Note that a similar recommendation has been made in connection with II.E.1.1 and is the subject of another Revision Options Checklist.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24732/NUREG 0737; 24910/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1048      **SRP Section Number:** 15.2.8

#### **Suggested Changes to the SRP Section:**

Consider editorial revision related to TMI Action Plan item II.E.1.1 and design review of the auxiliary feed water system.

TMI Action Plan item II.E.1.1 of NUREG-0737 required operating license holders and applicants to perform an evaluation of auxiliary feed water (AFW) systems to include: 1 ) a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques, 2) a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1 and 3) evaluate the AFW system flow rate design bases and criteria. 10 CFR 50.34(f)(1)(ii) establishes equivalent requirements applicable to Construction Permit and Manufacturing License applicants and, via 10 CFR 52.47, Design Certification applicants.

SRP Section 15.2.8 currently cites TMI Action Plan item II.E.1 in Acceptance Criteria, Review Procedures, and Evaluation Findings. In the context II.E.1 is presented, it is apparent that these citations are related to II.E.1.1. The SRP section also cites TMI action plan item II.K.2.1 that is directly related to this item. In addition to requirement of 10 CFR 50.34(f)(1)(ii) discussed above, there are other regulatory documents and positions related to the implementation of II.E.1.1.

SRP Section 10.4.9 addresses the design review of auxiliary feed water systems and addresses the detailed criteria and guidance associated with II.E.1.1. It does not appear appropriate to duplicate the citation of 10 CFR 50.34(f)(1)(ii) or other guidance related to II.E.1.1 in SRP Section 15.2.8 which is only indirectly related to the issues involved. Revision of the SRP to address citation of 10 CFR 50.34(f)(1)(ii) and other guidance related II.E.1.1 is the subject of another Revision Options Checklists associated with SRP Section 10.4.9. The related citation of II.K.2.1 is the subject of another Revision Options Checklist associated with this SRP section. Consideration should be given to revising the current citations of II.E.1 in SRP Section 15.2.8 to cite II.E.1.1. Alternately, it may be appropriate to delete the citations of II.E.1.1 and describe any relationship between the review of SRP Section 15.2.8 and the review of SRP Section 10.4.9 (including conformance with II.E.1.1) through a Review Interface.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24657/NRC GENERIC LETTER 80-33; 24662/NRC GENERIC LETTER 90-04;  
24667/NUREG 0611; 24672/NUREG 0635; 24677/CODE OF FED. REGS 10CFR50;  
24682/NUREG 0737

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**Integrated Impact Number:** 1049      **SRP Section Number:** 15.1.5

#### **Suggested Changes to the SRP Section:**

Consider editorial revision related to TMI Action Plan item II.E.1.2 regarding design review of instrumentation and controls for the auxiliary feedwater system.

TMI Action Plan item II.E.1.2 of NUREG 0737 required operating license holders and applicants to implement upgrades to AFW instrumentation and controls. 10 CFR 50.34(f)(2)(xii) establishes equivalent requirements applicable to Construction Permit and Manufacturing License applicants and, via 10 CFR 52.47, Design Certification applicants.

SRP Section 15.1.5 currently cites TMI Action Plan item II.E.1.2 in Acceptance Criteria, Review Procedures, and Evaluation Findings. The SRP section also cites TMI action plan item II.K.2.8 that is directly related to this item.

SRP Sections 7.1, 7.3, and 7.5 address the appropriate design review of instrumentation and controls associated with auxiliary feedwater systems and cites criteria and guidance associated with II.E.1.2. It does not appear appropriate to duplicate the citation of 10 CFR 50.34(f)(2)(xii) in SRP Section 15.1.5 which is only indirectly related to the issues involved. Revising the SRP to address citation of 10 CFR 50.34(f)(2)(xii) is the subject of another Revision Options Checklists associated with SRP Section 7.1. The related citation of II.K.2.8 is the subject of another Revision Options Checklist associated with this SRP section.

It may be appropriate to delete the citations of II.E.1.2 and describe any relationship between the review of SRP Section 15.1.5 and the instrumentation and control reviews of SRP Chapter 7 (including conformance with II.E.1.2) through a Review Interface. Note that a similar recommendation has been made in connection with II.E.1.1 and is the subject of another Revision Options Checklist.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24730/NUREG 0737; 24908/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1051      **SRP Section Number:** 13.2.1

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to address current requirements and criteria related to operator licensing and requalification related to TMI Action Plan item I.A.3.1.

TMI Action Plan item I.A.3.1 of NUREG-0660 called for the NRC to notify all operator license holders and applicants of the new scope of examinations and criteria for issuance of reactor operator and senior reactor operator licenses and renewal of licenses based on SECY 79-330E. Simulator examinations were to be included as part of license examinations. NUREG-0737 item I.A.3.1, currently cited in SRP Section 13.2.1, addressed a portion of this issue and established a requirement to include simulator examinations as part of operator licensing examinations for operating license holders and applicants.

Since NUREG-0737 was issued, 10 CFR 55 was amended establishing regulatory requirements related to simulators and operator licensing. NUREG-1021, "Operator Licensing Examiner Standards," Revision 7, contains the processes for licensing operators and conducting NRC initial licensing and requalification examinations and provides detailed discussion of the use of simulators in operator training, qualification, and licensing.

Since 10 CFR 55.45 now establishes regulatory requirements related to simulators and operator licensing, the current citation of NUREG-0737 item I.A.3.1 in SRP Section I.A.3.1 might be considered superfluous, however, NUREG-0737, Enclosure 3, item I.A.3.1, does not contain any detailed positions and does not conflict with more recent requirements and criteria.

Consideration should be given to updating Acceptance Criteria and Review Procedures to address current requirements and criteria related to the use of simulators in operator training, qualification, and licensing. In addition, a determination should be made during the development of this SRP content regarding whether or not to retain the existing citation of NUREG-0737 item I.A.3.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24691/NUREG 0737; 24888/NRC GENERIC LETTER 81-29; 24889/NRC ADMIN LETTER 94-05; 24890/NRC GENERIC LETTER 90-08; 24891/NRC NOTICE 93-24; 24892/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1052      **SRP Section Number:** 15.2.7

#### **Suggested Changes to the SRP Section:**

Delete current citations related to TMI Action Plan item II.K.2.1.

TMI Action Plan item II.K.2.1 of NUREG 0933, "Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents - Commission Orders on Babcock and Wilcox Plants - Upgrade Timeliness and Reliability of AFW System," relates to a short term action applicable to B&W licensees at the time of TMI accident. As indicated in NUREG 0660, these short term actions were transmitted via Commission Orders and were required to be completed prior to restart following the TMI accident. As indicated in NUREG 0933, actions related to this issue were completed by the appropriate licensees. Related actions applicable to other licensees and applicants are addressed under TMI Action Plan items II.E.1.1 and II.E.1.2 of NUREG 0737.

The SRP Section 15.2.7 currently cites II.E.1.1 and II.E.1.2 (refer to evaluations associated with these items) in addition to citations related to II.K.2.1 (i.e., II.K.2(1) and II.K.2 (C.2.1)).

Consideration should be given to deleting current citations related to TMI Action Plan item II.K.2.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25104/NUREG 0933

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**Integrated Impact Number:** 1053      **SRP Section Number:** 15.1.5

#### **Suggested Changes to the SRP Section:**

Delete current citations of TMI Action Plan item II.K.2.1.

TMI Action Plan item II.K.2.1 of NUREG 0933, 'Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents - Commission Orders on Babcock and Wilcox Plants - Upgrade Timeliness and Reliability of AFW System,' relates to a short term action applicable to B&W licensees at the time of TMI accident. As indicated in NUREG 0660, these short term actions were transmitted via Commission Orders and were required to be completed prior to restart following the TMI accident. As indicated in NUREG 0933, actions related to this issue were completed by the appropriate licensees. Related actions applicable to other licensees and applicants are addressed under TMI Action Plan items II.E.1.1 and II.E.1.2 of NUREG 0737. The SRP currently cites II.E.1.1 and II.E.1.2 (refer to evaluations associated

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with these items) in addition to II.K.2.1.

Consideration should be given to deleting current citations related to TMI Action Plan item II.K.2.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25105/NUREG 0933

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**Integrated Impact Number:** 1054      **SRP Section Number:** 15.2.7

#### **Suggested Changes to the SRP Section:**

Consider editorial revision related to TMI Action Plan item II.E.1.1 and design review of the auxiliary feed water system.

TMI Action Plan item II.E.1.1 of NUREG 0737 required operating license holders and applicants to perform an evaluation of auxiliary feed water (AFW) systems to include: 1) a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques 2) a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1, and 3) evaluate the AFW system flow rate design bases and criteria. 10 CFR 50.34(f)(1)(ii) establishes equivalent requirements applicable to Construction Permit and Manufacturing License applicants and, via 10 CFR 52.47, Design Certification applicants.

SRP Section 15.2.7 currently cites TMI Action Plan item II.E.1.1 in Acceptance Criteria. The SRP section also cites TMI action plan item II.K.2.1 that is directly related to this item. In addition to requirement of 10 CFR 50.34(f)(1)(ii) discussed above, there are other regulatory documents and positions related to the implementation of II.E.1.1.

SRP Section 10.4.9 addresses the design review of auxiliary feed water systems and addresses the detailed criteria and guidance associated with II.E.1.1. It does not appear appropriate to duplicate the citation of 10 CFR 50.34(f)(1)(ii) or other guidance related to II.E.1.1 in SRP Section 15.2.7 which is only indirectly related to the issues involved. Revising the SRP to address citation of 10 CFR 50.34(f)(1)(ii) and other guidance related II.E.1.1 is the subject of another Revision Options Checklists associated with SRP Section 10.4.9. The related citation of II.K.2.1 is the subject of another Revision Options Checklist associated with this SRP section.

It may be appropriate to delete the citations of II.E.1.1 and describe any relationship between the review of SRP Section 15.2.7 and the review of SRP Section 10.4.9 (including conformance with II.E.1.1) through a Review Interface.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24660/NRC GENERIC LETTER 80-33; 24665/NRC GENERIC LETTER 90-04;  
24670/NUREG 0611; 24675/NUREG 0635; 24680/CODE OF FED. REGS 10CFR50;  
24685/NUREG 0737

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**Integrated Impact Number:** 1055      **SRP Section Number:** 15.2.6

#### **Suggested Changes to the SRP Section:**

Consider editorial revision related to TMI Action Plan item II.E.1.1 and design review of the auxiliary feedwater system.

TMI Action Plan item II.E.1.1 of NUREG 0737 required operating license holders and applicants to perform an evaluation of auxiliary feedwater (AFW) systems to include: 1) a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques, 2) a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1, and 3) evaluate the AFW system flowrate design bases and criteria. 10 CFR 50.34(f)(1)(ii) establishes equivalent requirements applicable to Construction Permit and Manufacturing License applicants and, via 10 CFR 52.47, Design Certification applicants.

SRP Section 15.2.6 currently cites TMI Action Plan item II.E.1.1 in Acceptance Criteria. The SRP section also cites TMI action plan item II.K.2.1 that is directly related to this item. In addition to requirement of 10 CFR 50.34(f)(1)(ii) discussed above, there are other regulatory documents and positions related to the implementation of II.E.1.1.

SRP Section 10.4.9 addresses the design review of auxiliary feedwater systems and addresses the detailed criteria and guidance associated with II.E.1.1. It does not appear appropriate to duplicate the citation of 10 CFR 50.34(f)(1)(ii) or other guidance related to II.E.1.1 in SRP Section 15.2.6 which is only indirectly related to the issues involved. Revising the SRP to address citation of 10 CFR 50.34(f)(1)(ii) and other guidance related II.E.1.1 is the subject of another Revision Options Checklists associated with SRP Section 10.4.9. The related citation of II.K.2.1 is the subject of another Revision Options Checklist associated with this SRP section.

It may be appropriate to delete the citations of II.E.1.1 and describe any relationship between the review of SRP Section 15.2.6 and the review of SRP Section 10.4.9 (including conformance with II.E.1.1) through a Review Interface.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24659/NRC GENERIC LETTER 80-33; 24664/NRC GENERIC LETTER 90-04;  
24669/NUREG 0611; 24674/NUREG 0635; 24679/CODE OF FED. REGS 10CFR50;  
24684/NUREG 0737

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**Integrated Impact Number:** 1056      **SRP Section Number:** 15.1.5

#### **Suggested Changes to the SRP Section:**

Consider editorial revision related to TMI Action Plan item II.E.1.1 and design review of the auxiliary feedwater system.

TMI Action Plan item II.E.1.1 of NUREG 0737 required operating license holders and applicants to perform an evaluation of auxiliary feedwater (AFW) systems to include: 1) a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques, 2) a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1, and 3) evaluate the AFW system flowrate design bases and criteria. 10 CFR 50.34(f)(1)(ii) establishes equivalent requirements applicable to Construction Permit and Manufacturing License applicants and, via 10 CFR 52.47, Design Certification applicants.

SRP Section 15.1.5 currently cites TMI Action Plan item II.E.1 in Review Procedures and Evaluation Findings. In the context II.E.1 is presented, it is apparent that these citations are related to II.E.1.1. The SRP section also cites TMI action plan item II.K.2.1 that is directly related to this item. In addition to requirement of 10 CFR 50.34(f)(1)(ii) discussed above, there are other regulatory documents and positions related to the implementation of II.E.1.1.

SRP Section 10.4.9 addresses the design review of auxiliary feedwater systems and addresses the detailed criteria and guidance associated with II.E.1.1. It does not appear appropriate to duplicate the citation of 10 CFR 50.34(f)(1)(ii) or other guidance related to II.E.1.1 in SRP Section 15.1.5 which is only indirectly related to the issues involved. Revising the SRP to address citation of 10 CFR 50.34(f)(1)(ii) and other guidance related II.E.1.1 is the subject of another Revision Options Checklists associated with SRP Section 10.4.9. The related citation of II.K.2.1 is the subject of another Revision Options Checklist associated with this SRP section.

Consideration should be given to revising the current citations of II.E.1 in SRP Section 15.1.5 to cite II.E.1.1. Alternately, it may be appropriate to delete the citations of II.E.1.1 and describe any relationship between the review of SRP Section 15.1.5 and the review of SRP Section 10.4.9 (including conformance with II.E.1.1) through a Review Interface.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24658/NRC GENERIC LETTER 80-33; 24663/NRC GENERIC LETTER 90-04;  
24668/NUREG 0611; 24673/NUREG 0635; 24678/CODE OF FED. REGS 10CFR50;  
24683/NUREG 0737

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**Integrated Impact Number:** 1057      **SRP Section Number:** 15.2.8

#### **Suggested Changes to the SRP Section:**

Consider editorial revision related to TMI Action Plan item II.E.1.2 regarding design review of instrumentation and controls for the auxiliary feed water system.

TMI Action Plan item II.E.1.2 of NUREG-0737 required operating license holders and applicants to implement upgrades to AFW instrumentation and controls. 10 CFR 50.34(f)(2)(xii) establishes equivalent requirements applicable to Construction Permit and Manufacturing License applicants and, via 10 CFR 52.47, Design Certification applicants.

SRP Section 15.2.8 currently cites TMI Action Plan item II.E.1.2 in Acceptance Criteria, Review Procedures, and Evaluation Findings. The SRP section also cites TMI action plan item II.K.2.8 that is directly related to this item.

SRP Sections 7.1, 7.3, and 7.5 address the appropriate design review of instrumentation and controls associated with auxiliary feed water systems and cite criteria and guidance associated with II.E.1.2. It does not appear appropriate to duplicate the citation of 10 CFR 50.34(f)(2)(xii) in SRP Section 15.2.8 which is only indirectly related to the issues involved. Revising the SRP to address citation of 10 CFR 50.34(f)(2)(xii) is the subject of another Revision Options Checklist associated with SRP Section 7.1. The related citation of II.K.2.8 is the subject of another Revision Options Checklist associated with this SRP section.

It may be appropriate to delete the citations of II.E.1.2 and describe any relationship between the review of SRP Section 15.2.8 and the instrumentation and control reviews of SRP Chapter 7 (including conformance with II.E.1.2) through a Review Interface. Note that a similar recommendation has been made in connection with II.E.1.1 and is the subject of another Revision Options Checklist.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24733/NUREG 0737; 24911/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1058      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to reflect the requirements of 50.34(f)(3)(ii) to include all structures, systems, and components important to safety.

NUREG 0660 TMI Action Plan item I.F.1 was established to address the expansion of the QA list required by 10 CFR 50, Appendix B. The intent of this issue was to identify those systems, structures, and components beyond those labeled 'safety-related,' prioritize their importance to safety, and prepare a generic QA list that included these items. As indicated in NUREG 0933, a proposed rule on Important To Safety (ITS) was presented in SECY-85-119 and was later disapproved by the Commission who concluded that a specific listing of ITS equipment was not required to be maintained. Thus, the issue of an expansion of the QA list to cover ITS equipment was considered closed. Generic Letter 84-01 related to issue indicates that SSCs important to safety would be included in the QA list 'in specific situations.'

Prior the resolution of I.F.1, the CFRs were revised to incorporate additional TMI-related requirements applicable to certain applicants for construction permits and manufacturing licenses. 10 CFR 50.34(f)(3)(ii) requires that the QA list required by 10 CFR 50, Appendix B, include all structures, systems, and components (SSCs) important to safety. The requirement is also applicable to Design Certification applicants via 10 CFR 52.47(a)(I)(ii).

The SRP currently describes criteria for including SSCs in the QA list required by 10 CFR 50, Appendix B in SRP Sections 3.2.1, 17.1, and 17.3. SRP Section 3.2.1 indicates that the appropriate scope of components to be included in the list are those SSCs that fall within the criteria of Seismic Category 1 components as determined by applying the classification criteria of Regulatory Guide 1.29. Regulatory Guide 1.29 criteria for Seismic Category 1 component is based on 10 CFR 100 and establishes a scope of components referred to as 'safety-related'. SRP Section 3.2.1 does not specifically address criteria for SSCs important to safety that should be included in the QA list. SRP Section 17.1 Acceptance Criteria 2A1a refers, via footnote, to the introduction of 10 CFR 50, Appendix B and to Regulatory Guide 1.29 for guidance regarding the QA list. This would appear to imply that all SSCs classified under Regulatory Guide 1.29 criteria (positions C.1 and C.2), as opposed to only Seismic Category 1 components, should be included in the QA list. The footnote also identifies rulemaking in progress related to this issue, that as described above, appears to be no longer the case.

Consideration should be given to revising SRP Section 17.1 Acceptance Criteria to identify the requirements of 10 CFR 50.34(f)(3)(ii) for applicants subject to the requirements of 10 CFR 50.45(f).

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

23044/SECY 90-016; 24852/NUREG 0933; 24855/NRC GENERIC LETTER 84-01;  
24858/FINAL SER ABWR CH 20; 24861/FINAL SER ABWR CH 20; 24864/CODE OF  
FED. REGS 10CFR50

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**Integrated Impact Number:** 1059      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Delete Acceptance Criteria and Review Procedures related to interim degraded core accident rulemaking.

NUREG 0660 TMI Action Plan Item II.B.8 addresses rulemaking related to establishing additional requirements to address degraded core accidents (now generally referred to as severe accidents). As indicated in NUREG 0933 and the Commission Policy Statement on severe accidents (50 FR 32138), the interim degraded core accident rulemaking that was proposed under II.B.8 will not be pursued further. The policy statement indicates that future rulemakings regarding severe accident criteria are anticipated.

TMI-related issue II.B.8 was subsequently resolved in NUREG 0933 in consideration of a number of on-going regulatory activities and with regulatory actions taken associated with severe accident issues such as changes to 10 CFR 50.44 (related to hydrogen control) and the Commission Policy Statement on severe accidents (50 FR 32138). Also, requirements related to specific severe accident issues were established for certain applicants under 50.34(f)(1)(i), 50.34(f)(2)(ix), 50.34(f)(3)(iv), and 50.34(f)(3)(v).

SRP Section 5.4.7 directs a review of construction permits to establish the degree to which the proposed design conforms to the proposed interim rule on degraded core accidents.

Consideration should be given to removing references to the proposed rulemaking on degraded core accidents currently cited in the SRP, consistent with the current status of the rulemaking.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24867/SECY 93-087; 24870/NRC GENERIC LETTER 88-20; 24873/NRC POLICY  
STATEMENT 50 FR 32138; 24876/NUREG 0933

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**Integrated Impact Number:** 1062      **SRP Section Number:** 5.4.6

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to incorporate TMI action plan item II.K.3.13 regarding reporting of safety and relief valve failures and challenges.

NUREG-0660 Task II.K.3 was established to track the status of issues which were based on recommendations developed by the NRC staff Bulletins and Task Force. The task force conducted four design specific (Westinghouse, CE, B&W, and GE) reviews of loss-of-feedwater and small break loss-of-coolant accidents. Item II.K.3.13, applicable to BWRs, addressed the staff position that initiation levels of the HPCI and RCIC system should be separated so that the RCIC system initiates at a higher water level than the HPCI system. Further, the initiation logic of the RCIC system should be modified so that the RCIC system will restart on low water level. A clarification was provided for this item in Enclosure 3 of NUREG-0737. Generic Letter 83-02 provides a description of the staff's position on this item and identifies appropriate Technical Specifications to address this issue. The position described in the generic letter appears to be equivalent to the position described in NUREG-0737.

An application requirement related to this issue was incorporated in the 10 CFR 50.34(f) as 10 CFR 50.34(f)(1)(v). This requirement is applicable to certain BWR applicants for construction permits. Via 10 CFR 52.47, this requirement is also applicable to design certification applicants, if technically relevant to the design.

SRP Section 5.4.6 currently identifies II.K.3.13 in acceptance criteria and Review Procedures, however, the application requirement of 10 CFR 50.34(f)(1)(v) is not currently cited.

Consideration should be given to including a citation of the 10 CFR 50.34(f)(1)(v) applicable to BWRs in connection with the current citation of II.K.3.13.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24823/NUREG 0737; 25052/NUREG 0933; 25078/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1065      **SRP Section Number:** 17.2

#### **Suggested Changes to the SRP Section:**

Perform additional analysis regarding amendments to 10 CFR 21 and 10 CFR 50.55(e).

NUREG 0660 TMI Action Plan Item II.J.4.1 called for the NRC to revise, as necessary, the



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event-reporting requirements of 10 CFR 21 to assure that all reportable items are reported promptly and that the information submitted is complete. Improvements were to be implemented by rule changes as appropriate. As indicated in NUREG 0933, this issue was resolved with new requirements when amendments to 10 CFR 21 and 10 CFR 50.55(e) were issued. These rule changes occurred in July 1991 and appear in the Federal Register at 56 FR 36081.

Chapter 17 of the SRP currently cites 10 CFR 50.55(e) with regard to reporting requirement. TMI Action Plan item II.J.4.1 should not be recommended for incorporation in the SRP since this issue related to a staff action and there is no information associated with II.J.4.1 applicable to licensing reviews.

Consideration should be given to performing further analysis to determine what, if any, impact of these rule changes have on the current review criteria of SRP Section 17.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24851/NUREG 0933

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<b>Integrated Impact Number:</b> 1067	<b>SRP Section Number:</b> 6.3
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#### **Suggested Changes to the SRP Section:**

Delete the existing citation of TMI action plan item II.K.3.10

NUREG-0660 Task II.K.3 was established to track the status of issues which were based on recommendations developed by the NRC staff Bulletins and Task Force. The task force conducted four design specific (Westinghouse, CE, B&W, and GE) reviews of loss-of-feedwater and small break loss-of-coolant accidents. Item II.K.3.10 addresses anticipatory turbine trip modification proposed by some licensees to confine a reactor trip on turbine trip to high-power levels. The staff indicated that this modification should not be made until it has been shown on a plant-by-plant basis that the probability of a small-break loss-of-coolant accident (LOCA) resulting from a stuck-open power-operated relief valve (PORV) is substantially unaffected by the modification. II.K.3.10 was indicated as applicable to all Westinghouse operating license applicants and holders in NUREG-0737. A clarification was provided for this item in Enclosure 3 of NUREG-0737. No other guidance specific to II.K.3.10 was identified.

SRP Section 6.3 currently cites II.K.3.10 of NUREG-0660. This issue appears to have been resolved by Westinghouse licensees. NUREG-0830 and NUREG-0831 (Callaway and Wolf Creek SERs) indicate that Westinghouse provided the necessary analysis and showed that PORV challenges from turbine trips during startup are not substantially affected by the proposed

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interlock modification. The issue of appropriate interlocks for anticipatory trips is addressed by plant Technical Specifications. This item was in response to a Westinghouse design specific proposal that was being considered at the time of the TMI accident and it isn't clear that this item is relevant to SRP reviews and appears to be even less relevant to the ECCS review of SRP Section 6.3. This item is also cited in SRP Section 7.1 as criteria related to the review of the reactor trip system. The citation in SRP Section 7.1 appears to be a more appropriate location for this item.

Consideration should be given to deleting the current citation of TMI action plan item II.K.3.10 from SRP Section 6.3.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24822/NUREG 0737; 25051/NUREG 0933

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**Integrated Impact Number:** 1068      **SRP Section Number:** 17.3

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to reflect the requirements of 50.34(f)(3)(ii) to include all structures, systems, and components important to safety.

NUREG 0660 TMI Action Plan item I.F.1 was established to address the expansion of the QA list required by 10 CFR 50, Appendix B. The intent of this issue was to identify those systems, structures, and components beyond those labeled 'safety-related,' prioritize their importance to safety, and prepare a generic QA list that included these items. As indicated in NUREG 0933, a proposed rule on Important To Safety (ITS) was presented in SECY-85-119 and was later disapproved by the Commission who concluded that a specific listing of ITS equipment was not required to be maintained. Thus, the issue of an expansion of the QA list to cover ITS equipment was considered closed. Generic Letter 84-01 related to this issue indicates that SSCs important to safety would be included in the QA list 'in specific situations.'

Prior the resolution of I.F.1, the CFRs were revised to incorporate additional TMI-related requirements applicable to certain applicants for construction permits and manufacturing licenses. 10 CFR 50.34(f)(3)(ii) requires that the QA list required by 10 CFR 50, Appendix B, include all structures, systems, and components (SSCs) important to safety. The requirement is also applicable to Design Certification applicants via 10 CFR 52.47(a)(I)(ii).

The SRP currently describes criteria for including SSCs in the QA list required by 10 CFR 50, Appendix B in SRP Sections 3.2.1, 17.1, and 17.3. SRP Section 3.2.1 indicates that the appropriate scope of components to be included in the list are those SSCs that fall within the

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criteria of Seismic Category 1 components as determined by applying the classification criteria of Regulatory Guide 1.29. Regulatory Guide 1.29 criteria for Seismic Category 1 component is based on 10 CFR 100 and establishes a scope of components referred to as 'safety-related'. SRP Section 3.2.1 does not specifically address criteria for SSCs important to safety that should be included in the QA list. SRP Section 17.3 Acceptance Criteria 7.b refers to Regulatory Guides 1.26 and 1.29 for guidance regarding the identification of those items to be addressed by appropriate quality assurance requirements. This would appear to imply that all SSCs classified under Regulatory Guide 1.29 criteria (positions C.1 and C.2) and Regulatory Guide 1.26,, as opposed to only Seismic Category 1 components, should be included in the QA list.

Consideration should be given to revising SRP Section 17.3 Acceptance Criteria to identify the requirements of 10 CFR 50.34(f)(3)(ii) for applicants subject to the requirements of 10 CFR 50.45(f).

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24853/NUREG 0933; 24856/NRC GENERIC LETTER 84-01; 24859/FINAL SER ABWR CH 20; 24862/FINAL SER CE80 CH 20; 24865/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1070      **SRP Section Number:** 15.0

#### **Suggested Changes to the SRP Section:**

Revise SRP Section to incorporate TMI action plan item II.K.3.7 of NUREG-0737 regarding documentation that PORVs will open in less than 5% of all anticipated overpressure transients for B&W plants.

NUREG-0660 Task II.K.3 was established to track the status of issues which were based on recommendations developed by the NRC staff Bulletins and Task Force. The task force conducted four design specific (Westinghouse, CE, B&W, and GE) reviews of loss-of-feedwater and small break loss-of-coolant accidents. Item II.K.3.7 addresses documentation that PORVs will open in less than 5% of all anticipated overpressure transients. II.K.3.7 was indicated as applicable to all B&W operating license applicants and holders in NUREG-0737. A clarification was provided for this item in Enclosure 3 of NUREG-0737. No other guidance specific to II.K.3.7 was identified.

Since II.K.3.17 of NUREG-0737 is an item applicable to B&W plants, it would seem appropriate for the SRP to cite this item, however, the SRP does not currently cite II.K.3.17. The issue would not relate to any specific transient analysis section of Chapter 15. Section 15.0 current cites II.K.3.44 as a generic BWR transient analysis consideration and in may be appropriate to incorporate II.K.3.7 in Section 15.0 in a similar manner.

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Consideration should be given to describing TMI action plan item II.K.3.7 in SRP Section 15 as a transient analysis issue applicable to B&W reactors.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24821/NUREG 0737; 25050/NUREG 0933

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**Integrated Impact Number:** 1075      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Perform additional analysis regarding amendments to 10 CFR 21 and 10 CFR 50.55(e).

NUREG 0660 TMI Action Plan Item II.J.4.1 called for the NRC to revise, as necessary, the event-reporting requirements of 10 CFR 21 to assure that all reportable items are reported promptly and that the information submitted is complete. Improvements were to be implemented by rule changes as appropriate. As indicated in NUREG 0933, this issue was resolved with new requirements when amendments to 10 CFR 21 and 10 CFR 50.55(e) were issued. These rule changes occurred in July 1991 and appear in the Federal Register at 56 FR 36081.

Chapter 17 of the SRP currently cites 10 CFR 50.55(e) with regard to reporting requirement. TMI Action Plan item II.J.4.1 should not be recommended for incorporation in the SRP since this issue related to a staff action and there is no information associated with II.J.4.1 applicable to licensing reviews.

Consideration should be given to performing further analysis to determine what, if any, impact of these rule changes have on the current review criteria of SRP Section 17.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24850/NUREG 0933

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**Integrated Impact Number:** 1076      **SRP Section Number:** 15.0

#### **Suggested Changes to the SRP Section:**

Revise SRP Section 15.0 regarding NUREG-0737 TMI action plan item II.K.3.44 to reflect the generic resolution of this issue.

NUREG-0660 Task II.K.3 was established to track the status of issues which were based on

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recommendations developed by the NRC staff Bulletins and Task Force. The task force conducted four design specific (Westinghouse, CE, B&W, and GE) reviews of loss-of-feedwater and small break loss-of-coolant accidents. Item II.K.3.44, applicable to BWRs, addressed a demonstration by applicants that for anticipated transients combined with the worst single failure, and assuming proper operator actions, the core remains covered or provide analysis to show that no significant fuel damage results from core uncover. A clarification was provided for this item in Enclosure 3 of NUREG-0737.

SRP Section 15.0 currently identifies II.K.3.44 and NUREG-0737.

Generic Letter 81-32 documents the staff's finding that the BWRs owners group (BWROG) generic response to this issue was acceptable, but required plant-specific verification of the assumptions contained in the response. The ABWR FSER provides a description of how this issue was addressed for the ABWR and identifies the generic BWROG report involved as NEDO-24708.

Consideration should be given to revising SRP Section 15.0 to address the acceptable generic resolution of NUREG-0737 item II.K.3.44.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24828/NUREG 0737; 25062/NUREG 0933; 25084/NRC GENERIC LETTER 81-32;  
25085/FINAL SER ABWR CH 20

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**Integrated Impact Number:** 1079      **SRP Section Number:** 13.2.2

#### **Suggested Changes to the SRP Section:**

Modify Acceptance Criteria related to TMI Action Plan Item I.A.1.1 of NUREG-0737 to reflect current NRC policy.

TMI Action Plan item I.A.1.1 of NUREG-0737 required each operating license holder and applicant to provide an on-shift technical advisor to the shift supervisor. The Commission Policy Statement on Engineering Expertise on Shift (50 FR 43621) and Generic Letter 86-04 were subsequently issued that provide additional options for meeting the requirements of I.A.1.1 of NUREG-0737. The Policy Statement indicates that the Commission encourages licensees to move toward the dual-role (SRO/STA) position, with the eventual goal of the Shift Supervisor serving in the dual role.

TMI Action Plan item I.A.1.1 of NUREG-0737 is currently referenced in the Acceptance Criteria (Specific Criteria) item 6 of the SRP. The current discussion in Acceptance Criteria

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indicates that the STA position is a required plant position.

Consideration should be given to modifying Acceptance Criteria consistent with current NRC policy regarding the Shift Technical Advisor position.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24638/NUREG 0737; 24640/NRC GENERIC LETTER 86-04; 24654/NRC POLICY STATEMENT 50 FR 43621; 24687/NUREG 0933

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**Integrated Impact Number:** 1080      **SRP Section Number:** 15.1.5

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria associated with TMI action plan items II.K.3.25 and II.K.2.16.

NUREG-0660 TMI action plan item II.K.3.25 addressed the effect of a loss of ac power on recirculation pump seals for BWR plants. NUREG-0737 extended the applicability of this requirement to include all operating license holders and applicants except those with B&W plants. B&W plant licensees and applicants were found to be addressing this issue under II.K.2.16. Enclosure 3 of NUREG-0737 provides a clarification for II.K.3.25 that required licensees to determine, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. This clarification also indicates that the pump seals should be designed to withstand a complete loss of all ac power for at least 2 hours.

The clarification in NUREG-0737 for the related II.K.2.16 item (B&W plants) requires an evaluation of the impact of reactor coolant pump seal damage and leakage due to loss-of-seal cooling upon loss of offsite power. If damage cannot be precluded, licensees should provide an analysis of the limiting small-break loss-of-coolant accident (LOCA) with subsequent reactor coolant pump (RCP) seal damage.

An application requirement related to II.K.3.25 and II.K.2.16 was incorporated in 10 CFR 50.34(f) as 10 CFR 50.34(f)(1)(iii). This requirement is applicable to certain specified applicants for construction permits and manufacturing licenses. Via 10 CFR 52.47, this requirement is also applicable to design certification applicants.

SRP Section 15.1.5 currently cites II.K.3.25, II.K.2.16, and II.K.3.40 with regard to this issue and identifies NUREG-0737 as guidance related to this item. NUREG-0694 is also cited as guidance with regard to this item

Consideration should be given to revising Acceptance Criteria to eliminate the reference to

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NUREG-0694 since the positions of NUREG-0737 supersede NUREG-0694. The citation of II.K.3.40 is addressed under a separate Revision Options Checklist. A citation of 10 CFR 50.34(f)(1)(iii) is being recommended for SRP Section 9.2.2 under a separate Revision Options Checklist since it would appear that this review would only provide a finding of pump seal integrity in the context of the specific analysis described in SRP Section 15.1.1.

It should be noted that there are a substantial number of issues associated with reactor coolant pump seal integrity (e.g, GI-23, GI-65, GI-130, Generic Letter 91-07 Generic Letter 91-13, station blackout). These issues are addressed in Revision Options Checklists associated with other SRP sections and do not appear to be specifically relevant to SRP Section 15.1.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24758/NUREG 0737; 24762/NUREG 0737; 24995/FINAL SER ABWR CH 20;  
25058/NUREG 0933; 25091/FINAL SER ABWR CH 20; 25095/CODE OF FED. REGS  
10CFR50

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**Integrated Impact Number:** 1082      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Delete current Acceptance Criteria and Review Procedures related to TMI action plan items II.E.3.2 and II.E.3.3.

Task II.E.3 of NUREG-0660 had the objective of improving the reliability and capability of decay heat removal systems. Item II.E.3.2 and II.E.3.3 are related to NRC staff and ACRS studies to address decay heat removal systems reliability and to perform a coordinated study of shutdown heat removal requirements. There were no licensee actions associated with these items in NUREG-0660. II.E.3.2 and II.E.3.3 were not identified in NUREG-0737 or NUREG-0718 as issues to be addressed by license applicants. As indicated in NUREG-0933 these actions were subsumed by Generic Safety Issue A-45. Generic Issue A-45 was resolved with actions taken in regard to Individual Plant Examination (e.g., Generic Letter 88-20). There are a number of issues related to the concerns being addressed under II.E.3.2 and II.E.3.3. These issues involve reduced inventory operations (Generic Letter 88-17) and shutdown operations that are subject of Revision Options Checklists for SRP Section 5.4.7.

Since there are no requirements or guidance specifically associated with TMI action plan items II.E.3.2 and II.E.3.3, these activities were subsumed by other activities, and there is no requirement for applicants to respond to these items, consideration should be given to deleting these items from SRP Section 5.4.7.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25107/NUREG 0933

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**Integrated Impact Number:** 1083      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Modify Acceptance Criteria to include identification of the requirements of 50.34(f)(3)(iii).

NUREG 0660 TMI Action Plant item I.F.2 addressed the improvement of the QA program for design, construction, and operations to provide greater assurance that plant design, construction, and operational activities are conducted in a manner commensurate with their importance to safety. As indicated in NUREG 0933, this item was resolved with revisions to Chapter 17 of the SRP with the exception of a portion of the issue that addresses management acceptance of QA programs. That portion of the issue is designated as a low priority issue. As a result, the SRP currently reflects the existing regulatory positions that are the subject of this item.

The CFRs were amended to add 10 CFR 50.34(f) requirements to address TMI-related issues applicable to certain CP and ML applicants. 10 CFR 52.47(a)(1)(ii) requires that relevant 10 CFR 50.34(f) requirements be addressed by Design Certification applicants. 10 CFR 50.34(f)(3)(iii) contains specific requirements that are related to NUREG 0660 TMI Action Plant item I.F.2.

Consideration should be given to revising Acceptance Criteria of the SRP to reflect the requirements of 10 CFR 50.34(f)(3)(iii).

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24839/CODE OF FED. REGS 10CFR50; 24841/FINAL SER ABWR CH 20; 24843/FINAL SER CE80 CH 20; 24845/NUREG 0933

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**Integrated Impact Number:** 1086      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

NUREG-0737 TMI action plan item, II.K.3.45, 'Evaluate Depressurization with Other Than Full ADS,' is addressed in Revision Options Checklist 594 associated with SRP Section 6.3 which identifies new SRP development to be performed under IPD 7.0 Form, 6.3-1.

An application requirement related to this issue was incorporated in the 10 CFR 50.34(f) as 10



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CFR 50.34(f)(1)(xi). This requirement is applicable to certain BWR applicants for construction permits. Via 10 CFR 52.47, this requirement is also applicable to design certification applicants.

The new SRP section development work identified in IPD 6.3-1 should incorporate a citation of 10 CFR 50.34(f)(1)(xi) in connection with II.K.3.45 of NUREG-0737.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25063/NUREG 0933; 25083/CODE OF FED. REGS 10CFR50; 25595/FINAL SER ABWR CH 20

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**Integrated Impact Number:** 1090      **SRP Section Number:** 5.4.6

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria, Specific Criteria, to cite 10 CFR 50.34(f)(1)(ix) in connection with TMI action plan item II.K.3.24.

NUREG-0660 Task II.K.3 was established to track the status of issues which were based on recommendations developed by the NRC staff Bulletins and Task Force. The task force conducted four design specific (Westinghouse, CE, B&W, and GE) reviews of loss-of-feedwater and small break loss-of-coolant accidents. Item II.K.3.24 addresses a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least two (2) hours. This item was designated as applicable to all BWR operating license applicants and holders in NUREG-0737 and a clarification was provided for this item in Enclosure 3 of NUREG-0737. No other positions specifically related to II.K.3.24 were identified.

An application requirement related to this issue was incorporated in the 10 CFR 50.34(f) as 10 CFR 50.34(f)(1)(ix). This requirement is applicable to certain BWR applicants for construction permits. Via 10 CFR 52.47, this requirement is also applicable to design certification applicants.

SRP Section 5.4.6 currently cites II.K.3.24 and NUREG-0737, however does not cite the related application requirement of 10 CFR 50.34(f)(1)(ix).

Consideration should be given to revising Acceptance Criteria to cite 10 CFR 50.34(f)(1)(ix) in connection with TMI action plan item II.K.3.24 of NUREG-0737.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25056/NUREG 0933; 25081/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1092      **SRP Section Number:** 15.2.6

#### **Suggested Changes to the SRP Section:**

Delete current citations related to TMI Action Plan item II.K.2.1.

TMI Action Plan item II.K.2.1 of NUREG 0933, 'Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents - Commission Orders on Babcock and Wilcox Plants - Upgrade Timeliness and Reliability of AFW System,' relates to a short term action applicable to B&W licensees at the time of TMI accident. As indicated in NUREG 0660, these short term actions were transmitted via Commission Orders and were required to be completed prior to restart following the TMI accident. As indicated in NUREG 0933, actions related to this issue were completed by the appropriate licensees. Related actions applicable to other licensees and applicants are addressed under TMI Action Plan items II.E.1.1 and II.E.1.2 of NUREG 0737.

The SRP Section 15.2.6 currently cites II.E.1.1 and II.E.1.2 (refer to evaluations associated with these items) in addition to citations related to II.K.2.1 (i.e., II.K.2(1) and II.K.2 (C.2.1)).

Consideration should be given to deleting current citations related to TMI Action Plan item II.K.2.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25108/NUREG 0933

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**Integrated Impact Number:** 1093      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

NUREG-0660 Task II.K.3 was established to track the status of issues which were based on recommendations developed by the NRC staff Bulletins and Task Force. The task force conducted four design specific (Westinghouse, CE, B&W, and GE) reviews of loss-of-feedwater and small break loss-of-coolant accidents. Item II.K.3.21 addressed modifications to BWR LPCI system logic so that these systems will restart, if required, to assure adequate core cooling stopped by the operator. This item was designated as applicable to all BWR operating license applicants and holders in NUREG-0737. A clarification was provided for this item in Enclosure

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3 of NUREG-0737. No other positions specifically related to II.K.3.21 were identified.

An application requirement related to this issue was incorporated in the 10 CFR 50.34(f) as 10 CFR 50.34(f)(1)(viii). This requirement is applicable to certain BWR applicants for construction permits. Via 10 CFR 52.47, this requirement is also applicable to design certification applicants.

SRP Section 6.3 currently cites TMI action plan item II.K.3.21 of NUREG-0660 in Acceptance Criteria and Review Procedures, however, does not currently cite the related application requirement of 10 CFR 50.34(f)(1)(viii).

Consideration should be given to revising Acceptance Criteria to cite 10 CFR 50.34(f)(1)(viii) in connection with the existing citation of II.K.3.21. The citation of NUREG-0660 should be updated to reflect NUREG-0737 since a clarification for this item was provided in NUREG-0737 for implementation purposes.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24826/NUREG 0737; 25055/NUREG 0933; 25080/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1095      **SRP Section Number:** 5.4.6

#### **Suggested Changes to the SRP Section:**

Revise current Acceptance Criteria and associated Review Procedures related to TMI Item II.K.1.22 regarding automatic and manual actions when feedwater is not available.

NUREG-0660 Task II.K.1 was established to track licensee actions in response to a series of Bulletins issued between April 1, 1979, and July 26, 1979 in response to the TMI accident. Item II.K.1.22 required BWR licensees to describe the actions, both automatic and manual, necessary for proper functioning of the auxiliary heat removal systems (e.g., RCIC) that are used when the main feedwater system is not operable. For any manual action necessary, licensees were required to describe in summary form the procedure by which this action is taken in a timely sense. This item tracks licensee responses to Bulletin 79-08 (Item 3).

This Bulletin item was identified as II.K.1.22 and was approved for implementation without further clarification in NUREG 0737 as an item applicable to BWR applicants. II.K.1.22 was also identified in NUREG 0718 as an item applicable to BWRs. An application requirement related to this issue was incorporated in the 10 CFR 50.34(f) as 10 CFR 50.34(f)(2)(xxi). This requirement is applicable to certain BWR applicants for construction permits. Via 10 CFR 52.47, this requirement is also applicable to design certification applicants.

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SRP Section 5.4.6 currently identifies II.K.1.22 in Acceptance Criteria and Review Procedures. The SRP identifies NUREG-0737 and NUREG-0718 with regard to this item.

Consideration should be given to updating Acceptance Criteria to reflect the requirement of 10 CFR 50.34(f)(2)(xxi) related to II.K.1.22. Consideration might also be given to providing a statement in the existing review procedure that indicates this item is based on Bulletin 79-08, item 3, since the identified NUREGs do not provide clarification for this item.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24989/NRC BULLETIN 79-08; 24990/NUREG 0737; 24991/CODE OF FED. REGS  
10CFR50

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**Integrated Impact Number:** 1099      **SRP Section Number:** 5.2.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to incorporate TMI action plan item II.K.3.3 regarding reporting of safety and relief valve failures and challenges.

NUREG-0660 Task II.K.3 was established to track the status of issues which were based on recommendations developed by the NRC staff Bulletins and Task Force. The task force conducted four design specific (Westinghouse, CE, B&W, and GE) reviews of loss-of-feedwater and small break loss-of-coolant accidents. Item II.K.3.3 required all operating plants to report safety and relief valve failures promptly and challenges annually. The item was indicated as applicable to all operating license applicants and holders in NUREG-0737, however, a clarification was not provided for this item in Enclosure 3 of NUREG-0737. Generic Letters 83-02 and 82-16 provide descriptions of this item and identify appropriate Technical Specifications to address this issue. The SRP does not currently identify this NUREG-0737 item.

The Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors describes current NRC policy on the appropriate content of Technical Specifications to comply with 10 CFR 50.36. It isn't clear if the guidance of the above cited Generic Letters are consistent with current policy regarding administrative content of Technical Specification or the current versions of Improved Standard Technical Specifications.

Consideration should be given to citing II.K.3.3 in Acceptance Criteria and providing a Review Procedure regarding the applicant's conformance with II.K.3.3 of NUREG-0737. Additional analysis should be performed to determine if the Technical Specification guidance of Generic Letters 83-02 and 82-16 are consistent with current Improved Standard Technical Specifications.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25044/NUREG 0933; 25073/NRC GENERIC LETTER 83-02; 25076/NRC POLICY STATEMENT 58 FR 39132; 25772/NRC GENERIC LETTER 82-16; 25775/NUREG 1430; 25776/NUREG 1431; 25777/NUREG 1432; 25778/NUREG 1433; 25779/NUREG 1434

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**Integrated Impact Number:** 1101      **SRP Section Number:** 15.1.5

#### **Suggested Changes to the SRP Section:**

TMI action plan item II.K.2.16 requires evaluation of the impact of reactor coolant pump seal damage and leakage due to loss-of-seal cooling upon loss of offsite power. This NUREG-0737 item is applicable to B&W operating license holders and applicants. This issue originated with B&W plants but was later extended to all plants. As such, this item is identical to TMI action plan item II.K.3.25 which is applicable to all plants (BWR and PWR) except B&W plants. Since the only difference is the assignment of different numbers under the TMI action plan, II.K.2.16 is addressed along with II.K.3.25 in Revision Options Checklists for II.K.3.25. Refer to that Revision Options Checklist for recommended changes to SRP Section 15.1.5 related to this issue.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25019/NUREG 0737; 25022/NUREG 0933

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**Integrated Impact Number:** 1103      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

Delete Acceptance Criteria and Review Procedures related to interim degraded core accident rulemaking.

NUREG 0660 TMI Action Plan Item II.B.8 addresses rulemaking related to establishing additional requirements to address degraded core accidents (now generally referred to as severe accidents). As indicated in NUREG 0933 and the Commission Policy Statement on severe accidents (50 FR 32138), the interim degraded core accident rulemaking that was proposed under II.B.8 will not be pursued further. The policy statement indicates that future rulemakings regarding severe accident criteria are anticipated.

TMI-related issue II.B.8 was subsequently resolved in NUREG 0933 in consideration of a number of on-going regulatory activities and with regulatory actions taken associated with severe accident issues such as changes to 10 CFR 50.44 (related to hydrogen control) and the

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Commission Policy Statement on severe accidents (50 FR 32138). Also, requirements related to specific severe accident issues were established for certain applicants under 50.34(f)(1)(i), 50.34(f)(2)(ix), 50.34(f)(3)(iv), and 50.34(f)(3)(v).

SRP Section 6.3 directs a review of construction permits to establish the degree to which the proposed design conform to the proposed interim rule on degraded core accidents.

Consideration should be given to removing references to the proposed rulemaking on degraded core accidents currently cited in the SRP consistent with the current status of the rulemaking. Delete Acceptance Criteria and Review Procedures related to interim degraded core accident rulemaking.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24869/SECY 93-087; 24872/NRC GENERIC LETTER 88-20; 24875/NRC POLICY STATEMENT 50 FR 32138; 24878/NUREG 0933

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**Integrated Impact Number:** 1105      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

Add and revise review interfaces related to TMI Action Plan item II.B.2 and plant shielding for post-accident operation.

TMI Action Plan item II.B.2 addresses radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain source term radioactive materials. 10 CFR 50.34(f)(2)(vii) establishes related requirements applicable to certain construction permit and manufacturing license applicants and, via 10 CFR 52.47(a)(1)(ii), design certification applicants.

There are two aspects to this issue. The first involves the post-accident qualification of SSCs important to safety. This aspect of II.B.2 is addressed in SRP Section 3.11. The remaining part of II.B.2 involves accessibility to plant areas or equipment as necessary for post-accident plant operations. This aspect of II.B.2 is addressed primarily under Chapter 12 SRP Sections.

SRP Section 6.3 addresses this issue in a review interface discussion that identifies SRP Sections 12.1 through 12.5. The current discussion identifies NUREG 0694 and NUREG 0718 related to II.B.2. NUREG 0737 was issued after NUREG 0694 and contains a clarification for II.B.2 different from that of NUREG 0694. SRP Section 6.3 does not currently identify the post-accident equipment qualification and shielding aspect of this issue.

Consideration should be given revising SRP Section 6.3 to update the citation of II.B.2 in

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relation to the Chapter 12 SRP reviews and add a review interface that addresses the equipment shielding and qualification aspect of II.B.2. Since SRP Section 6.3 does not actually contain the criteria or review procedures related to this issue, the requirement of 10 CFR 50.34(f)(2)(vii) need not be cited since this requirement will be identified and discussed in the SRP sections being referenced.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24722/NUREG 0737; 24880/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1106      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Add review interfaces related to TMI Action Plan item II.B.2 and plant shielding for post-accident operation.

TMI Action Plan item II.B.2 addresses radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain source term radioactive materials. 10 CFR 50.34(f)(2)(vii) establishes related requirements applicable to certain construction permit and manufacturing license applicants and, via 10 CFR 52.47(a)(1)(ii), design certification applicants.

There are two aspects to this issue. The first involves the post-accident qualification of SSCs important to safety. This aspect of II.B.2 is addressed in SRP Section 3.11. The remaining part of II.B.2 involves accessibility to plant areas or equipment as necessary for post-accident plant operations. This aspect of II.B.2 is addressed primarily under Chapter 12 SRP Sections.

SRP Section 5.4.7 addresses this issue in a review interface discussion that identifies SRP Sections 12.1 through 12.5. The current discussion identifies NUREG 0737 and NUREG 0718 related to II.B.2. SRP Section 5.4.7 does not currently identify the post-accident equipment qualification and shielding aspect of this issue.

Consideration should be given revising SRP Section 5.4.7 to add a review interface that addresses the equipment shielding and qualification aspect of II.B.2. Since SRP Section 5.4.7 does not actually contain the criteria or review procedures related to this issue, the requirement of 10 CFR 50.34(f)(2)(vii) need not be cited since this requirement will be identified and discussed in the SRP sections being referenced.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24723/NUREG 0737; 24881/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1107      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

Revise or delete the current citation of TMI action plan item II.K.1.10 regarding procedures for removing safety-related systems from service.

NUREG-0660 Task II.K.1 was established to track licensee actions in response to a series of Bulletins issued between April 1, 1979, and July 26, 1979 in response to the TMI accident. Item II.K.1.10 addressed a licensee review of all valve positioning requirements and controls along with a review of all related test and maintenance procedures to assure proper functioning of ESF systems. This item tracks licensee responses to Bulletins 79-05A (Item 10), 79-06A (item 10), 79-06B (Item 9), and 79-08 (item 8). II.K.1.5 is identified in NUREG-0737 as an item applicable to operating license applicants, however, no clarification was provided in Enclosure 3.

SRP Section 6.3 cites TMI action plan item II.K.1 (C.1.10) of NUREG-0694. NUREG-0694 provides no additional clarification for the item. NUREG-0933 indicates that this issue was covered under TMI action plan items I.C.2 and I.C.6. No other positions related specifically to II.K.1.5 were identified.

Consider an editorial change to SRP Section 6.3 to cite TMI action plan item II.K.1.10 (as opposed to the NUREG-0660 form). Also, consider either identifying the Bulletins cited above, or incorporating the positions from the Bulletins in Review Procedures as guidance related to this item. Since (1) this issue is covered under other TMI action plan items (I.C.2 and I.C.6), (2) the issue is not specifically relevant to the review described in 6.3, and (3) since no other guidance specifically related to this item was identified, consideration might also be given to deleting the citation of II.K.1.10 from SRP Section 6.3.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24982/NRC BULLETIN 79-05A; 24984/NRC BULLETIN 79-06B; 24986/NRC BULLETIN 79-08; 24988/NUREG 0737

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**Integrated Impact Number:** 1108      **SRP Section Number:** 5.4.7

#### **Suggested Changes to the SRP Section:**

Revise or delete the current citation of TMI action plan item II.K.1.10 regarding procedures for removing safety-related systems from service.

NUREG-0660 Task II.K.1 was established to track licensee actions in response to a series of Bulletins issued between April 1, 1979, and July 26, 1979 in response to the TMI accident. Item II.K.1.10 addressed a licensee review of all valve positioning requirements and controls along with a review of all related test and maintenance procedures to assure proper functioning of ESF systems. This item tracks licensee responses to Bulletins 79-05A (Item 10), 79-06A (item 10), 79-06B (Item 9), and 79-08 (item 8). II.K.1.5 is identified in NUREG-0737 as an item applicable to operating license applicants, however, no clarification was provided in Enclosure 3.

SRP Section 5.4.7 cites TMI action plan item II.K.1. (C.1.10) of NUREG-0737 (the numbering in the citation is based on NUREG-0660 identification of the item). NUREG-0933 indicates that this issue was covered under TMI action plan items I.C.2 and I.C.6. No other positions related specifically to II.K.1.5 were identified.

Consider an editorial change to SRP Section 5.4.7 to cite TMI action plan item II.K.1.10 (as opposed to the NUREG-0660 form). Also, consider either identifying the Bulletins cited above, or incorporating the positions from the Bulletins in Review Procedures as guidance related to this item. Since; (1) this issue is covered under other TMI action plan items (I.C.2 and I.C.6), (2) the issue is not specifically relevant to the review described in 5.4.7, and (3) since no other guidance specifically related to this item was identified, consideration might also be given to deleting the citation of II.K.1.10 from SRP Section 5.4.7.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24981/NRC BULLETIN 79-05A; 24983/NRC BULLETIN 79-06B; 24985/NRC BULLETIN 79-08; 24987/NUREG 0737

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**Integrated Impact Number:** 1113      **SRP Section Number:** 15.0

#### **Suggested Changes to the SRP Section:**

Additional analysis required to determine the impact of TMI action plan item II.K.2.17, related to analysis of the potential for voiding in the reactor coolant system (RCS) during anticipated transients.

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NUREG-0660 Task II.K.2 was established to track the status of issues related to confirmatory shutdown orders issued to the seven B&W operating plants shortly after the TMI-2 accident. Item II.K.2.17 called for licensees with B&W reactors to perform an analysis of the potential for voiding in the RCS during anticipated transients. The applicability of this item was changed in NUREG-0737 and was made applicable to all PWR operating license applicants. A clarification was provided in Enclosure 3 of NUREG-0737. No other positions related specifically to II.K.2.17 were identified.

It would appear that the analysis referred to in II.K.2.17 would be addressed in the normal conduct of transient analysis performed by applicants and reviewed by the staff. In any event, since II.K.2.17 of NUREG-0737 is an item applicable to all PWR operating license applicants, it would seem appropriate for the SRP to cite II.K.2.17 of NUREG-0737. The SRP does not currently cite this item. The specific transient analysis sections of Chapter 15 that may be at issue under this item isn't clear and this appears to be a general consideration regarding transient analysis. Section 15.0 current cites II.K.3.44 as a generic BWR transient analysis consideration and in may be appropriate to incorporate II.K.2.17 in Section 15.0 in a similar manner.

Consideration should be given to performing additional analysis of the documents identified in NUREG-0737 and possibly coordinating with the responsible PRB to determine if II.K.2.17 of NUREG-0737 is relevant to current licensing reviews.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24765/NUREG 0737; 25003/NUREG 0933

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<b>Integrated Impact Number:</b> 1117	<b>SRP Section Number:</b> 15.2.8
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#### **Suggested Changes to the SRP Section:**

Revise existing Review Procedures to incorporate the resolution of TMI action plan item II.K.3.5 regarding automatic reactor coolant pump trips.

NUREG-0660 Task II.K.3 was established to track the status of issues which were based on recommendations developed by the NRC staff Bulletins and Task Force. The task force conducted four design specific (Westinghouse, CE, B&W, and GE) reviews of loss-of-feedwater and small break loss-of-coolant accidents. Item II.K.3.5 addresses automatic tripping of reactor coolant pumps (RCP) during a small break LOCA. II.K.3.5 was indicated as applicable to all PWR operating license applicants and holders in NUREG-0737. A clarification was provided for this item in Enclosure 3 of NUREG-0737.

Generic Letters 83-10A through 83-10F documents the review of vendor and NRC analyses

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performed to address the resolution of II.K.3.5. The NRC concluded that there is a wide range of transients and LOCAs where it is beneficial for the operators to maintain forced circulation cooling and mixing through operation of the RCPs. However, for certain small break LOCAs, continued operation of the RCPs or delayed RCP trips could lead to core damage. Generic Letters 85-12, 86-05 and 86-06 describe the staff's safety evaluation of the three PWR owners' groups response to Generic Letters 83-10A through 83-10F.

Consideration should be given to revising the existing Review Procedures to incorporate the guidance described above related to the resolution of TMI action plan item II.K.3.5.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24772/NRC GENERIC LETTER 83-10A; 24777/NUREG 0737; 24782/NRC GENERIC LETTER 83-10B; 24787/NRC GENERIC LETTER 83-10C; 24792/NRC GENERIC LETTER 83-10D; 24797/NRC GENERIC LETTER 83-10E; 24802/NRC GENERIC LETTER 83-10F; 24807/NRC GENERIC LETTER 85-12; 24812/NRC GENERIC LETTER 86-05; 24817/NRC GENERIC LETTER 86-06; 25046/NUREG 0933

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**Integrated Impact Number:** 1118      **SRP Section Number:** 15.1.5

#### **Suggested Changes to the SRP Section:**

Revise existing Review Procedures to incorporate the resolution of TMI action plan item II.K.3.5 regarding automatic reactor coolant pump trips.

NUREG-0660 Task II.K.3 was established to track the status of issues which were based on recommendations developed by the NRC staff Bulletins and Task Force. The task force conducted four design specific (Westinghouse, CE, B&W, and GE) reviews of loss-of-feedwater and small break loss-of-coolant accidents. Item II.K.3.5 addresses automatic tripping of reactor coolant pumps (RCP) during a small break LOCA. II.K.3.5 was indicated as applicable to all PWR operating license applicants and holders in NUREG-0737. A clarification was provided for this item in Enclosure 3 of NUREG-0737.

Generic Letters 83-10A through 83-10F documents the review of vendor and NRC analyses performed to address the resolution of II.K.3.5. The NRC concluded that there is a wide range of transients and LOCAs where it is beneficial for the operators to maintain forced circulation cooling and mixing through operation of the RCPs. However, for certain small break LOCAs, continued operation of the RCPs or delayed RCP trips could lead to core damage. Generic Letters 85-12, 86-05 and 86-06 describe the staff's safety evaluation of the three PWR owners' groups response to Generic Letters 83-10A through 83-10F.

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Consideration should be given to revising the existing Review Procedures to incorporate the guidance described above related to the resolution of TMI action plan item II.K.3.5.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24771/NRC GENERIC LETTER 83-10A; 24776/NUREG 0737; 24781/NRC GENERIC LETTER 83-10B; 24786/NRC GENERIC LETTER 83-10C; 24791/NRC GENERIC LETTER 83-10D; 24796/NRC GENERIC LETTER 83-10E; 24801/NRC GENERIC LETTER 83-10F; 24806/NRC GENERIC LETTER 85-12; 24811/NRC GENERIC LETTER 86-05; 24816/NRC GENERIC LETTER 86-06; 25045/NUREG 0933

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**Integrated Impact Number:** 1120      **SRP Section Number:** 15.2.7

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to cite TMI action plan item II.K.2.19 of NUREG-0737, related to a benchmark analysis of sequential auxiliary feed water (AFW) flow to the steam generators following a loss of main feed water.

NUREG-0660 Task II.K.2 was established to track the status of issues related to confirmatory shutdown orders issued to the seven B&W operating plants shortly after the TMI-2 accident. Item II.K.2.19 called for licensees with B&W reactors to perform a benchmark analysis of sequential auxiliary feed water (AFW) flow to the steam generators following a loss of main feed water. The applicability of this item was changed in NUREG-0737 and was made applicable to all PWR operating license applicants. A clarification was provided in Enclosure 3 of NUREG-0737. No other positions related specifically to II.K.2.19 were identified.

Since II.K.2.19 of NUREG-0737 is an item applicable to all PWR operating license applicants, it would seem appropriate for the SRP to cite II.K.2.19. The SRP does not currently cite this item.

Consideration should be given to revising the Acceptance Criteria and Review Procedures of SRP Section 15.2.7 to cite II.K.2.19 of NUREG-0737.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24766/NUREG 0737; 25004/NUREG 0933

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**Integrated Impact Number:** 1121      **SRP Section Number:** 15.2.8

#### **Suggested Changes to the SRP Section:**

TMI action plan item II.K.2.16 requires evaluation of the impact of reactor coolant pump seal damage and leakage due to loss-of-seal cooling upon loss of offsite power. This NUREG-0737 item is applicable to B&W operating license holders and applicants. This issue originated with B&W plants but was later extended to all plants. As such, this item is identical to TMI action plan item II.K.3.25 which is applicable to all plants (BWR and PWR) except B&W plants. Since the only difference is the assignment of different numbers under the TMI action plan, II.K.2.16 is addressed along with II.K.3.25 in Revision Options Checklists for II.K.3.25. Refer to that Revision Options Checklist for recommended changes to SRP Section 15.2.8 related to this issue.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25020/NUREG 0737; 25023/NUREG 0933

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**Integrated Impact Number:** 1122      **SRP Section Number:** 6.3

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to incorporate the TMI action plan item II.K.2.15 related to the evaluation of the effects of slug flow on steam generator tubes for B&W plants.

NUREG-0660 Task II.K.2 was established to track the status of issues related to confirmatory shutdown orders issued to the seven B&W operating plants shortly after the TMI-2 accident. Item II.K.2.15 called for licensees with B&W reactors to perform an evaluation of the effects of slug flow on steam generator tubes following reactor coolant system voiding to assure that the tubes could withstand any mechanical loading which could result from slug flow. This item was approved for implementation in NUREG-0737 and was made applicable to all B&W operating license applicants. A clarification was provided in Enclosure 3 of NUREG-0737. No other positions related specifically to II.K.2.15 were identified.

SRP Section 6.3 currently cites TMI action plan item II.K.3.39 of NUREG-0660 which is somewhat related to this issue. II.K.3.39 was based on NUREG-0565, item 2.6.2.e, which recommended evaluation of the effects of water slugs caused by ECCS flows. However, when NUREG-0660 was published, this item was identified as being addressed under TMI action plan item I.C.1. II.K.3.39 was not identified in NUREG-0660 as an action plan item. This item was not approved for implementation in NUREG-0737 and did not appear in NUREG-0718 as an

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item applicable to construction permit holders.

Consideration should be given to revising Acceptance Criteria and Review procedures to incorporate a review of the issue associated II.K.2.15 of NUREG-0737 for applicants with B&W reactors. Consideration should also be given to deleting the current citation of II.K.3.19.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24756/NUREG 0737; 25002/NUREG 0933

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**Integrated Impact Number:** 1127      **SRP Section Number:** 15.2.8

#### **Suggested Changes to the SRP Section:**

II.K.3.40 of NUREG-0660 is based on NUREG-0565, item 2.6.2.f. II.K.3.40 was originated to evaluate RCP seal damage and leakage during a SBLOCA in B&W plants. However, prior to the publication of NUREG-0660, it was determined that the issue was being addressed by Item II.K.2.16. II.K.3.40 was not approved for implementation and does not appear in NUREG-0694, -0718, or -0737. TMI action plan item II.K.2.16 was approved for implementation in NUREG-0737 and an application requirement, 10 CFR 50.34(f)(1)(iii), was established related to this item. II.K.2.16 is addressed together with II.K.3.25 and 10 CFR 50.34(f)(1)(iii) under a Revision Options Checklist associated with II.K.3.25.

Since II.K.3.40 was superseded by II.K.2.16 prior to the publication of NUREG-0660 and no unique guidance or requirements are associated with this item, consideration should be given to deleting citations of II.K.3.40.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25025/NUREG 0933

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**Integrated Impact Number:** 1128      **SRP Section Number:** 15.1.5

#### **Suggested Changes to the SRP Section:**

Delete citation of TMI action plan item II.K.3.40.

II.K.3.40 of NUREG-0660 is based on NUREG-0565, item 2.6.2.f. II.K.3.40 was originated to evaluate RCP seal damage and leakage during a SBLOCA in B&W plants. However, prior to the publication of NUREG-0660, it was determined that the issue was being addressed by Item II.K.2.16. II.K.3.40 was not approved for implementation and does not appear in NUREGs

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-0694, -0718, or -0737. TMI action plan item II.K.2.16 was approved for implementation in NUREG-0737 and an application requirement, 10 CFR 50.34(f)(1)(iii), was established related to this item. II.K.2.16 is addressed together with II.K.3.25 and 10 CFR 50.34(f)(1)(iii) under a Revision Options Checklist associated with II.K.3.25.

Since II.K.3.40 was superseded by II.K.2.16 prior to the publication of NUREG-0660 and no unique guidance or requirements are associated with this item, consideration should be given to deleting citations of II.K.3.40.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25024/NUREG 0933

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**Integrated Impact Number:** 1130      **SRP Section Number:** 3.7.3

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria, Review Procedures and Evaluation Findings, applicable to evolutionary plants, for review of seismic system analysis.

As discussed in the ABWR and CE 80+ FSERs and in SECY 93-087, evolutionary plants may eliminate the Operating Basis Earthquake (OBE) analysis from the design basis for structures, systems and components (SSCs). In such cases, the OBE will serve as an inspection level earthquake above which the licensee would shut down the plant and inspect for potential damage to SSCs important to safety. Design certification based upon this single-earthquake design approach is predicated, in part, on the ability to ascertain the need to take action to shut down the plant in case of occurrence of the OBE, and to evaluate the performance of SSCs important to safety. In the ABWR and CE80+ FSERs, the NRC staff reviewed criteria developed by the Electric Power Research Institute (EPRI) for evaluating whether or not the plant needs to be shut down following an earthquake.

Consider revising Acceptance Criteria, Review Procedures and Evaluation Findings, applicable to evolutionary plants, for review of seismic system analysis.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23004/SECY 93-087; 25099/FINAL SER CE80 CH 1; 25100/FINAL SER ABWR CH 1; 25565/FINAL SER CE80 CH 3; 25566/FINAL SER CE80 CH 3; 25569/FINAL SER ABWR CH 3

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**Integrated Impact Number:** 1131      **SRP Section Number:** 5.3.2

#### **Suggested Changes to the SRP Section:**

Revise the Review Procedures to address review of plant specific temperature limits for combined license applicants.

In both the CE System 80+ and GE ABWR FSERs, the staff accepted the projected values for RTndt and pressure-temperature limits as required by 10 CFR 50, Appendix G. The staff further required that the combined operating license applicant submit plant specific calculations for RTndt, stress intensity factors and pressure temperature curves.

Consider revising the Review Procedures to address the review of plant specific RTndt calculations, stress intensity factors, and pressure-temperature curves for combined license applications.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24176/FINAL SER ABWR CH 5; 24177/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 1132      **SRP Section Number:** 5.3.2

#### **Suggested Changes to the SRP Section:**

Revise the Branch Technical Position, BTP MTEB 5-2, to cite the latest version of ASTM E-185.

BTP MTEB 5-2 cites ASTM E-185-73 with regard to selection of material to be included in the reactor vessel surveillance program. The reactor vessel surveillance program requirements are established in 10 CFR 50, Appendix H. Appendix H endorses the 1982 version of ASTM E-185. The latest version of the standard is ASTM E-185-94.

Consider revising the BTP to cite ASTM E-185-94. This is a placeholder integrated impact and will not be processed further.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23574/C&S: ASTM E185-73; 25116/CODE OF FED. REGS 10CFR50

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## **APPENDIX I**

### **Integrated Impacts**

**Integrated Impact Number:** 1202      **SRP Section Number:** 16.0

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria and Review Procedures to incorporate the requirements of 10 CFR 50.36 as amended on July 19, 1995.

A proposed rule at 59 FR 48180 involved changes to amend the technical specification requirements of 10 CFR 50.36, paragraph (c)(2). The proposed changes establish criteria for determining the need for a technical specification limiting condition for operation (LCO). The proposed rule provides licensees the opportunity to remove certain LCOs that do not meet the criteria from the technical specifications to licensee-controlled documents. A notice of the final rule, which accomplishes the same purposes and amends paragraph (c)(2) as did the proposed rule, was published in the Federal Register on July 19, 1995.

Consideration should be given to revising SRP Section 16.0 as necessary to incorporate the current requirements of 10 CFR 50.36(c)(2).

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25183/CODE OF FED. REGS 10CFR50; 25948/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1221      **SRP Section Number:** 3.7.1

#### **Suggested Changes to the SRP Section:**

Revise the SRP to incorporate the new and revised requirements from proposed rulemaking 59 FR 52255.

This proposed rule involves extensive changes to amend 10 CFR 100 with regard to source term and dose considerations, and seismic and earthquake considerations related to reactor siting. In addition to changes within 10 CFR 100, the proposed rule also amends 10 CFR 50.34, 10 CFR 50.54, and adds a new Appendix S to 10 CFR 50.

This is a placeholder integrated impact for the proposed rulemaking and will not be processed further.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25234/CODE OF FED. REGS 10CFR100

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## **APPENDIX I**

### **Integrated Impacts**

**Integrated Impact Number:** 1222      **SRP Section Number:** 3.7.2

#### **Suggested Changes to the SRP Section:**

Revise the SRP to incorporate the new and revised requirements from proposed rulemaking 59 FR 52255.

This proposed rule involves extensive changes to amend 10 CFR 100 with regard to source term and dose considerations, and seismic and earthquake considerations related to reactor siting. In addition to changes within 10 CFR 100, the proposed rule also amends 10 CFR 50.34, 10 CFR 50.54, and adds a new Appendix S to 10 CFR 50.

This is a placeholder integrated impact for the proposed rulemaking and will not be processed further.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25235/CODE OF FED. REGS 10CFR100

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**Integrated Impact Number:** 1223      **SRP Section Number:** 3.7.3

#### **Suggested Changes to the SRP Section:**

Revise the SRP to incorporate the new and revised requirements from proposed rulemaking 59 FR 52255.

This proposed rule involves extensive changes to amend 10 CFR 100 with regard to source term and dose considerations, and seismic and earthquake considerations related to reactor siting. In addition to changes within 10 CFR 100, the proposed rule also amends 10 CFR 50.34, 10 CFR 50.54, and adds a new Appendix S to 10 CFR 50.

This is a placeholder integrated impact for the proposed rulemaking and will not be processed further.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25236/CODE OF FED. REGS 10CFR100; 25570/FINAL SER CE80 CH 3

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## **APPENDIX I**

### **Integrated Impacts**

**Integrated Impact Number:** 1248      **SRP Section Number:** 9.5.1

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria and Review Procedures to address fire protection during low-power and shutdown operations.

The current NRC fire protection requirements and guidance address the capability to accomplish safe shutdown of the reactor in the event of fire. These requirements are premised on achieving hot shutdown followed by cold shutdown modes of operation from at-power conditions.

NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States," reported that 10 CFR Part 50 (Appendix R) and the current NRC fire-protection philosophy do not address shutdown and refueling conditions and the impact a fire may have on the plant's ability to remove decay heat and maintain reactor coolant temperature. The insights obtained from NUREG-1449 were utilized as part of the reviews of the CE System 80+ and ABWR fire protection programs during shutdown conditions.

Consider revising the Acceptance Criteria and Review Procedures to incorporate review of shutdown and low-power operations for vulnerabilities to fire.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23149/NRC GENERIC LETTER 88-17; 23152/FINAL SER EPRI CH 5; 23158/DRAFT SER CE80 CH 5; 23164/ADVANCE SER ABWR CH 19; 25274/FINAL SER CE80 CH 19; 25275/NUREG 1449

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**Integrated Impact Number:** 1278      **SRP Section Number:** 6.1.1

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria to identify acceptable alternatives to compliance with Regulatory Guide 1.50.

In the CE System 80+ FSER, the staff indicated that the applicant took exception to the recommendations in Position C.2 of RG 1.50 for controls imposed on preheat temperatures for welding ferritic steels. These controls normally provide reasonable assurance that components made from low-alloy steels will not crack during fabrication and minimize the possibility of subsequent cracking from hydrogen in the weldment. The applicant indicated that its basis for taking exception to Position C.2 was Westinghouse Topical Report WCAP-8678, "Effect of Preheat and Post Weld Heat Treat on Hydrogen-Induced Cracking in Pressure Vessel Steels,"

## **APPENDIX I**

### **Integrated Impacts**

September 1975. The staff evaluated and accepted this report. This report presents three acceptable alternatives, and the applicant's position is that a particular alternative will be specified based upon various factors such as the configuration or the capabilities of the fabrication facility. The staff concluded that this approach should provide adequate assurance that low-alloy steel weldments will not develop cracking due to hydrogen.

SRP Section 5.2.3, subsection II.3.b currently indicates that the preheat controls described in Westinghouse Topical Report WCAP-8577 are an acceptable alternate to compliance with those of Regulatory Guide 1.50. ROC 845 adds description of WCAP-8678 as a further acceptable alternative in SRP Section 5.2.3, subsection II.3.b.

Consideration should be given to revising Acceptance Criteria to cite SRP Section 5.2.3, subsection II.3.b for description of acceptable alternatives to compliance with Regulatory Guide 1.50.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25322/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 1279      **SRP Section Number:** 9.3.4

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for review of capabilities to control the composition of primary coolant within acceptable limits/specifications.

The current Areas of Review and Review Procedures for SRP Section 9.3.4 include reviews of capabilities to control the composition of primary coolant within acceptable limits/specifications.

The EPRI Evolutionary Plant FSER Chapter 1 provides staff positions that PWR plant systems water chemistry design basis in accordance with EPRI Report NP-7077, Rev. 2, November 1990, (primary water) and EPRI Report NP-6239, Rev. 2, (secondary water) as supplemented by Table 1.3 of the FSER are acceptable.

In the CE System 80+ FSER, the staff accepted the applicant's primary and secondary water chemistry commitments based upon a determination that the applicant meets or exceeds the guidelines of the EPRI Utility Requirements Document.

Consideration should be given to revising Review Procedures for review of capabilities to control the composition of primary coolant based upon information provided in the above FSERs.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25328/FINAL SER CE80 CH 5; 25329/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 1280      **SRP Section Number:** 6.1.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to identify acceptable alternatives to compliance with Regulatory Guide 1.44 controls to verify non-sensitization of austenitic stainless steel materials and weldments.

In the CE System 80+ FSER, the staff indicated that the applicant proposes to allow the continued reference to American Society for Testing and Materials (ASTM) A-708 for the purpose of maintaining the qualifications of older weld procedures. ASTM A-262, Practice A or E are the methods explicitly recommended in RG 1.44, and have been used by the applicant since the mid-1970s for verifying non-sensitization of austenitic stainless steel materials and weldments. The applicant qualified weld procedures developed prior to the mid-1970s and used A-708 in their qualifications. These weld procedures are still in use. There have been no intergranular stress corrosion cracking (IGSCC) failures involving stainless steel weldments in the applicant's nuclear steam supply system (NSSS) units. In addition, the staff has allowed in SRP 4.5.1, paragraph III.2, ASTM A-708 as an acceptable alternative test for ASTM A-262, Practice A or E.

ROC 846 addresses this issue in SRP Section 5.2.3, subsection II.4.a via addition of a description of use of ASTM A-708 as an acceptable alternative to compliance with Regulatory Guide 1.44 controls to verify non-sensitization of austenitic stainless steel materials and weldments.

Consideration should be given to revising Review Procedures to reference the SRP Section 5.2.3, subsection II.4.a description of an acceptable alternative to compliance with Regulatory Guide 1.44 controls to verify non-sensitization of austenitic stainless steel materials and weldments.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25319/FINAL SER CE80 CH 5

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## **APPENDIX I**

### **Integrated Impacts**

**Integrated Impact Number:** 1281      **SRP Section Number:** 19.2

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria for severe accident containment performance. Also, develop a background discussion for Areas of Review including a definition of "severe accidents."

NRC Policy Statement 50 FR 32138, "Policy Statement On Severe Reactor Accidents Regarding Future Designs And Existing Plants" describes the policy the Commission intends to use to resolve safety issues related to reactor accidents more severe than design basis accidents. The Commission believes that a new design for a nuclear power plant can be shown to be acceptable for severe accident concerns if it meets the following criteria and procedural requirements: a. Demonstration of compliance with the procedural requirements and criteria of the current Commission regulations, including the Three Mile Island requirements for new plants; b. Demonstration of technical resolution of all applicable Unresolved Safety Issues and the medium- and high- priority Generic Safety Issues; c. Completion of a Probabilistic Risk Assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes; and d. Completion of a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analysis and judgment complemented by PRA. The NRC stated that "The Commission fully expects that vendors engaged in designing new standard (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs."

The ABWR and System 80+ FSERs cite NRC Policy Statement 50 FR 32138 as providing applicable guidance for addressing severe reactor accidents. These FSERs also reference SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," and SECY-93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," as providing criteria for resolving severe accident issues that will be incorporated into the ABWR and System 80+ design certification rulemaking as applicable regulations.

NUREG 1070, "NRC Policy on Future Reactor Designs," defines a severe accident as a reactor accident more severe than design-basis accidents in which, as a minimum, substantial damage is done to the reactor core.

Consider developing Acceptance Criteria and Technical Rationale for severe accident containment performance. Also, consideration should be given to developing background discussion for Areas of Review including a definition of "severe accidents."

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### **Integrated Impacts**

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24608/FINAL SER CE80 CH 19; 24609/FINAL SER ABWR CH 19; 24610/NUREG 1070; 24611/NRC POLICY STATEMENT 50 FR 32138

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**Integrated Impact Number:** 1282      **SRP Section Number:** 19.2

#### **Suggested Changes to the SRP Section:**

Provide Acceptance Criteria and Review Procedures for the purpose of establishing a design basis for severe accident hydrogen control.

10 CFR 52.47, Item (a)(1)(ii) requires an applicant for a standard design certification to include a demonstration of compliance with technically relevant portions of the TMI requirements identified in 10 CFR 50.34(f). Further, in SECY-90-016, the staff recommended to the Commission that the hydrogen control requirements for evolutionary plants be identical to those stated in 10 CFR 50.34(f). The Commission approved the staff's recommendation in a staff requirements memorandum dated June 26, 1990.

Paragraphs (2)(ix) and (3)(v) of section 50.34 (f) provide requirements pertaining to hydrogen control measures. 10 CFR 50.34(f)(2)(ix) requires a hydrogen control system based on a 100-percent fuel-cladding metal-water reaction and a hydrogen concentration limit of 10 percent on uniformly distributed hydrogen in the containment or a post accident atmosphere that will not support hydrogen combustion. 10 CFR 50.34(f)(3)(v) requires containment integrity to be maintained during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent; Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220; and the containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.

In the ABWR FSER and the CE80+ FSER, the staff reviewed GE's and CE's proposed methods for addressing severe accident hydrogen control. GE proposed inerting and CE proposed igniters to address severe accident hydrogen control requirements which the staff found acceptable.

Consider providing Acceptance Criteria and Review Procedures establishing design bases from 10 CFR 50.34(f) to address severe accident hydrogen control.

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### **Integrated Impacts**

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25307/CODE OF FED. REGS 10CFR50; 25308/CODE OF FED. REGS 10CFR52;  
25309/SECY 90-016; 25310/FINAL SER CE80 CH 19; 25311/FINAL SER ABWR CH 19

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**Integrated Impact Number:** 1283      **SRP Section Number:** 19.2

#### **Suggested Changes to the SRP Section:**

Develop Specific Criteria and Review Procedures for the containment performance goals.

In SECY-90-016 the staff recommended that evolutionary plant designs achieve a containment performance goal of either a conditional containment failure probability (CCFP) of 0.1 or a deterministic goal that offers comparable protection to a CCFP of 0.1. The staff also provided a deterministic goal that would be appropriate for evolutionary designs in place of a CCFP goal as follows: "The containment should maintain its role as a reliable leak tight barrier by ensuring that containment stresses do not exceed ASME service level C limits for a minimum period of 24 hours following the onset of core damage and that following this 24 hour period the containment should continue to provide a barrier against the uncontrolled release of fission products." In an SRM dated June 26, 1990, the Commission approved the use of a CCFP of 0.1 as a basis for establishing regulatory guidance for evolutionary plants, but also stated that this CCFP should not be imposed as a requirement in and of itself. The Commission indicated that use of the CCFP should not discourage accident prevention and that the staff should review suitable alternative, deterministically-established, containment performance objectives providing comparable mitigation capability if submitted by applicants.

In SECY-93-087 the staff restated the same containment performance goal that was recommended in SECY-90-016 and approved via SRM, and also provided additional insight regarding this position. The staff stated they recognized limitations in the CCFP approach, but that they believed use of the CCFP maintained a balance between accident prevention and consequence mitigation.

In the ABWR and System 80+ FSERs the staff utilized the SECY-90-016 criteria in reviewing and accepting the licensees' containment performance goals. The FSERs utilized two alternative definitions of containment failure: (1) loss of containment structural integrity and (2) releases that result in doses of 25 rem or greater at 0.8 km from the reactor. The ABWR and System 80+ designs both met the CCFP goal of 0.1 for the first 24 hours following core damage utilizing either containment failure definition.

Consideration should be given to incorporating containment performance goals as Specific Criteria and Review Procedures in this new section.



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### **Integrated Impacts**

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24601/SECY 90-016; 25312/FINAL SER ABWR CH 19; 25313/FINAL SER CE80 CH 19;  
25314/SECY 93-087

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**Integrated Impact Number:** 1284      **SRP Section Number:** 19.2

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria, including Specific Criteria, and Review Procedures for mitigation of containment overpressure.

10 CFR 52.47, Item (a)(1)(ii) requires an application for a standard design certification to include a demonstration of compliance with technically relevant portions of the TMI requirements identified in 10 CFR 50.34(f). 10 CFR 50.34(f)(3)(iv) requires plants to have a 3-ft diameter equivalent vent in order not to preclude future installation of systems to prevent containment failure, such as a filtered containment vent.

In SECY-90-016 the staff recommended that ABWR plant design should include a containment overpressure protection system to meet containment vent requirements. The Commission approved this position in the corresponding SRM dated June 26, 1990.

This issue is also discussed in a similar manner in SECY-93-087, however no changes to the position for ABWRs were indicated. Rather, SECY-93-087 provides a general insight into the NRC's current approach to the issue of containment vents. This SECY states that a containment vent is just one of many systems that can be used to mitigate the consequences of an accident, and that if acceptable analyses indicate that a vent would not be needed to meet severe accident criteria, then the staff would not propose to implement a vent requirement. SECY-93-087 also established an acceptable containment performance deterministic goal as follows: "The containment should maintain its role as a reliable leak tight barrier by ensuring that containment stresses do not exceed ASME service level C limits for a minimum period of 24 hours following the onset of core damage and that following this 24 hour period the containment should continue to provide a barrier against the uncontrolled release of fission products."

In the ABWR and System 80+ FSERs the staff utilized the SECY 93-087 containment performance deterministic goal as criteria in reviewing and accepting the licensees mitigation features for containment overpressure protection.

Consideration should be given to developing Acceptance Criteria, including Specific Criteria, and Review Procedures for mitigation of containment overpressure.

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### **Integrated Impacts**

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24605/SECY 90-016; 25102/CODE OF FED. REGS 10CFR50; 25308/CODE OF FED. REGS 10CFR52; 25316/FINAL SER ABWR CH 19; 25317/FINAL SER CE80 CH 19; 25318/SECY 93-087

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**Integrated Impact Number:** 1285      **SRP Section Number:** 19.2

#### **Suggested Changes to the SRP Section:**

Develop Specific Criteria and Review Procedures for core debris coolability and core debris mitigating factors.

In SECY-90-016 the staff recommended that evolutionary plant designs should include the following features to enhance core debris coolability and mitigate the effects of core debris melting through the reactor core; (1) sufficient reactor cavity floor space to enhance debris spreading, (2) provide for quenching debris in the reactor cavity. These recommendations were approved by the Commission in an SRM dated June 26, 1990.

In SECY-93-087 the staff expanded their recommendations for evolutionary plant design core debris coolability and mitigation as follows: (1) provide reactor cavity floor space to enhance debris spreading, (2) provide a means to flood the reactor cavity to assist in the cooling process, (3) protect the containment liner and other structural members with concrete, if necessary, and (4) ensure best estimate environmental conditions resulting from core-concrete interactions do not exceed Service Level C limits for steel containments or Factored Load Category limits for concrete containments, for approximately 24 hours, and ensure that containment capability has margin to accommodate uncertainties in the environmental conditions from core-concrete interactions. The Commission approved these recommendations in their July 21, 1993 SRM.

In the ABWR and System 80+ FSERs the staff utilized SECY-90-016 and 93-087 positions in reviewing and accepting the licensees' core debris coolability features and core melt mitigating factors. In these FSERs the staff stated: "Therefore, the staff's proposed applicable regulation for core debris coolability is as follows: The standard design must include features that reduce the potential for and effect of interactions with molten core debris by: (1) providing reactor cavity floor space to enhance debris spreading; (2) providing a means to flood the reactor cavity to assist in the cooling process; (3) protecting the containment liner and other structural members with concrete, if necessary; and (4) providing design features that ensure that the best-estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed service level C for steel containments or factored load category for concrete containments, for approximately 24 hours."

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### **Integrated Impacts**

Consideration should be given to developing Specific Criteria and Review Procedures for core debris coolability and mitigating factors.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24602/SECY 90-016; 24603/SECY 93-087; 25320/FINAL SER ABWR CH 19; 25321/FINAL SER CE80 CH 19

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**Integrated Impact Number:** 1286      **SRP Section Number:** 19.2

#### **Suggested Changes to the SRP Section:**

Develop Specific Criteria and Review Procedures to address high pressure core melt ejection (HPME) and direct containment heating.

In SECY-90-016 the staff recommended that evolutionary plant designs include the following features to mitigate the effects of molten core debris being ejected from the reactor vessel under high pressure; a) a depressurization system, and b) cavity design features to contain ejected core debris. The Commission approved this recommendation in their June 26, 1990 SRM. This SRM also indicated that cavity design, as a mitigating feature, should not unduly interfere with operations such as refueling, maintenance, and surveillance.

In SECY-93-087 the staff restated their high pressure core melt ejection mitigation recommendations slightly reworded as follows: (a) provide a reliable depressurization system and (b) provide cavity design features to decrease the amount of ejected core debris that reaches the upper containment. The Commission approved this reworded recommendation in their July 21, 1993 SRM.

In the ABWR and System 80+ FSERs the staff utilized SECY-93-087 and 90-016 positions in reviewing and accepting the licensees' design features to mitigate the effects of high pressure core melt ejection. These FSERs made the following statement: "Therefore, the staff's proposed applicable regulation for HPME is as follows: The standard design must provide a reliable means to depressurize the RCS and cavity design features to reduce the amount of ejected core debris that may reach the upper containment so that the potential for and effects of interactions with molten core ejected under high pressure are reduced."

Consideration should be given to developing Specific Criteria and Review Procedures to address high pressure core melt ejection and direct containment heating.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24604/SECY 93-087; 25323/FINAL SER ABWR CH 19; 25324/FINAL SER CE80 CH 19;  
25325/SECY 90-016

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**Integrated Impact Number:** 1287      **SRP Section Number:** 19.2

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria, including Specific Criteria, and Review Procedures for severe accident equipment survivability considerations.

10 CFR 50.34(f) provides equipment survivability requirements as follows: Part 50.34(f)(2)(ix)(c) states that equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system. Part 50.34(f)(3)(v) states that systems necessary to ensure containment integrity shall be demonstrated to perform their function under conditions associated with an accident that releases hydrogen generated from 100 percent fuel-clad metal-water reaction. Part 50.34(f)(2)(xvii) requires instrumentation to measure containment pressure, containment water level, containment hydrogen concentration, containment radiation intensity, and noble gas effluents at all potential accident release points. Part 50.34(f)(2)(xix) requires instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

In SECY-90-016 the staff recommended that features provided for severe-accident protection (prevention and mitigation) only (not for design basis accidents) need not be subject to (a) the 10 CFR 50.49 environmental qualification requirements, (b) all aspects of 10 CFR part 50, Appendix B quality assurance requirements, or (c) 10 CFR Part 50, Appendix A redundancy/diversity requirements. However, mitigation features must be designed so there is reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed. In instances where safety related equipment, (which is provided for design basis accidents) is relied upon to cope with severe accident situations; there should also be a high confidence that this equipment will survive severe accident conditions for the period that is needed to perform its intended function. However, it is not necessary for redundant trains to be qualified to meet this goal. The position proposed by the staff was approved by the Commission in an SRM dated June 26, 1990.

In SECY-93-087, the staff restated their earlier SECY-90-016 position that evolutionary plant features provided only for severe-accident protection (prevention and mitigation) need not be

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subject to the environmental qualification requirements of 10 CFR 50.49; quality assurance requirements of 10 CFR Part 50, Appendix B; or redundancy/diversity requirements of 10 CFR Part 50, Appendix A, and that mitigation features must be designed to provide reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed.

In the ABWR and System 80+ FSERs the staff utilized SECY-93-087 and 90-016 positions in reviewing and accepting the licensees' severe accident equipment survivability considerations. These FSERs made the following statement: "Therefore, the staff's proposed applicable regulation for equipment survivability is as follows: The standard design must include analyses, based on best available methods, to demonstrate that: Equipment, both electrical and mechanical, needed to prevent and mitigate the consequences of severe accidents is capable of performing its function for the time period needed in the best-estimate environmental conditions of the severe accident (e.g., pressure, temperature, radiation) in which the equipment is relied upon to function. Instrumentation needed to monitor plant conditions during a severe accident is capable of performing its function for the time period needed in the best-estimate environmental conditions of the severe accident (e.g., pressure, temperature, radiation) in which the instrumentation is relied upon to function."

Consideration should be given to developing Acceptance Criteria, including Specific Criteria, and Review Procedures for severe accident equipment survivability considerations.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24606/SECY 90-016; 24607/SECY 93-087; 25307/CODE OF FED. REGS 10CFR50;  
25326/FINAL SER CE80 CH 19; 25327/FINAL SER ABWR CH 19

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**Integrated Impact Number: 1303      SRP Section Number: 6.8**

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for PWRs to ensure that PORVs used to perform safety-related functions are reliable.

NRC Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block-Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," requires that specific improvements be taken with respect to PORVs used in safety-related functions including: 1) Inclusion in the QA and Preventive Maintenance programs, 2) Inclusion in the Inservice Inspection program, and 3) Upgraded PORV technical specifications.

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NRC Memo 11/16/89 from Beckjord (Office of Nuclear Regulatory Research) to Gillespie (Office of Nuclear Reactor Regulation) provides guidance concerning the applicability of Generic Letter 90-06.

As described in the CE FSER, the System 80+ design does not utilize PORVs, but rather includes a full safety-grade Safety Depressurization System (SDS).

Consideration should be given to developing Review Procedures for PWRs to ensure that PORVs used to perform safety-related functions are reliable.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25372/FINAL SER CE80 CH 20; 25373/NUREG 0933; 25375/NRC GENERIC LETTER 90-06; 25422/NUREG 1316; 25603/NRC MEMORANDUM 11/16/89, from E.S. Beckjord to F.P. Gillespie

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**Integrated Impact Number:** 1304      **SRP Section Number:** 6.8

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures regarding functions of a depressurization system to address beyond design basis events and mitigate the potential for high pressure core melt ejection during a severe accident.

NRC Policy Statement 50 FR 32138, "Policy Statement On Severe Reactor Accidents Regarding Future Designs and Existing Plants," requires that new designs, as well as custom plants that are variations of the current generation of LWRs, achieve a higher standard of severe accident safety performance.

SECY 90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," and SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," both contain recommendations by the staff to incorporate a reliable depressurization system in advanced LWRs to mitigate the potential for high pressure core melt ejection during a severe accident. The staff additionally stated in both SECY papers that equipment used to mitigate severe accidents "must be designed so there is reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed." These staff recommendations were accepted by the Commission in their SRMs dated June 26, 1990 and July 21, 1993.

The System 80+ FSER describes how the depressurization system is designed for beyond DBA

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events and meets the SECY 90-016 and 93-087 criteria to mitigate the potential for high pressure core melt ejection from severe accidents.

Generic Letter 88-20 discusses the requirement for plants to conduct systematic evaluations (PRAs) to identify any plant specific vulnerabilities to severe accidents.

The EPRI URD FSER includes staff conclusions that the depressurization system should be defined by DBA requirements as well as beyond design basis events. The EPRI URD FSER also states that the depressurization system should preclude containment challenges through direct containment heating from a severe accident and be capable of supporting primary feed and bleed during a total loss of feedwater and severe accident scenarios.

Consideration should be given to developing Acceptance Criteria and Review Procedures regarding functions of a depressurization system to address beyond design basis events and mitigate the potential for high pressure core melt ejection during a severe accident.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25376/SECY 90-016; 25377/SECY 93-087; 25378/FINAL SER EPRI CH 5; 25379/FINAL SER EPRI CH 5; 25380/FINAL SER CE80 CH 6; 25384/NRC GENERIC LETTER 88-20; 25403/NRC POLICY STATEMENT 50 FR 32138

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**Integrated Impact Number:** 1305      **SRP Section Number:** 4.5.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and Review Procedures to identify acceptable alternatives to compliance with Regulatory Guide 1.44 controls to verify non-sensitization of austenitic stainless steel materials and weldments. SRP Section 4.5.2 cites Regulatory Guide 1.44 as criteria for preventing intergranular corrosion austenitic stainless steel components.

In the CE System 80+ FSER, the staff indicated that the applicant proposes to allow the continued reference to American Society for Testing and Materials (ASTM) A-708 for the purpose of maintaining the qualifications of older weld procedures. ASTM A-262, Practice A or E are the methods explicitly recommended in RG 1.44, and have been used by the applicant since the mid-1970s for verifying non-sensitization of austenitic stainless steel materials and weldments. The applicant qualified weld procedures developed prior to the mid-1970s and used A-708 in their qualifications. These weld procedures are still in use. There have been no intergranular stress corrosion cracking (IGSCC) failures involving stainless steel weldments in the applicant's nuclear steam supply system (NSSS) units. In addition, the staff has allowed in SRP 4.5.1, paragraph III.2, ASTM A-708 as an acceptable alternative test for ASTM A-262,

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Practice A or E.

ROC 846 addresses this issue in SRP Section 5.2.3, subsection II.4.a via addition of a description of use of ASTM A-708 as an acceptable alternative to compliance with Regulatory Guide 1.44 controls to verify non-sensitization of austenitic stainless steel materials and weldments.

Consideration should be given to revising Acceptance Criteria and Review Procedures to identify acceptable alternatives to compliance with Regulatory Guide 1.44 controls to verify non-sensitization of austenitic stainless steel materials and weldments.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

1842/RG 1.44; 23483/C&S: ASTM A262; 25350/FINAL SER CE80 CH 4

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**Integrated Impact Number:** 1306      **SRP Section Number:** 6.8

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for applicable General Design Criteria regarding designing the depressurization system to withstand natural phenomena and environmental and dynamic effects, and to incorporate redundant and independent power supplies.

The following 10 CFR 50 Appendix A General Design Criteria (GDC) are applicable to the design of structures, systems, and components (SSC) important to safety: GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that SSC be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. GDC 4, "Environmental and Dynamic Effects Design Bases," requires that SSC be designed to withstand expected environmental conditions and dynamic effects without loss of capability to perform their safety functions. GDC 17, "Electric Power Systems," requires that SSC be designed with power supplies that provide sufficient redundancy and independence to assure that SSC function properly to protect and cool the reactor core. These GDC were selected based on the criteria used by the staff in the ABWR and System 80+ FSERs to review the depressurization system.

As noted in the System 80+ FSER, the Rapid Depressurization System (RDS) meets GDCs 2, 4, and 17.

Consideration should be given to developing Acceptance Criteria and Review Procedures for applicable General Design Criteria regarding designing the depressurization system to withstand



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natural phenomena and environmental and dynamic effects, and to incorporate redundant and independent power supplies.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25394/FINAL SER CE80 CH 3; 25395/FINAL SER CE80 CH 6; 25397/CODE OF FED. REGS 10CFR50; 25398/CODE OF FED. REGS 10CFR50; 25399/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1313      **SRP Section Number:** 13.6

#### **Suggested Changes to the SRP Section:**

This is a placeholder integrated impact and will not be processed further.

Revise the Acceptance Criteria to cite the latest version of ANSI N18.17.

SRP Section 13.6 cites ANSI N18.17 in the Acceptance Criteria. The latest version of the standard is ANS 3.3-1988.

Consider updating SRP Section 13.6 to cite ANS 3.3-1988.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24034/C&S: ANSI N18.17

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**Integrated Impact Number:** 1316      **SRP Section Number:** 14.2

#### **Suggested Changes to the SRP Section:**

Revise SRP Section 14.2 to provide review of the twelve topics related to the initial test program for which SAR information is requested in subsections of Regulatory Guide 1.70, Section 14.2. SRP Section 14.2 is currently organized to provide review of only eight of the twelve topics covered in Regulatory Guide 1.70.

Regulatory Guide 1.70 identifies Organization and Staffing; Conduct of the Test Program; Review, Evaluation, and Approval of Test Results; and Test Records as four additional topics which the FSAR description of the initial test program should cover and provides information content guidance related to these topics.

In the ABWR and CE System 80+ FSERs, in conjunction with its review of the applicant's

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initial test program description, the staff explicitly reviewed the four additional areas/topics identified above and guided the applicants to clarify information in these areas where necessary to provide an acceptable overall initial test program description reflecting adequate initial testing commitments.

Consideration should be given to revising SRP Section 14.2, including Areas of Review, Acceptance Criteria, and Review Procedures, to provide review of the additional areas/topics identified above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22739/FINAL SER CE80 CH 14; 25462/FINAL SER ABWR CH 14; 25463/RG 1.70

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**Integrated Impact Number:** 1317      **SRP Section Number:** 14.2

#### **Suggested Changes to the SRP Section:**

Develop Review Procedures for verification that initial testing of security systems and related features is appropriately considered and addressed in the initial test program. SRP Section 14.2 and Regulatory Guide 1.68 (a principal source of guidance cited in SRP Section 14.2 as criteria for initial test programs) do not explicitly discuss initial testing of security systems and features.

In the EPRI Evolutionary Plant FSER, the staff indicated that during review of a COL application, it would address matters such as establishment of security boundaries under the startup test program and the detailed startup schedule for NRC review and approval of the installed security system for the operating phase before the first fuel loading. The staff also stated that it expects the COL licensee to have confirmed, based upon tests conducted under realistic operating conditions of sufficient duration, that security systems described in security plans have achieved operational status and are available for NRC inspection at least 60 days before loading fuel.

In the ABWR FSER, the staff indicated that initial testing related to utility specific aspects of the plant will be necessary to satisfy certain ABWR requirements. The staff explicitly identified the site security plan as an example of systems that may require such testing.

In the CE System 80+ FSER, the staff reviewed the applicant's description of proposed security lighting and security radio system tests in conjunction with review of the initial test program description. The staff identified test description deficiencies, requested related additional information, and determined, based upon additional information provided by the applicant and subsequent CESSAR-DC amendments, that the test descriptions provided for the above security systems were acceptable. The applicant proposed to defer description of tests for the rest of the

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security system for COL action. The staff accepted the proposed deferral and established a COL action item to provide detailed description of the tests and acceptance criteria for rest of the security system in the COL applicant's security plan.

Consideration should be given to developing Review Procedures for verification that initial testing of security systems and related features is appropriately considered and addressed in the initial test program.

Consideration should also be given, as a future work item, to revision of Regulatory Guide 1.68 and/or a suitable Division 5 Regulatory Guide to include guidance which identifies the security systems and features which should be initially tested and to describe the associated initial testing which should be conducted. Future work will be tracked by IPD-7.0 Form Number 14.2-4.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22740/FINAL SER CE80 CH 14; 22743/FINAL SER EPRI CH 1; 25466/FINAL SER ABWR CH 14

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**Integrated Impact Number:** 1318      **SRP Section Number:** 6.8

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures to ensure that applicable TMI Action Plan items for PORVs and PORV block valves are addressed in the design of Reactor Coolant Depressurization Systems (RCDS).

The RCDS may utilize PORVs and PORV block valves to accomplish parts of its safety functions. 10 CFR 50.34, "Contents of Applications; Technical Information," paragraph (f), "Additional TMI-Related Requirements," contains several items that are applicable to PORVs and PORV block valves. NUREG-0737, "Post TMI Requirements," October 30, 1980, provides clarification and amplification for some of these items.

TMI Action Item II.K.3.3 requires licensees to report safety and relief valve challenges and failures. Although NUREG-0737 does not provide amplification for this item, Generic Letter 83-02 provides a description of this item and includes sample technical specifications that the staff considers adequate to address this issue.

Consideration should be given to developing Acceptance Criteria and Review Procedures to ensure that TMI Action Plan items for PORVs and PORV block valves are addressed in the design of Reactor Coolant Depressurization Systems.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25369/NUREG 0737; 25370/NUREG 0933; 25371/CODE OF FED. REGS 10CFR50;  
25374/FINAL SER CE80 CH 20; 25470/NRC GENERIC LETTER 83-02

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**Integrated Impact Number:** 1321      **SRP Section Number:** 3.7.2

#### **Suggested Changes to the SRP Section:**

Revise Areas of Review and Acceptance Criteria for allowable damping values.

SRP Section 3.7.2 reviews overall critical damping values using Regulatory Guide 1.61 as guidance. The EPRI Evolutionary Plant FSER, the ABWR FSER and the CE80+ FSER all allow alternate damping values derived from the Regulatory Guide 1.84 conditional acceptance of ASME Code Case N-411-1. This code case allows damping values which are somewhat less restrictive than the values provided by Appendix N to Section III, Division 1 of the ASME Code or by Regulatory Guide 1.61.

Modify the Areas of Review and Acceptance Criteria to accept ASME Code Case N-411-1, as conditioned by Regulatory Guide 1.84, as a source of damping values.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

21627/FINAL SER EPRI CH 1; 21876/FINAL SER CE80 CH 3; 21891/FINAL SER CE80  
CH 3; 25555/NUREG 0933; 25556/C&S: ASME N-411; 25557/FINAL SER ABWR CH 3

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**Integrated Impact Number:** 1322      **SRP Section Number:** 4.2

#### **Suggested Changes to the SRP Section:**

Modify acceptance criteria to include the use of the analytical models presented in ANSI/ANS 5.4 as an acceptable method for calculating the fractional release of volatile fission products from oxide fuel during steady-state conditions.

The NRC has given a de facto approval for the use of the ANS 5.4 model in licensing applications. The model was approved in a safety evaluation of the impact of radioactive gas released from the fuel-cladding gap as a result of a postulated fuel handling accident in a licensing amendment for St. Lucie Unit 2. SPR 4.2 does not currently address ANS 5.4.

Consideration should be given to including ANS 5.4 as guidance for predicting the fission product inventory in the void volume of a fuel rod and should be cited as a reference.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25553/C&S: ANS 5.4; 25554/NRC LETTER From Tourigay, E.G. to Woody, C.O.

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**Integrated Impact Number:** 1323      **SRP Section Number:** 3.2.2

#### **Suggested Changes to the SRP Section:**

Add quality group and seismic classification information for BWR main steam systems and alternate steam leakage paths in SRP Section 3.2.2, Appendix A.

In the ABWR FSER, the staff indicates that a main steam isolation valve leakage control system is not used. Instead, the design relies on the use of an alternative leakage path that takes advantage of the large volume and surface area in the main steam piping, main steam drain lines, turbine bypass line, and condenser to hold up and plate out the release of fission products following a design-basis accident. In this manner, the alternative leakage path and condenser are used to mitigate the consequences of an accident and are required to remain functional during and after an SSE. For this reason, the staff stated new positions related to quality group and seismic classification of portions of the main steam system, main steam drain lines, turbine system, and condenser, which are discussed in the FSER and in Section II.E of SECY-93-087. The position discussed in SECY-93-087 was approved by the Commission in its SRM dated July 21, 1993. With respect to the positions for quality group classification and application of quality group B requirements to some non-safety-related portions of the main steam system, the positions appear to be consistent with the positions of Appendix A of SRP Section 3.2.2 which are applied for current BWRs (except for BWR-6 designs).

The staff positions related to BWR seismic classification of the main steam system and alternative leakage path described above involve seismic Category I classification for portions of the main steam system within the main steam tunnel up to a seismic interface restraint and non-safety/seismic classification but dynamic seismic analyses for steam piping and alternative leakage path components within the turbine building. The affected turbine building steam piping is also subject to quality group classification and/or enhanced quality assurance provisions not normally required for non-safety-related portions of systems. A figure depicting the overall staff positions on quality group and seismic classification for the BWR main steam system and alternative leakage path would assist both quality group and seismic classification reviewers. Since the staff position on quality group classification of the main steam system and alternate leakage path appear consistent with the positions reflected in SRP Section 3.2.2, Appendix A, Appendix A appears to be the logical location for such a figure. It should also be noted that the SRP-UDP draft revision of SRP Section 3.2.1, "Seismic Classification," references SRP Section 3.2.2 for such information.

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Consideration should be given to adding quality group and seismic classification information for BWR main steam systems and alternate steam leakage paths in SRP Section 3.2.2, Appendix A. Addition of a figure appears to be the most effective means of adding this information.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23016/SECY 93-087; 25134/FINAL SER ABWR CH 3; 25511/FINAL SER ABWR CH 10;  
25613/FINAL SER ABWR CH 3; 25614/FINAL SER ABWR CH 3

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**Integrated Impact Number:** 1324      **SRP Section Number:** 3.2.2

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to update the list of fluid systems important to safety for current and evolutionary plants.

SRP Section 3.2.2 currently does not list the containment isolation system as important to safety. In addition, SRP Section 3.2.2 does not explicitly list the ultimate heat sink, makeup water sources for fuel pool cooling, essential chilled water systems, essential portions of equipment and floor drainage systems, the BWR nuclear boiler system, nor essential closed cooling water systems which provide functions other than or in addition to containment cooling for BWRs.

Chapter 5 of the SRP addresses the BWR nuclear boiler system as important to safety and the RCPB as quality group A or B as appropriate. In the ABWR FSER, the staff indicated that the nuclear boiler system is a part of the reactor coolant system.

SRP Section 6.2.4 addresses the containment isolation system as important to safety.

SRP Section 6.3 refers to the class of systems which includes BWR high and low core spray systems (terminology currently used in SRP Section 3.2.2) as "Emergency Core Cooling Systems."

SRP Section 9.1.3 addresses makeup water systems for spent fuel pool cooling as important to safety.

SRP Section 9.2.1 uses the "Station Service Water System" nomenclature to describe systems addressed in SRP Section 3.2.2 by other nomenclatures.

SRP Section 9.2.2 describes essential portions of reactor auxiliary cooling water systems and provides quality group classification positions applicable to essential portions of such systems. In the ABWR and CE System 80+ FSERs, SRP Section 9.2.2 was also used for review of chilled

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water systems.

SRP Section 9.2.5 addresses the ultimate heat sink and support systems as important to safety and includes findings that these systems are classified as Quality Group C.

SRP Section 9.3.2 refers to "sampling systems" (terminology currently used in SRP Section 3.2.2) as "Process and Post-Accident Sampling Systems."

SRP Section 9.3.3 defines the essential portions of equipment and floor drainage systems and provides quality group classification positions applicable to essential portions of such systems.

In the ABWR and CE System 80+ FSERs, the staff's discussion of its review of systems design and classification reflects that the above discussed references were used in part to verify acceptable classification of systems and components including new systems (or systems for which new nomenclature is used) performing functions considered important to safety.

Consideration should be given, based upon the above information, to revising Review Procedures to update the list of fluid systems important to safety for current and evolutionary plants. The revised lists should reflect the fluid systems important to safety for which positions classifying the system as quality group C or higher exist. When revising the list of systems important to safety, existing SRP system nomenclature should be used to the extent possible.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25482/FINAL SER CE80 CH 3; 25485/FINAL SER CE80 CH 9; 25486/FINAL SER CE80 CH 9; 25487/FINAL SER CE80 CH 6; 25489/FINAL SER CE80 CH 6; 25490/FINAL SER CE80 CH 6; 25491/FINAL SER CE80 CH 9; 25494/FINAL SER CE80 CH 9; 25500/FINAL SER CE80 CH 9; 25517/FINAL SER ABWR CH 9; 25520/FINAL SER ABWR CH 9; 25521/FINAL SER ABWR CH 9; 25522/FINAL SER ABWR CH 9; 25527/FINAL SER ABWR CH 9; 25529/FINAL SER ABWR CH 9; 25531/FINAL SER ABWR CH 6; 25551/FINAL SER ABWR CH 6; 25552/FINAL SER CE80 CH 6; 25615/FINAL SER ABWR CH 1

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**Integrated Impact Number:** 1327      **SRP Section Number:** 4.3

#### **Suggested Changes to the SRP Section:**

Modify Review Procedures to verify that the effective delayed neutron fraction, prompt neutron lifetime, reactivity coefficients, and rod worths used in the reactivity insertion analyses reviewed under SRP Sections 15.4.8 and 15.4.9 are appropriate and based on limiting values. This SRP section reviews the nuclear design for conformance to relevant General Design Criteria (GDC).

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In particular, reactivity coefficients, control rod worths, and other core physics values are reviewed.

The Evaluation Findings section addresses compliance with GDC 28, "Reactivity Limits," requiring that the reactivity control system is designed to limit the amount of reactivity or rate of reactivity increase to prevent damage to the reactor coolant pressure boundary and reactor core. However, the Review Procedures are not written to include a verification that limiting reactivity coefficients, rod worths, and other core physics values were used in the reactivity insertion accident analyses.

Consider modifying the Review Procedures to include a verification that reactivity coefficients, control rod worths, and pertinent core physics values are appropriately used in the analysis of reactivity insertion accidents.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

13929/RG 1.77; 14877/CODE OF FED. REGS 10CFR50; 25585/FINAL SER CE80 CH 4

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**Integrated Impact Number:** 1329      **SRP Section Number:** 3.2.2

#### **Suggested Changes to the SRP Section:**

Add quality group classification guidance and references supplemental to Reg. Guide 1.26 for containment isolation systems in SRP Section 3.2.2, Appendix C.

SRP Section 3.2.2 currently does not list the containment isolation system as important to safety and does not reflect guidance supplemental to Reg. Guide 1.26 which exists for quality group classification of system components and application of quality standards thereto.

Reg. Guide 1.141 describes a method acceptable to the NRC staff for complying with the Commission's requirements with respect to containment isolation of fluid systems, including the application of ANSI N271-1976 for containment isolation of fluid systems that penetrate the primary containment of light-water-cooled reactors. The Reg. Guide is used as specific criteria in SRP Section 6.2.4 for definition of essential and non-essential systems with respect to TMI Action Plan Requirement II.E.4.2. SRP Section 6.2.4 provides detailed positions, which appear supplemental to Reg. Guide 1.26, for the quality group classification of the containment isolation system. Branch Technical Position CSB 6-3 (an attachment of SRP Section 6.2.3) also provides a position related to the quality group classification of closed systems relied upon as containment bypass leakage boundaries. Reg. Guide 1.11 describes a suitable basis which may be used for demonstrating the acceptability of instrument lines penetrating primary containment and includes positions that the quality of such lines should be at least equivalent to that of the



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containment. The Reg. Guide is also used as specific criteria in SRP Section 6.2.4.

In the ABWR and CE System 80+ FSERs, the staff's discussion of its review of containment isolation system design and classification reflects that the above discussed references were used in part to verify acceptable classification of containment isolation systems and their components.

Consideration should be given to adding quality group classification information supplemental to Reg. Guide 1.26 for containment isolation systems in SRP Section 3.2.2, Appendix C, based upon the regulatory documents identified above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25487/FINAL SER CE80 CH 6; 25531/FINAL SER ABWR CH 6; 25549/RG 1.11; 25550/RG 1.141; 25551/FINAL SER ABWR CH 6; 25552/FINAL SER CE80 CH 6

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**Integrated Impact Number:** 1330      **SRP Section Number:** 3.2.2

#### **Suggested Changes to the SRP Section:**

Add quality group classification guidance and references supplemental to Reg. Guide 1.26 for the ultimate heat sink and associated support systems in SRP Section 3.2.2, Appendix C.

SRP Section 3.2.2 currently does not list the ultimate heat sink and associated support systems as important to safety and does not reflect guidance supplemental to Reg. Guide 1.26 which exists for quality group classification of ultimate heat sink support system components and application of quality standards thereto.

SRP Section 9.2.5 addresses the ultimate heat sink and support systems as important to safety and includes findings that these systems are classified as Quality Group C. Reg. Guide 1.72 prescribes supplemental quality measures, including application of an ASME Code case as supplemented by the regulatory positions (which include a position that ASME Code Class 3 requirements be applied for inspection purposes), for safety-related spray pond piping made from reinforced thermosetting resin.

In the ABWR FSER, the staff evaluated the ultimate heat sink design interface requirements against SRP Section 9.2.5 and also indicated that the requirements of Reg. Guide 1.72 apply to the design of a spray pond as an ultimate heat sink.

Consideration should be given to adding quality group classification information supplemental to Reg. Guide 1.26 for the ultimate heat sink and associated support systems in SRP Section 3.2.2, Appendix C, based upon the regulatory documents identified above.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

12963/RG 1.72; 25612/FINAL SER ABWR CH 9

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**Integrated Impact Number:** 1331      **SRP Section Number:** 3.2.2

#### **Suggested Changes to the SRP Section:**

Add quality group classification guidance and references supplemental to Reg. Guide 1.26 for the equipment and floor drainage system in SRP Section 3.2.2, Appendix C.

SRP Section 3.2.2 currently does not list the equipment and floor drainage system as important to safety and does not reflect guidance supplemental to Reg. Guide 1.26 which exists for quality group classification of equipment and floor drainage system components and application of quality standards thereto.

SRP Section 9.3.3 provides criteria defining the safety-related portions of equipment and floor drainage systems and provides quality group classification positions applicable to safety-related portions of such systems which appear supplemental to the guidance of Reg. Guide 1.26.

In the ABWR and CE System 80+ FSERs, the staff evaluated equipment and floor drainage systems against SRP Section 9.3.3 and verified that safety-related portions of the systems were classified as quality group C or higher.

Consideration should be given to adding quality group classification information supplemental to Reg. Guide 1.26 for the equipment and floor drainage system in SRP Section 3.2.2, Appendix C, based upon the information in SRP Section 9.3.3 identified above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25482/FINAL SER CE80 CH 3; 25486/FINAL SER CE80 CH 9; 25500/FINAL SER CE80 CH 9; 25529/FINAL SER ABWR CH 9

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**Integrated Impact Number:** 1333      **SRP Section Number:** 17.3

#### **Suggested Changes to the SRP Section:**

Add identification of Generic Letter 85-06 as criteria for quality assurance controls to be provided for nonsafety-related equipment required by the ATWS rule, 10 CFR 50.62.

Section 50.62(d) of the ATWS rule requires that each licensee develop and submit (to the

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Director of the Office of Nuclear Reactor Regulation) a proposed schedule for meeting the requirements of the rule within 180 days after issuance of QA guidance applicable to equipment required by the rule. NRC Generic Letter 85-06 provides the staff's QA guidance for non-safety-related equipment encompassed by the ATWS rule. The staff's guidance consists of the application of quality controls comparable to selected portions of a 10 CFR 50, Appendix B program. The lesser safety significance of the equipment encompassed by 10 CFR 50.62, as compared to safety-related equipment, necessarily resulted in less stringent QA guidance.

Consideration should be given to adding identification of Generic Letter 85-06 as criteria for quality assurance controls to be provided for nonsafety-related equipment required by the ATWS rule.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25640/NRC GENERIC LETTER 85-06; 25655/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1335      **SRP Section Number:** 5.2.1.1

#### **Suggested Changes to the SRP Section:**

Revise SRP Section 5.2.1.1 to comply with 10 CFR 50.55a(3).

Subsection III of SRP Section 5.2.1.1, second paragraph, second sentence, states: "A decision to accept a component which is not fully in compliance with the rules is based on a judgment of the relative importance of the specific provisions in the Code or Code Addenda not complied with, and a determination that any noncompliance will not result in an unacceptable level of safety and quality." This guidance is no longer in accordance with the provisions of 10 CFR 50.55a(3) that now allow the Director of NRR to approve certain alternatives to the requirements of 10 CFR 50.55a.

Consider revising the SRP to conform to the provisions of 10 CFR 50.55a.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20478/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1336      **SRP Section Number:** 6.5.5

#### **Suggested Changes to the SRP Section:**

Revise the Review Procedures to address the potential for errors in the calculation of fission

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product removal from containment.

In their review of the draft SRP Sections 6.2.5 and 6.5.5, the NRC identified the potential for errors to occur in calculating fission product removal in accordance with the existing SRP section guidance. The Acceptance Criteria in SRP Section 6.5.5 states that the assumptions concerning the release of radioactivity are to be taken from Regulatory Guide 1.3. In a memorandum from R. Hermann of the Materials and Chemical Engineering Branch, to R.W. Borchart of the Inspection Program Branch, the NRC concludes that use of Regulatory Guide 1.3 values in conjunction with the procedures in the SRP section, will result in double counting of fission product removal by plateout. This double counting error would result in nonconservative estimates of fission product removal.

Consider revising the Review Procedures to identify the potential for double counting of the fission product plateout removal mechanism if Regulatory Guide 1.3 values are utilized in the calculational procedures described in the Review Procedures subsection of the SRP.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25664/NRC MEMORANDUM 09/21/95, from Borchart, R to Hermann R

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**Integrated Impact Number:** 1338      **SRP Section Number:** 6.5.2

#### **Suggested Changes to the SRP Section:**

Revise the Review Procedures to address the potential for errors in the calculation of fission product removal from containment.

In their review of the draft SRP Section 6.2.5, the NRC identified the potential for an error to occur in calculating fission product removal in accordance with the existing SRP section guidance. The Review Procedures in SRP Section 6.5.2 state that the amounts of fission products released to the containment space are obtained from Regulatory Guides 1.3 and 1.4. In a memorandum from R. Hermann of the Materials and Chemical Engineering Branch, to R.W. Borchart of the Inspection Program Branch, the NRC concludes that use of values from Regulatory Guides 1.3 or 1.4 in conjunction with the procedures in the SRP section, will result in double counting of fission product removal by plateout. This double counting error would result in nonconservative estimates of fission product removal.

Consider revising the Review Procedures to identify the potential for double counting of the fission product plateout removal mechanism if Regulatory Guide 1.3 or 1.4 values are utilized in the calculational procedures described in the Review Procedures subsection of the SRP.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25666/NRC MEMORANDUM 09/21/95, from Borchart, R to Hermann R

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**Integrated Impact Number:** 1339      **SRP Section Number:** 3.9.4

#### **Suggested Changes to the SRP Section:**

Add further review procedures for review of the adequacy of the design of BWR control rod drive return lines.

SRP Section 3.9.4 provides review of the adequacy of control rod drive system pressure boundary components and includes review procedures addressing review of BWR control rod drive hydraulic system piping.

Generic Letter 80-95 directs implementation of the final edition (November 1980) of NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," which provides the staff's resolution of Unresolved Safety Issue A-10.

NUREG-0619 describes the technical issues, the studies and the analyses performed by the General Electric Company and the NRC staff, the staff's technical positions based on these studies, and the staff's requirements for licensee and applicant implementation of the technical positions. Staff positions are detailed with respect to specific repairs and modification options for BWR control rod drive return lines to address observed thermal cycling crack generation at associated reactor vessel nozzles. Associated inspections to demonstrate the integrity of piping and nozzles affected by modifications and/or repairs and tests to demonstrate adequate return flow and overall control rod drive system operability are also specified in the NUREG report.

Consideration should be given to adding further review procedures for review of the adequacy of the design of BWR control rod drive return lines based on Generic Letter 80-95 and NUREG-0619.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25624/NRC GENERIC LETTER 80-95

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**Integrated Impact Number:** 1340      **SRP Section Number:** 2.4.2

**Suggested Changes to the SRP Section:**

In SRP Section 2.4.2, under AREAS OF REVIEW, factors that are considered include seismically induced dam failures and maximum water level at site from failure during an operating basis earthquake (OBE) coincident with a standard project flood. The Commission has approved specific staff positions and criteria for elimination of the OBE (see Section 1.M of SECY 93-087 and the associated SRM.) An assessment should be performed regarding the continuing need for a flood analysis in the SAR assuming coincident floods and earthquakes of moderately high probability of occurrence. Depending upon the outcome of this assessment, the coincident OBE and standard project flood addressed in this subsection should be eliminated or replaced with an appropriate combination of flood and seismic event.

**Potential Impacts/Documents supporting the Suggested Changes:**

22996/SECY 93-087

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**Integrated Impact Number:** 1341      **SRP Section Number:** 2.4.4

**Suggested Changes to the SRP Section:**

In SRP Section 2.4.4, under AREAS OF REVIEW, factors that are considered include seismically induced dam failures and maximum water level at site from failure during an operating basis earthquake (OBE) coincident with a standard project flood. The Commission has approved specific staff positions and criteria for elimination of the OBE (see Section 1.M of SECY 93-087 and the associated SRM.) An assessment should be performed regarding the continuing need for a flood analysis in the SAR assuming coincident floods and earthquakes of moderate probability. Depending upon the outcome of this assessment, the coincident OBE and standard project flood addressed in this subsection should be eliminated or replaced with an appropriate combination of flood and seismic event.

**Potential Impacts/Documents supporting the Suggested Changes:**

22965/SECY 93-087

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**Integrated Impact Number:** 1343      **SRP Section Number:** 2.4.13

#### **Suggested Changes to the SRP Section:**

10 CFR Part 20 was substantially revised on May 21, 1991 (56 FR 23396).

SRP Section 2.4.13 refers to 10 CFR Part 20, Appendix B, Table II, in three places.

Review the changes to 10 CFR Part 20 and the specific reference to Appendix B above to consider an appropriate update to the references to 10 CFR Part 20 in SRP Section 2.4.13.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23733/CODE OF FED. REGS 10CFR20

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**Integrated Impact Number:** 1344      **SRP Section Number:** 3.6.2

#### **Suggested Changes to the SRP Section:**

Revise SRP Section 3.6.2 to include a discussion of the exemption, accepted by the NRC in the evolutionary FSERs, which allowed the designers to eliminate operating basis earthquake (OBE) loads from some design analyses.

In SECY 90-016 the staff discussed its proposal to decouple the OBE from the safe-shutdown earthquake (SSE). The staff recommended that the Commission approve the review approach to consider requests to decouple the OBE from the SSE on a design-specific basis for evolutionary designs. In the SRM for SECY 90-016 the Commission approved the staff's recommendation.

In SECY 93-087 the staff recommended that the Commission approve their approach to eliminate the OBE from the design of systems, structures, and components. When the OBE is eliminated from the design, no replacement earthquake loading should be used to establish the postulated pipe rupture and leakage crack locations. The staff recommended that the criteria for postulating pipe ruptures and leakage cracks in high- and moderate-energy piping systems be based on factors attributed to normal and operational transients alone. However, for establishing pipe breaks and leakage cracks due to fatigue effects, calculation of the cumulative usage factor should continue to include seismic cyclic effects. In its July 21 1993, SRM, the Commission approved and agreed with the staff's positions.

Consider revising Branch Technical Position MEB 3-1 Rev. 2 to incorporate a brief discussion of the exemption that was granted in the evolutionary FSERs along with some discussion regarding the fact that no replacement earthquake loading was considered to establish postulated

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pipe rupture and leakage crack locations.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23018/SECY 93-087; 23756/ADVANCE SER ABWR CH 3; 23757/ADVANCE SER CE CH 3

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**Integrated Impact Number:** 1345      **SRP Section Number:** 3.8.2

#### **Suggested Changes to the SRP Section:**

In SECY 90-016 the staff discussed its proposal to decouple the operating-basis earthquake (OBE) from the safe-shutdown earthquake (SSE). The staff recommended that the Commission approve the review approach to consider requests to decouple the OBE from the SSE on a design-specific basis for evolutionary designs. In the SRM for SECY 90-016 the Commission approved the staff's recommendations.

In SECY 93-087 the staff proposed additional guidance and positions for the design of structures, systems, and components (SSC) when the OBE is eliminated. The staff evaluated the effect on safety of eliminating the OBE from the design load combinations for selected SSC and has developed proposed criteria for an analysis using only the SSE. In the SRM for SECY 93-087 the Commission approved specific staff positions and criteria for elimination of the OBE and agreed that these criteria should also apply to passive ALWRs. The specific approved positions and guidance are contained in section I.M of SECY 93-087 and the associated SRM for SECY 93-087.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23048/SECY 93-087

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**Integrated Impact Number:** 1346      **SRP Section Number:** 3.8.3

#### **Suggested Changes to the SRP Section:**

Incorporate in the REVIEW PROCEDURES information regarding the exemption, accepted by the Staff in the evolutionary FSERs, allowing the elimination of the Operating Basis Earthquake (OBE) from seismic design considerations.

In SECY 90-016 the staff discussed its proposal to decouple the operating-basis earthquake (OBE) from the safe-shutdown earthquake (SSE). The staff recommended that the Commission approve the review approach to consider requests to decouple the OBE from the SSE on a



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design-specific basis for evolutionary designs. In the SRM for SECY 90-016 the Commission approved the staff's recommendation.

In SECY 93-087 the staff proposed additional guidance and positions for the design of structures systems and components (SSC) when the OBE is eliminated. The staff evaluated the effect on safety of eliminating the OBE from the design load combinations for selected SSC and has developed proposed criteria for an analysis using only the SSE. In the SRM for SECY 93-087 the Commission approved specific staff positions and criteria for the elimination of the OBE. The specific approved positions and guidance can be found in section I.M of SECY 93- 087 and the associated SRM for SECY 93-087.

As discussed in the ABWR and CE 80+ FSERs, the Staff accepted an exemption allowing the designers to eliminate the OBE from the design basis for structures, systems, and components (SSCs). In this case, the OBE serves only as an inspection level earthquake above which the licensee would shut down the plant and inspect for potential damage to SSCs.

Consider eliminating citations to the OBE in SRP Section 3.8.3. Consider providing information in the REVIEW PROCEDURES relating to the staff's acceptance of an exemption eliminating the OBE from design requirements.

(NOTE: Part A of this ROC has been changed upon receipt of guidance regarding the incorporation of evolutionary plant issues. Information related to exemptions accepted by the Staff in the evolutionary FSERs should be provided in REVIEW PROCEDURES as an informational item.)

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23049/SECY 93-087; 25787/FINAL SER ABWR CH 3; 25788/FINAL SER CE80 CH 3;  
25789/SECY 90-016

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**Integrated Impact Number:** 1347      **SRP Section Number:** 3.8.5

#### **Suggested Changes to the SRP Section:**

Incorporate in the REVIEW PROCEDURES information regarding the exemption, accepted by the Staff in the evolutionary FSERs, allowing the elimination of the Operating Basis Earthquake (OBE) from seismic design considerations.

Item I.M of SECY-93-087 states that the Commission approves the staff's recommendation to eliminate the OBE from the design of systems, structures, and components for evolutionary and advanced ALWRs.

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The SECY further states that with the elimination of the OBE the equipment should be qualified in accordance with the provisions of the IEEE Standard 344-1987, with either (1) five one-half SSE events followed by one full SSE event, or (2) a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events, in accordance with Appendix D of IEEE Standard 344-1987 when followed by one full SSE.

As discussed in the ABWR and CE 80+ FSERs, the Staff accepted an exemption allowing the designers to eliminate the OBE from the design basis for structures, systems, and components (SSCs). In this case, the OBE serves only as an inspection level earthquake above which the licensee would shut down the plant and inspect for potential damage to SSCs.

Consider providing information in the REVIEW PROCEDURES relating to the staff's acceptance of an exemption eliminating the OBE from design requirements.

(NOTE: Part A of this ROC has been changed upon receipt of guidance regarding the incorporation of evolutionary plant issues. Information related to exemptions accepted by the Staff in the evolutionary FSERs should be provided in REVIEW PROCEDURES as an informational item.)

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23051/SECY 93-087; 25906/FINAL SER ABWR CH 3; 25907/FINAL SER CE80 CH 3;  
25908/SECY 90-016

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**Integrated Impact Number:** 1348      **SRP Section Number:** 3.9.3

#### **Suggested Changes to the SRP Section:**

Revise acceptance criteria related to component supports to agree with the 1989 edition of the ASME Code. Specifically, the revision to 10 CFR 50.55a now includes Class 2 and 3 components as well as Class 1. SRP Section 3.9.3 requires revision to the acceptance criteria for Class 1, 2, and 3 component supports which now must all be designed in accordance with the Code, subsection NF. The current version of SRP Section 3.9.3 (Revision 1 - July 1981) cites RGs 1.124 and 1.130 regarding design criteria for Class 1 component supports. These RGs cite the 1974-1977 versions of the Code and do not apply to the 1989 version of the Code. 10 CFR 50.55a requires that the staff review new license applications in accordance with the 1989 version of the Code. Accordingly, SRP Section 3.9.3 must be revised.

Consider eliminating citation of RGs 1.124 and 1.130 in SRP Section 3.9.3 and substituting citations to subsection NF of the Code for the design of component supports.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

12978/CODE OF FED. REGS 10CFR50; 12981/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1349      **SRP Section Number:** 3.11

#### **Suggested Changes to the SRP Section:**

Section 50.34(f)(2)(ix) of 10 CFR describes the hydrogen control system to be provided within containment. Section 50.44 of 10 CFR, "Standards For Combustible Gas Control Systems In Light-Water-Cooled Power Reactors," describes requirements for existing plants and "Near-Term Operating License" (NTOL) plants and is not pertinent to this forthcoming revision of the SRP. The provisions of 10 CFR 50.34(f) are referred to in 10 CFR 52.79(b) with respect to combined license applications and are pertinent to this forthcoming revision of the SRP. This issue was addressed in SECY-90-016 dated January 12, 1990, and it was confirmed in a Staff Requirements Memorandum dated June 26, 1990 that the provisions of 10 CFR 50.34(f)(2)(ix) should be applicable to evolutionary light water reactors.

SRP Section 3.11 does not now include a specific discussion of necessary environmental qualification requirements for equipment to resist the environmental effects of hydrogen burning, post-accident inerting, unintended detonation or combustion, or inadvertent actuation of the post-accident inerting system.

Consider modifying SRP Section 3.11 to include a discussion of the acceptance criteria and evaluation findings regarding environmental qualification of equipment to accommodate hydrogen burning and post-accident inerting. These SRP Section 3.11 provisions should be based on the requirements of 10 CFR 50.34(f)(2)(ix).

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23921/CODE OF FED. REGS 10CFR50; 24189/C&S: IEEE 382-72

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**Integrated Impact Number:** 1351      **SRP Section Number:** 15.1.1 - 15.1.4

#### **Suggested Changes to the SRP Section:**

SRP Section 15.1.1 involves the review of four anticipated operational occurrences for increased heat removal by the secondary system. In the ABWR FSER, Section 15.1, the NRC staff cited GDC 10, GDC 15, and GDC 20 as acceptance criteria. All of these GDCs are applicable to anticipated operational occurrences.

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SRP Section 15.1.1 does not cite GDC 20 as an acceptance criterion.

Consider adding GDC 20 to Acceptance Criteria, Evaluation Findings, and References to SRP Section 15.1.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25690/FINAL SER ABWR CH 15

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**Integrated Impact Number:** 1352      **SRP Section Number:** 15.2.1 - 15.2.5

#### **Suggested Changes to the SRP Section:**

Revise acceptance criteria and references to reflect the staff's acceptance review for the ODYNA and REDYA transient analysis computer codes.

SRP Section 15.2.1-15.2.5 cites various analytical models acceptable to the staff for transient analysis (References 4 through 7). In Generic Letter 81-08, dated January 29, 1981, all BWR licensees and applicants were informed that transient analyses performed by the General Electric Co. (GE) supporting reload submittals received after February 1, 1981, must contain appropriate ODYN analyses in place of those previously performed with REDY for the limiting transients. Additional guidance was provided by Generic Letter 81-37, dated December 29, 1981. These codes have since been modified by GE for use in the analysis of limiting transients on the Advanced Boiling Water Reactor (ABWR) standard design. These modified codes, ODYNA and REDYA, have been reviewed by the NRC staff and its consultant Brookehaven National Laboratory, and approved for design analysis of the ABWR.

Consider revising the specific acceptance criteria and references to include the ODYNA and REDYA computer codes as being acceptable to the staff.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24434/NRC GENERIC LETTER 81-08; 24441/NRC GENERIC LETTER 81-37

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**Integrated Impact Number:** 1353      **SRP Section Number:** 15.2.8

#### **Suggested Changes to the SRP Section:**

The General Design Criteria 17 requires (in part) that an onsite and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel

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design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs and (2) the core is cooled and containment and other vital functions are maintained in the event of postulated accidents.

In the CE 80+ FSER, the staff stated that, in accordance with the requirements of GDC 17, LOOP should not be considered as a single failure event, but should be assumed in the analysis for each event without changing the event category. In addition, the staff stated that the applicant is required to discuss each of the transient and accident analyses to justify that the analyses meet the GDC 17 requirements.

Although SRP Section 15.2.8 mentions loss of offsite power and loss of onsite power it does not specifically address General Design Criteria 17 or provide the staff guidance recommended by Integrated Impact No. 508 for SRP Section 15.2.7.

Consider modifying SRP Section 15.2.8 Acceptance Criteria to include GDC 17, and Review Procedures to incorporate staff guidance for the assumption of the LOOP, in addition to the limiting single failure event, for the analysis of transients and accidents.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25708/DRAFT SER CE80 CH 15; 25709/CODE OF FED. REGS 10CFR50; 25710/DRAFT SER CE80 CH 15

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**Integrated Impact Number:** 1354      **SRP Section Number:** 15.3.1 - 15.3.2

#### **Suggested Changes to the SRP Section:**

Include a discussion of the unique acceptance criterion accepted in the ABWR FSER for the postulated trip of all of the reactor internal pumps with offsite power available. The transient is caused by a postulated common-mode software failure of the adjustable speed drives. The NRC staff classified this postulated event in a special category of anticipated transients involving a common-mode software failure and established a special acceptance criterion for the radiological dose calculation. (The postulated pressure regulator down-scale failure was also classified in this special category. This is a transient involving an increase in reactor pressure and is not applicable to this SRP section.)

Normally, such transients are treated as anticipated operational occurrences (AOOs) for which the acceptance criteria is that specified acceptable fuel design limits should not be exceeded.

For these special cases, the NRC staff concluded that the two transients were uncertain to occur during the plant lifetime (not AOOs, but classified as anticipated transients involving a

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common-mode software failure), and therefore established a special acceptance criterion for the radiological dose calculation: fuel failure need not be assumed in dose calculations for fuel rods that are at or below a temperature of approximately 600 degrees C (1111 degrees F) for less than 60 seconds for fuel that has achieved a burnup 20 gigawatt-days per metric ton or less. (For fuel beyond this burnup, the dose calculations must assume fuel failure for all fuel rods that achieve transition boiling because the test data does not go beyond 20 gigawatt-days per metric ton).

The resulting dose should not exceed 10 percent of 10 CFR Part 100, which the staff considered appropriate for an event of such postulated frequency. This is due to the unique design features of the ABWR instrumentation and control systems, which reduce the frequency of such events and therefore allow these events to be recategorized as special cases. Consideration should be given to discussing this unique acceptance criterion for the ABWR to SRP Section 15.3.1 15.3.2 for the postulated trip of all of the reactor internal pumps with offsite power available for the ABWR.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24525/FINAL SER ABWR CH 15

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**Integrated Impact Number:** 1355      **SRP Section Number:** 15.3.3 - 15.3.4

#### **Suggested Changes to the SRP Section:**

Revise acceptance criteria and references to reflect the staff's acceptance review for the ODYNA and REDYA transient analysis computer codes.

SRP Section 15.3.3 15.3.4 cites various analytical models acceptable to the staff for transient analysis (References 7 through 11). In Generic Letter 81-08, dated January 29, 1981, all BWR licensees and applicants were informed that transient analyses performed by the General Electric Co. (GE) supporting reload submittals received after February 1, 1981, must contain appropriate ODYN analyses in place of those previously performed with REDY for the limiting transients. Additional guidance was provided by Generic Letter 81-37, dated December 29, 1981. These codes were subsequently modified by GE for use in the analysis of limiting transients on the Advanced Boiling Water Reactor (ABWR) standard design. These modified codes, ODYNA and REDYA, have been reviewed by the NRC staff and its consultant Brookhaven National Laboratory, and approved for design analysis of the ABWR.

Consider revising the specific acceptance criteria and references to include the ODYNA and REDYA computer codes as being acceptable to the staff.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24437/NRC GENERIC LETTER 81-08; 24444/NRC GENERIC LETTER 81-37

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**Integrated Impact Number:** 1356      **SRP Section Number:** 15.4.2

#### **Suggested Changes to the SRP Section:**

General Design Criteria 17 requires (in part) that an onsite and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs and (2) the core is cooled and containment and other vital functions are maintained in the event of postulated accidents.

In the CE 80+ FSER, the staff stated that, in accordance with the requirements of GDC 17, LOOP should not be considered as a single failure event, but should be assumed in the analysis for each event without changing the event category.

Consider modifying SRP 15.4.2, Acceptance Criteria, to include the requirement of GDC 17, and Review Procedures to incorporate staff guidance for the assumption of the LOOP, in addition to the limiting single failure event, for the analysis of transients and accidents.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25723/DRAFT SER CE80 CH 15; 25724/CODE OF FED. REGS 10CFR50; 25725/DRAFT SER CE80 CH 15

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**Integrated Impact Number:** 1357      **SRP Section Number:** 15.4.9

#### **Suggested Changes to the SRP Section:**

In SSAR Section 15.4.9.6, GE states that the radiological consequences of a control rod drop accident need not be considered because such an accident is extremely unlikely with the improved design of the ABWR. Accordingly, GE did not provide an analysis of a control rod drop.

However, the NRC staff specifically evaluated this accident because it is the first application involving the ABWR standard design with hypothetical site boundaries. This evaluation should establish a reference for comparison of future applications incorporating the ABWR design.

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This analysis and its results are presented in ABWR FSER Section 15.4.1, and Tables 15.1 and 15.2.

Consider adding the staff's analysis of a control rod drop accident to Areas of Review, Review Procedures, and References to SRP Section 15.4.9 in order that it serve as a reference for comparison with future applications.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25691/FINAL SER ABWR CH 15

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**Integrated Impact Number:** 1358      **SRP Section Number:** 13.1.1

#### **Suggested Changes to the SRP Section:**

NUREG-0711, "Human Factors Engineering Program Review Model," published in July 1994, presents a model for performing design certification reviews of human factors engineering program elements. Element 1 is "HFE Program Management." The NUREG contains important factors for NRC staff reviewers to consider during reviews.

Consider incorporating relevant review considerations from NUREG-0711, particularly considerations from Element 1, into SRP Section 13.1.1 and making appropriate interfaces with Chapter 18 of the SRP.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25685/NUREG 0711

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**Integrated Impact Number:** 1359      **SRP Section Number:** 13.1.2 - 13.1.3

#### **Suggested Changes to the SRP Section:**

Revise SRP 13.1.2 to include guidance from NUREG-0711 on operating organizations as they relate to Human Factors Engineering.

NUREG-0711, "Human Factors Engineering Program Review Model," published in July 1994, presents a model for performing design certification reviews of human factors engineering program elements. NUREG-0711 presents the HFE PRM (Human Factors Engineering Program Review Model) which describes an appropriate process for considering human-system interactions in the design of nuclear power plants. This guidance document seeks to ensure (1) that applicants integrate HFE into plant development, design, and evaluation, and (2) that the



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HFE programs reflect state-of-the-art human factors principles. Important components of the review model are Element 5 - Staffing, and Element 6 - Human Reliability Analysis. Element 5 addresses staffing considerations for operating organizations and Element 6 provides guidance on appropriate considerations in the assessment of potential effects of human error.

Consider incorporating relevant review considerations from NUREG-0711, particularly considerations from Element 5 and 6, into SRP Section 13.1.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25686/NUREG 0711

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**Integrated Impact Number:** 1360      **SRP Section Number:** 13.2.1

#### **Suggested Changes to the SRP Section:**

NUREG-0711, "Human Factors Engineering Program Review Model," published in July 1994, presents a model for performing design certification reviews of human factors engineering program elements. Element 9 is "Training Program Development." The NUREG contains important factors for NRC staff reviewers to consider during reviews.

Consider incorporating relevant review considerations from NUREG-0711, particularly considerations from Element 9, into SRP Section 13.5.2 and making appropriate interfaces with Chapter 18 of the SRP.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25687/NUREG 0711

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**Integrated Impact Number:** 1361      **SRP Section Number:** 13.2.2

#### **Suggested Changes to the SRP Section:**

NUREG-0711, "Human Factors Engineering Program Review Model," published in July 1994, presents a model for performing design certification reviews of human factors engineering program elements. Element 9 is "Training Program Development." The NUREG contains important factors for NRC staff reviewers to consider during reviews.

The staff had also previously developed detailed review criteria and procedures for training program content and effectiveness in NUREG-1220, "Training Review Criteria and Procedures."

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Consider incorporating relevant review considerations from NUREG-0711 (particularly considerations from Element 9) and NUREG-1220, into SRP Section 13.2.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25688/NUREG 0711

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**Integrated Impact Number:** 1362      **SRP Section Number:** 13.2.2

#### **Suggested Changes to the SRP Section:**

Appendix E to 10 CFR 50 provides information on emergency planning and preparedness. Two sections, II.F and IV.F provide information on elements of emergency preparedness (EP) training. Section II.F requires that the PSAR shall describe provisions for a training program for licensee employees and for licensee non-employees who might assist in a radiological emergency. Section IV.F requires that EP training should include instruction and periodic drills, and be provided to among others, radiological monitoring teams, fire brigade members, first aid and damage control teams. Both of these sections relate to training for non-licensed personnel.

Consider revising section 13.2.2 to reflect these requirements for training of EP-related information.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

17504/CODE OF FED. REGS 10CFR50; 17505/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1363      **SRP Section Number:** 13.5.1.1

#### **Suggested Changes to the SRP Section:**

Regulatory Guide 1.114, "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Plant," defines an acceptable method to the NRC staff for complying with the requirement for presence of an operator at the controls and a senior operator in the control room from which a nuclear power unit is being operated. It also clarifies the boundaries of the control room as referenced in 10 CFR 73.2(h) and 10 CFR 73.55(c).

Consider adding Regulatory Guide 1.114 guidance on the definition of "surveillance area" and presence of operator at the controls and senior operator in the control room, with clarification of the boundaries of the control room to Areas of Review and Acceptance Criteria II.A, as areas where administrative procedures are needed.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

18536/RG 1.114

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**Integrated Impact Number:** 1364      **SRP Section Number:** 13.5.2.1

#### **Suggested Changes to the SRP Section:**

NUREG-0711, "Human Factors Engineering Program Review Model," published in July 1994, presents a model for performing design certification reviews of human factors engineering program elements. Element 8 is "Procedure Development." The NUREG contains important factors for NRC staff reviewers to consider during reviews.

Consider incorporating relevant review considerations from NUREG-0711, particularly considerations from Element 8, into SRP Section 13.5.2 and making appropriate interfaces with Chapter 18 of the SRP.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25689/NUREG 0711

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**Integrated Impact Number:** 1366      **SRP Section Number:** 8.3.2

#### **Suggested Changes to the SRP Section:**

Add reference to RG 1.47 to SRP Section 8.3.2.

Regulatory Guide 1.47, Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems, provides an acceptable method for meeting the requirements of IEEE 279-1971 and Criterion XIV of Appendix B to 10 CFR 50 for indicating when certain safety-related functions of a nuclear power plant are bypassed or made inoperable during the performance of periodic tests or maintenance. In addition, Branch Technical Position ICSB-21 was issued to provide additional details for implementing RG 1.47. RG 1.47 provides guidance regarding administrative procedures and indicating systems.

Consider adding references to RG 1.47 and BTP ICSB-21 to SRP Section 8.3.2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

11957/RG 1.47

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**Integrated Impact Number:** 1368      **SRP Section Number:** 15.1.5

#### **Suggested Changes to the SRP Section:**

In the CE 80+ FSER, the staff stated that, in accordance with the requirements of GDC 17, the loss-of-offsite power (LOOP) should not be considered as a single failure event, but should be assumed in the analysis for each event without changing the event category. In addition, the staff stated that the applicant is required to discuss each of the transient and accident analyses to justify that the analyses meet the GDC 17 requirements.

This action is appropriate for SRP Section 15.1.5, which addresses the review of steam system piping failures inside and outside of containment. For this series of accidents, the CE 80+ FSER presents the results of the applicant's analyses, which were performed assuming LOOP and no LOOP.

Consider modifying Acceptance Criteria to include GDC 17 and revising Review Procedures and Evaluation Findings to incorporate staff guidance for the assumption of LOOP, in addition to the limiting single failure event, for the analysis of all transient and accidents.

#### **PIPB Comment:**

SRXB confirmed that this staff position was new to the CE 80+ SER and not previously addressed. GDC 17 requires consideration of a LOOP in transient analysis and LOOP should be considered in all events. Lacking specific guidance in the SRP, GDC 17 prevails as an acceptance criterion and should be added to those sections where turbine trip results from the transient.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25711/CODE OF FED. REGS 10CFR50; 25712/ADVANCE SER CE CH 15

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**Integrated Impact Number:** 1369      **SRP Section Number:** 15.1.5.A

#### **Suggested Changes to the SRP Section:**

Incorporate applicable provisions of the revised source term.

SRP Section 15.3.3-15.3.4 includes the determination of the radiological consequences of a postulated rotor seizure or broken shaft event. Acceptance criteria refer to Part 100 dose limits, as does 15.1.5A. In Section 15.4.2.3 of the system 80+ FSER, the staff discusses the use of draft NUREG-1465 with regard to gap fractions and relevant isotopes (noble gases, iodines, cesiums,

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and rubidiums) and chemical species of the iodines in the fuel rod gap. SRP Section 15.1.5, Appendix A uses the same "small fraction of the 10 CFR Part 100 guidelines" criteria used in SRP Section 15.3.3-15.3.4. Therefore, it would seem appropriated to refer to NUREG-1465 for source term guidance on any radiological dose analysis required for the assessment of reactor fuel loading errors.

Consider revising acceptance criteria and references to reflect the staff's acceptance of NUREG-1465 "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995.

In accordance with the Evolutionary Issues Review performed by PNNL, the acceptance criteria addressing NUREG-1465 has been removed. The issues associated with NUREG-1465 are addressed for all of SRP Chapter 15 under Section 15.0.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25720/FINAL SER CE80 CH 15

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**Integrated Impact Number:** 1370      **SRP Section Number:** 15.2.6

#### **Suggested Changes to the SRP Section:**

Update the reference on approved analyses methods.

SRP Section 15.2.6, Revision 1, states in Acceptance Criteria that the models which have been approved by the NRC are identified in References 2 through 8. Reference 7 is the December 1975 CESSAR System 80 SER. In the FSER for the System 80+ design, Section 15.1 lists the analytical methods used for the transient analyses, and the associated NRC approval letters, in Tables 15.3 and 15.4.

Consider updating Reference 7 to reflect more recent staff-approved analyses methods.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25719/FINAL SER CE80 CH 15; 25721/FINAL SER CE80 CH 15

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### **Integrated Impacts**

**Integrated Impact Number:** 1371      **SRP Section Number:** 15.2.6

#### **Suggested Changes to the SRP Section:**

Update the reference on approved analyses methods.

SRP Section 15.2.6, Revision 1, states in Acceptance Criteria that the models which have been approved by the NRC are identified in References 2 through 8. References 2 and 3 are the March 1977 GESSAR-251 and GESSAR-238 SERs. In Section 15.1 of the ABWR FSER, the staff discusses its acceptance review of the ODYNA and REDYA computer codes used in the ABWR transient analyses.

Consider updating References 2 and 3 to reflect more recent staff-approved analyses methods.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25718/FINAL SER ABWR CH 15; 25722/FINAL SER ABWR CH 15

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**Integrated Impact Number:** 1373      **SRP Section Number:** 15.6.5.D

#### **Suggested Changes to the SRP Section:**

Revise subsections of the SRP as necessary to address BWR designs that do not have a Main Steam Isolation Valve Leakage Control System (MSIVLCS). The SRP currently identifies Regulatory Guide 1.96 as the principal source of guidance with regard to the control of MSIV leakage in BWRs following a LOCA. The Regulatory Guide indicates that "any leakage of contaminated steam through these valves is controlled by a leakage control system" and describes staff positions defining acceptable characteristics of such a system. In the EPRI FSER and ABWR DFSE, the staff has taken the position that proposals to eliminate the MSIVLCS by taking credit for fission product plate-out and holdup in main steam lines and the condenser would be acceptable, subject to complying with additional requirements. In the ABWR DFSE, the staff described positions in Sections 10.3, 10.4, and 3.2.1, regarding design approaches to provide assurance that necessary portions of the main steam system and condenser maintain their integrity following an SSE. In Section 15.4.4.2, the staff addresses several considerations affecting the ability of this alternative design approach to comply with 10 CFR 100 requirements.

Consideration should be given to revising SRP Section 15.6.5, Appendix D to indicate that a design without an MSIVLCS may be acceptable and to indicate the appropriate Sections of the SRP (to be identified following integration of the balance of the SRP Sections) where specific regulatory positions regarding this approach are located.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25697/NRC GENERIC LETTER 86-17; 25702/FINAL SER EPRI CH 1; 25703/DRAFT FINAL SER ABWR CH 15; 25704/DRAFT FINAL SER ABWR CH 10; 25705/SECY 93-087; 25706/DRAFT FINAL SER ABWR CH 3; 25707/DRAFT FINAL SER ABWR CH 14

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**Integrated Impact Number:** 1375      **SRP Section Number:** 10.4.4

#### **Suggested Changes to the SRP Section:**

Add review of the post-accident function of the turbine bypass system for those BWRs which take credit for the main steam line, turbine bypass system, and condenser for fission product holdup and plateout, in lieu of a main steam isolation valve leakage control system in recognition of the staff position in SECY-93-087, Item II.E.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25731/SECY 93-087; 25950/FINAL SER ABWR CH 10; 25951/FINAL SER EPRI CH 5

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**Integrated Impact Number:** 1377      **SRP Section Number:** 13.2.2

#### **Suggested Changes to the SRP Section:**

Federal Register Notice 59 FR 5132 identifies proposed rulemaking with regard to 10 CFR 20 and associated changes to 10 CFR 19.12. Federal Register Notice 60 FR 36038 identifies final rulemaking for 10 CFR 20 and associated changes to 10 CFR 19.12. The final rule revises the radiation protection training requirement so that it applies to workers who are likely to receive, in a year, occupational dose in excess of 100 mrem, as well as revised definitions of "member of the public," "occupational dose," and "public dose."

Consider updating the discussions in 13.2.2 related to 10 CFR 19.12 to reflect current regulation as given in 60 FR 36038.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25158/CODE OF FED. REGS 10CFR20; 25733/CODE OF FED. REGS 10CFR19

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### **Integrated Impacts**

**Integrated Impact Number:** 1378      **SRP Section Number:** 5.3.2

#### **Suggested Changes to the SRP Section:**

SRP Section 5.3.2 cites ASTM E208. The citation is non-date-specific with regard to the applicable standard version.

Standard ASTM E208-1969 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ASTM E208 to cite the 1969 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23990/C&S: ASTM E208

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**Integrated Impact Number:** 1379      **SRP Section Number:** 4.5.1

#### **Suggested Changes to the SRP Section:**

SRP Section 4.5.1 cites ASTM A262. The citation is non-date-specific with regard to the applicable standard version.

Standard ASTM A262-1970 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ASTM A262 to cite the 1970 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22206/C&S: ASTM A262

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**Integrated Impact Number:** 1380      **SRP Section Number:** 4.5.1

#### **Suggested Changes to the SRP Section:**

SRP Section 4.5.1 cites ASTM A708. The citation is non-date-specific with regard to the applicable standard version.

Standard ASTM A708-1974 has been determined to be the version applicable to the SRP citation.



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Consider updating the citation of ASTM A708 to cite the 1974 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22286/C&S: ASTM A708

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**Integrated Impact Number:** 1381      **SRP Section Number:** 13.6

#### **Suggested Changes to the SRP Section:**

SRP Section 13.6 cites ANSI N18.17. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI N18.17-1973 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI N18.17 to cite the 1973 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24034/C&S: ANSI N18.17

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**Integrated Impact Number:** 1382      **SRP Section Number:** 5.2.3

#### **Suggested Changes to the SRP Section:**

SRP Section 5.2.3 cites ASTM A262. The citation is non-date-specific with regard to the applicable standard version.

Standard ASTM A262-1970 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ASTM A262 to cite the 1970 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23448/C&S: ASTM A262

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### **Integrated Impacts**

**Integrated Impact Number:** 1383      **SRP Section Number:** 5.2.3

#### **Suggested Changes to the SRP Section:**

The staff's FSER for the CE System 80+ cites ASTM A708 as an acceptable standard. The citation is non-date-specific with regard to the applicable standard version.

The standard has been superseded or withdrawn since the SRP was last published. However, ASTM A708-1974 has been determined to be the version applicable to other SRP citations.

Consider updating the citation of ASTM A708 to cite the 1974 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23449/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 1393      **SRP Section Number:** 2.4.7

#### **Suggested Changes to the SRP Section:**

SRP Section 2.4.7 inappropriately specifies 10 CFR Part 50, 50.55a as an acceptance criterion for this section "as it requires structures, systems, and components to be designed and constructed to quality standards commensurate with the importance of the safety function to be performed." Paragraph 50.55a applies to quality standards and inservice inspection of vessels and other components subject to the ASME Boiler and Pressure Vessel Code. General Design Criterion 1 would be a more appropriate quality standards criterion for structures, systems and components important to safety.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25831/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1394      **SRP Section Number:** 2.5.4

#### **Suggested Changes to the SRP Section:**

Currently, 10 CFR Part 100 requires that the magnitude of the OBE be at least one-half the magnitude of the SSE. Industry experience has been that such an OBE controls the design of some safety systems and produces unnecessary and inconsistent margins for the SSE loading. The staff agrees that the OBE should not control the design of safety-related systems, and is involved in the rulemaking process to amend 10 CFR Part 100, Appendix A.

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EPRI, with the concurrence of all other evolutionary plant vendors, requested that the regulation be changed to reduce the magnitude of the OBE relative to the SSE. In SECY-90-016 (Item IV.A.), the staff recommended that the Commission approve the review approach to consider requests to decouple the OBE from the SSE on a design-specific basis for evolutionary designs. This was approved by the Commission in its SRM of June 26, 1990.

In SECY-93-087 (Item I.M.) the staff proposed additional guidance and positions for the design of structures systems and components (SSC) when the OBE is eliminated. The staff evaluated the effect on safety of eliminating the OBE from the design load combinations for selected SSC and has developed proposed criteria for an analysis using only the SSE. The staff recommended that the Commission approve the approach to eliminate the OBE from the design of systems, structures, and components. When the OBE is eliminated from the design, no replacement earthquake loading should be used to establish the postulated pipe rupture and leakage crack locations. In the SRM for SECY-93-087 dated July 21, 1993, the Commission approved specific staff positions and criteria for the elimination of the OBE. In the SRM for SECY-93-087 the Commission also stated that the OBE will continue to be used as a threshold criterion for conducting inspections following an earthquake event.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23015/SECY 93-087; 25226/CODE OF FED. REGS 10CFR100; 25887/FINAL SER CE80 CH 3; 25888/FINAL SER ABWR CH 3

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**Integrated Impact Number:** 1397      **SRP Section Number:** 4.4

#### **Suggested Changes to the SRP Section:**

Add a new Review Procedure incorporating staff guidance on thermal-hydraulic analyses that should be performed to develop sufficient bases for shutdown operations.

Shutdown and low-power operations have presented challenges to safe plant operation and the NRC staff has concerns about the overall safety of operations during shutdown and low-power conditions. In the staff's evaluation, shutdown and low-power operations encompasses operation when the reactor is in a subcritical state or is in transition between subcriticality and power operation up to 5% of rated power and addresses only conditions for which there is fuel in the reactor vessel. The NRC staff is considering the most appropriate regulatory approach to improve safety during shutdown and low-power operations. NUREG-1449 documents the staff's evaluation and recommendations for new regulatory guidance or requirements in five key areas: (1) Outage Planning and Control; (2) Fire Protection; (3) Operations, Training, Procedures, and Other Contingency Plans; (4) Technical Specifications; and (5) Instrumentation. The ABB-CE80+ FSER addressed shutdown and low-power risk by requiring documentation of an

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assessment of the risks and the design features utilized to minimize the risks. Generic Letter 88-17 program enhancements required additional analyses for PWRs with an emphasis placed on obtaining a complete understanding of NSSS behavior under nonpower operation.

Consideration should be given to adding a Review Procedure to address thermal-hydraulic analyses that may be required to support shutdown operations.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25783/SECY 93-087; 25784/NRC GENERIC LETTER 88-17; 25785/FINAL SER CE80 CH 19; 25786/NUREG 1449

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**Integrated Impact Number:** 1398      **SRP Section Number:** 9.5.4

#### **Suggested Changes to the SRP Section:**

SRP Section 9.5.4 cites ANSI N195. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI N195-1976 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI N195 to cite the 1976 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22698/C&S: ANSI N195

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**Integrated Impact Number:** 1399      **SRP Section Number:** 3.7.1

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures to provide information relating to the exemption accepted in the evolutionary FSERs that allowed the designers to eliminate the Operating Basis Earthquake (OBE) from seismic design considerations.

Design response spectra, time histories, and damping values are reviewed in SRP Section 3.7.1 for both OBE and Safe Shutdown Earthquake (SSE). At present, 10 CFR Part 100 Appendix A and SRP 2.5.2 require the magnitude of the OBE to be at least 1/2 the SSE. Industry experience has shown such an OBE controls the design of some safety systems. The staff agreed in SECY-93-087, Section IV.A, that the OBE should not control safety system design.

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As discussed in the ABWR and CE 80+ FSERs and in SECY 93-087, evolutionary plants may eliminate the OBE analysis from the design basis for structures, systems and components (SSCs). In such cases, the OBE will then serve as an inspection level earthquake above which the licensee would shut down the plant and inspect for potential damage to SSCs important to safety. Design certification based upon this single-earthquake design approach is predicated, in part, on the ability to ascertain the need to take action to shut down the plant in case of occurrence of the OBE, and to evaluate the performance of SSCs important to safety.

Consider changes to the Review Procedures to provide information regarding the staff's past consideration of an applicant's proposal to eliminate or reduce the OBE as an exemption to 10 CFR 100.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25780/FINAL SER ABWR CH 3; 25781/FINAL SER CE80 CH 2; 25782/SECY 93-087

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**Integrated Impact Number:** 1400      **SRP Section Number:** 9.2.4

#### **Suggested Changes to the SRP Section:**

An evaluation of radiological contamination, including accidental, and safety implications of sharing (for multi-unit facilities) should be described. This change is needed to conform to the requirements of the second sentence in subsection 9.2.4 of Regulatory Guide 1.70.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

6364/RG 1.70

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**Integrated Impact Number:** 1401      **SRP Section Number:** 3.8.4

#### **Suggested Changes to the SRP Section:**

SRP Section 3.8.4 cites ACI 318. The citation is non-date-specific with regard to the applicable standard version.

Standard ACI 318-1977 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ACI 318 to cite the 1977 version.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

23968/C&S: ACI 318

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**Integrated Impact Number:** 1402      **SRP Section Number:** 3.8.4

#### **Suggested Changes to the SRP Section:**

SRP Section 3.8.4 cites ACI 349. The citation is non-date-specific with regard to the applicable standard version.

Standard ACI 349-1976 (S79) has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ACI 349 to cite the 1976 (S79) version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23971/C&S: ACI 349

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**Integrated Impact Number:** 1404      **SRP Section Number:** 3.6.1

#### **Suggested Changes to the SRP Section:**

Currently, 10 CFR Part 100 requires that the magnitude of the OBE be at least one-half the magnitude of the SSE. Industry experience has been that such an OBE controls the design of some safety systems and produces unnecessary and inconsistent margins for the SSE loading. The requirement that the OBE be at least one-half the SSE was written into the regulation when the staff did not have much experience with the resistance of plants to ground motion from seismic events.

The staff agreed that the OBE should not control the design of safety-related systems, and is involved in a rulemaking process to amend 10 CFR Part 100, Appendix A.

EPRI, with the concurrence of all other evolutionary plant vendors, requested that the regulation be changed to reduce the magnitude of the OBE relative to the SSE. In SECY-90-016 (Item IV.A) the staff recommended that the Commission approve the review approach to consider requests to decouple the OBE from the SSE on a design-specific basis for evolutionary designs. This was approved by the Commission in its SRM of June 26, 1990.

For the System 80+ design, the magnitude of the OBE is anchored at 0.3 g, or one-third of the

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SSE. Based on Appendix A to 10 CFR Part 100, the OBE should be at least one-half of the SSE, unless a lower OBE can be justified on the basis of probability calculations. The justification given in the CE80+ DSER by the staff for accepting the OBE less than one-half of the SSE is not based on probability calculations. The staff accepted the OBE less than one-half the SSE by stating that decoupling of the OBE from the SSE for evolutionary reactors is acceptable, per SECY-90-016.

Consider revising SRP Section 3.6.1 to incorporate the intent of SECY 90-016 and SECY 93-087 (Item I.M), and to reflect the Commission approved staff position that the OBE be eliminated from the design of systems, structures, and components.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23046/SECY 93-087; 25844/DRAFT SER CE80 CH 2; 25845/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 1405      **SRP Section Number:** 9.4.2

#### **Suggested Changes to the SRP Section:**

In the EPRI evaluation of GSI 83, Control Room Habitability, the staff indicated, in addition to verifying conformance with current SRP Section 9.4.1 Acceptance Criteria and guidance, that they would also verify conformance with the following HVAC design and testing related standards: ANSI/ANS 59.2-1985, ANSI/ASME AG-1-1985, and ASTM D3803-89. The staff indicated that these verifications will demonstrate that the CRAVS design meets GDC 19 which is currently identified as Acceptance Criteria for SRP Section 9.4.1.

The requirement to verify conformance with these standards would also apply to the spent fuel pool area ventilation system (SFPavs).

Consideration should be given to revising Review Procedures for review of the adequacy of the (SFPavs) design, to include verification of conformance with the above standards.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25838/NUREG 0933; 25839/DRAFT SER CE80 CH 20; 25840/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 1407      **SRP Section Number:** 3.6.1

#### **Suggested Changes to the SRP Section:**

There is a need to review the current version of IEEE-308 standard to determine if it is

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acceptable to the staff.

Appendix B, Branch Technical Position SPLB 3-1, paragraph 11(a), IEEE-308 is cited without date of issue specified. Review of NUREG/CR-5973, Rev. 1 disclosed that the latest issue of IEEE-308 is 1991. IEEE-308, 1974 issue has been endorsed by RG 1.32 (with exceptions). Since then, various industry documents cited 1991 version of the IEEE-308, but it is not clear if the 1991 version is acceptable to the staff.

An IPD 7.0 Form (Research/Regulatory Action Needs Form) has been developed to address the need to review the IEEE-308 to determine its compliance with the current positions of the staff.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23960/C&S: IEEE 308

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**Integrated Impact Number:** 1408      **SRP Section Number:** 9.4.3

#### **Suggested Changes to the SRP Section:**

In the EPRI evaluation of GSI 83, Control Room Habitability, the staff indicated, in addition to verifying conformance with current SRP Section 9.4.1 Acceptance Criteria and guidance, that they would also verify conformance with the following HVAC design and testing related standards: ANSI/ANS 59.2-1985, ANSI/ASME AG-1-1985, and ASTM D3803-89. The staff indicated that these verifications will demonstrate that the CRAVS design meets GDC 19 which is currently identified as Acceptance Criteria for SRP Section 9.4.1.

The requirement to verify conformance with these standards would also apply to the auxiliary and radwaste area ventilation system (ARAVS).

Consideration should be given to revising Review Procedures for review of the adequacy of the (ARAVS) design, to include verification of conformance with the above standards.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25841/NUREG 0933; 25842/DRAFT SER CE80 CH 20; 25843/FINAL SER EPRI CH 1

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**Integrated Impact Number:** 1409      **SRP Section Number:** 3.6.1

#### **Suggested Changes to the SRP Section:**

There is a need to review the Standard Code for Pressure Piping, ANSI/ASME B31.1-1967,



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Pressure Piping to determine their acceptability for use in the SRP Section 3.6.1.

Appendix B to BTP SPLB 3-1 specifies that SA can be calculated according to the rules of ANSI/ASME B31.1-1967, Pressure Piping. Review of NUREG/CR-5973, Rev. 1, disclosed that the latest edition of ANSI/ASME B31.1-1967 is 1992. Review of this document is needed to determine its acceptability for use in the SRP Section 3.6.1.

An IPD 7.0 Form (Research/Regulatory Action Needs Form) has been developed to address the need to review the latest version of ANSI/ASME B31.1 to determine its compliance with the current positions of the staff.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25880/C&S: ASME III

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**Integrated Impact Number:** 1411      **SRP Section Number:** 9.4.4

#### **Suggested Changes to the SRP Section:**

In the EPRI evaluation of GSI 83, Control Room Habitability, the staff indicated, in addition to verifying conformance with current SRP Section 9.4.1 Acceptance Criteria and guidance, that they would also verify conformance with the following HVAC design and testing related standards: ANSI/ANS 59.2-1985, ANSI/ASME AG-1-1985, and ASTM D3803-89. The staff indicated that these verifications will demonstrate that the CRAVS design meets GDC 19 which is currently identified as Acceptance Criteria for SRP Section 9.4.1.

The requirement to verify conformance with these standards would also apply to the turbine area ventilation system (TAVS).

Consideration should be given to revising Review Procedures for review of the adequacy of the (TAVS) design, to include verification of conformance with the above standards.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25848/FINAL SER EPRI CH 1; 25849/DRAFT SER CE80 CH 20; 25850/NUREG 0933

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**Integrated Impact Number:** 1412      **SRP Section Number:** 5.3.1

#### **Suggested Changes to the SRP Section:**

SRP Section 5.3.1 cites ASTM E185. The citation is non-date-specific with regard to the

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applicable standard version.

Standard ASTM E185-1982 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ASTM E185 to cite the 1982 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23575/C&S: ASTM E185

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**Integrated Impact Number:** 1413      **SRP Section Number:** 5.3.1

#### **Suggested Changes to the SRP Section:**

The staff's FSER for the CE System 80+ cites ASTM A708 as an acceptable standard. The citation is non-date-specific with regard to the applicable standard version.

The standard has been superseded or withdrawn since the SRP was last published. However, ASTM A708-1974 has been determined to be the version applicable to other SRP citations.

Consider updating the citation of ASTM A708 to cite the 1974 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24181/FINAL SER CE80 CH 5

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**Integrated Impact Number:** 1415      **SRP Section Number:** 11.4

#### **Suggested Changes to the SRP Section:**

Update Appendix 11.4-A, "Design Guidance for Temporary Onsite Storage of Low Level Radioactive Waste," to SRP Section 11.4 to include the appropriate provisions of Generic Letter (GL) 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites," November 11, 1981.

The solid waste system reviewed under SRP 11.4 is a radioactive solid waste management system required by 10 CFR Part 50, Section 50.34a, with system operation in accordance with Section 50.36a. The provisions contained in GL 81-38 are much more exacting than the Appendix 11.4-A provisions. To list some of the more important provisions: GL 81-38 places a time limit (5 years) on the length of onsite storage, raises the unreviewed safety question

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concern, invokes 10 CFR Part 50.59 and Part 30 licensing approval, and states that the storage facilities and equipment must be reviewed under SRP Section 11 criteria as opposed to named regulatory guides.

Consider updating Appendix 11.4-A to SRP 11.4 to incorporate the more stringent provisions of Generic Letter 81-38.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25852/ADVANCE SER ABWR CH 11; 25853/ADVANCE SER CE CH 11

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**Integrated Impact Number:** 1416      **SRP Section Number:** 9.5.1

#### **Suggested Changes to the SRP Section:**

Revise Branch Technical Position CMEB 9.5-1 to include the guidance in Generic Letter 82-21.

The NRC issued Generic Letter 82-21 in response to inquiries regarding fire protection program audits required by Technical Specifications. The Generic Letter was provided to licensees as information and guidance, and did not establish any new staff positions. The Generic Letter describes the differences in the required audits, including audit scope and auditor independence and affiliation relative to the fire protection program and licensee organization. The Generic Letter also provides additional guidance and information with regard to fire protection quality assurance program content.

Consider revising BTP CMEB 9.5-1 to include the fire protection quality assurance program and audit guidance from Generic Letter 82-21.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25663/NRC GENERIC LETTER 82-21

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**Integrated Impact Number:** 1417      **SRP Section Number:** 3.7.4

#### **Suggested Changes to the SRP Section:**

Revise the Review Procedures to incorporate additional information regarding an exemption to eliminate the Operating Basis Earthquake (OBE) from seismic design considerations as discussed in the ABWR and System 80+ design certification FSERs. The Staff approved exemption was based in part on commitments to develop additional seismic related procedural requirements such as pre-earthquake planning and post-earthquake operator actions.

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SRP Section 3.7.4 provides for the review of procedures that are to be followed after a seismic event to inform the control room operator of the earthquake's peak acceleration level and criteria that should be used to compare the measured responses of Category I structures with the results of seismic analyses.

As discussed in the ABWR and CE 80+ FSERs, and in SECY 93-087, evolutionary plants may eliminate the OBE as a design basis for structures, systems and components. In such cases, the OBE will serve as an inspection level earthquake above which the licensee would be required to shut down the plant and inspect for damage. Design certification based upon this single-earthquake design approach is predicated on the adequacy of pre- earthquake planning and post-earthquake damage inspections that are to be implemented by the combined operating licensee. For the evolutionary reactor design certifications the NRC staff reviewed criteria developed by the Electric Power Research Institute (EPRI) in EPRI NP-5930, EPRI NP-6695, and EPRI TR-100082 for evaluating the need to shut down the plant following an earthquake and the commitments in the ABWR and CE 80+ FSERs for ensuring that such actions can be taken. The Staff concluded that EPRI NP-5930 was adequate for use by a combined operating licensee; however, the Staff documented several exceptions regarding the installation and use of seismic instruments.

The Staff is currently developing draft Regulatory Guides for changes to seismic instrumentation requirements (DG-1033), pre-earthquake planning and post-earthquake operator actions (DG-1034), and restart of a nuclear power plant shutdown as the result of a seismic event (DG-1035). These draft Regulatory Guides are based on requirements in the proposed 10 CFR 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants." (Note: ROC 950 and 951 serve as placeholders for changes that may be proposed as a result of these future regulatory guides.)

Consider revising the Review Procedures to provide information regarding commitments made in the evolutionary FSERs upon which an exemption was approved to eliminate the OBE from seismic design considerations.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25667/SECY 93-087; 25668/FINAL SER ABWR CH 3; 25669/FINAL SER CE80 CH 3;  
25670/CODE OF FED. REGS 10CFR100

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**Integrated Impact Number:** 1418      **SRP Section Number:** 4.6

#### **Suggested Changes to the SRP Section:**

Add GDC 4 as acceptance criteria to SRP Section 4.6 and develop corresponding evaluation

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findings to support the conclusion that the control rod drive system meets the requirements of GDC 4. Also, modify the review interface to include other reviews that demonstrate compliance of the control rod drive system with relevant GDC 4 requirements.

During the SRP-UDP draft revision of SRP Section 4.5.1 it was discovered that the SRP does not appear to directly apply GDC 4 as acceptance criteria for the control rod drive system. SRP Sections 3.9.4, 4.5.1, and 4.6 are the principal sections under which control rod drives are explicitly reviewed. Control rod drives are important to safety and therefore they should meet the applicable requirements of GDC 4. SRP Section 3.9.4 evaluates control rod drive loadings including those influenced by environmental factors and dynamic loadings and references SRP Section 3.9.3 for applicable stress limits. To address consideration of cyclic and fatigue effects related to cumulative environmental stresses in the design of components, SRP Section 3.9.3 applies GDC 4 as acceptance criteria primarily for ASME Code Class components associated with pressure retention. This does not appear to cover non-pressure retaining portions of control rod drives (e.g. housings and other structural components) nor aspects of the control rod drive system's compliance with GDC 4 requirements.

SRP Section 4.6 provides the only plant/system layout based review of control rod drive systems. Numerous aspects of compliance with the requirements of GDC 4 (e.g., missile protection, protection against dynamic effects, protection against flooding due to equipment failure, qualification for accident environments, etc.) require evaluation of the system arrangement and the plant/system layout to determine compliance. Subsection III.2 of SRP Section 4.6 already describes a review verifying that the control rod drive systems are protected against the effects of postulated pipe breaks which is clearly a review which has its regulatory basis supported by GDC 4.

Consideration should be given to include GDC 4 as acceptance criteria to support review procedures currently presented in this section and corresponding evaluation findings should also be developed. Furthermore, review interfaces with other sections should be added to support the conclusion that the control rod drive system complies with GDC 4 requirements.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20820/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1419      **SRP Section Number:** 12.2

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria, Review Procedures, and Evaluation Findings to replace citations of superseded sections in 10 CFR Part 20.

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The Acceptance Criteria subsection of SRP 12.2 cites 10 CFR Part 20, 20.101 as it relates to limiting radiation doses to protect individuals in restricted areas from whole or partial body exposures, 20.103 as it relates to limiting average concentrations of airborne radioactive materials to protect individuals in restricted areas, and control of inhalation or absorption of such materials, 20.104 as it relates to limiting exposure to minors to one-tenth of the limits for adults, 20.106 as it relates to determining radiation levels and radioactive materials concentrations within the components of waste treatment systems, and 20.207 as it relates to securing licensed materials against unauthorized removal.

On May 21, 1991, the NRC issued a revision to the standards for protection against ionizing radiation. The revised Part 20 (Sections 20.1001-20.2401) became effective January 1, 1994, and the requirements specified in paragraph 20.101 have been replaced by paragraph 20.1201, 20.1202, and 20.1206, the requirements specified in paragraph 20.103 have been replaced by paragraphs 20.1203 and 20.1204, the requirements specified in paragraph 20.104 have been replaced by paragraph 20.1207, the requirements specified in paragraph 20.106 have been replaced by paragraph 20.1301, and the requirements specified in paragraph 20.207 have been replaced by paragraph 20.1801.

Consider revising the Acceptance Criteria, Review Procedures, and Evaluation Findings to replace citations to the superseded sections of 10 CFR Part 20 with citations to sections that reflect the current version of 10 CFR Part 20.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23741/CODE OF FED. REGS 10CFR20

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**Integrated Impact Number:** 1421      **SRP Section Number:** 11.3

#### **Suggested Changes to the SRP Section:**

Update Branch Technical Position ETSB 11-5 to comply with the current revision of the GALE code, revised staff practices, SI unit nomenclature, and RG 1.170 changes.

The waste gas system reviewed under SRP 11.3 is a radioactive gaseous waste management system required by 10 CFR Part 50, Section 50.34a, with system operation in accordance with Section 50.36a. Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," provides guidance based on the PWR GALE code listed in NUREG-0017, Revision 0, which is now outdated and inconsistent with current staff practices provided in Revision 1. Also, the current practice is to provide reasonable assurance that the radiological consequences of a single failure of an active component in the waste gas system would not result in exceeding a small fraction (i.e., 10%) of 10 CFR Part 100 limit for

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wholebody dose to any offsite individual for a postulated event.

Consideration should be given to revising BTP ETSB 11-5 to reflect current staff practices in the evaluation of postulated radioactive releases due to a waste gas system leak or failure.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25855/ADVANCE SER CE CH 11; 25856/ADVANCE SER ABWR CH 11

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**Integrated Impact Number:** 1423      **SRP Section Number:** 2.4.2

#### **Suggested Changes to the SRP Section:**

SRP Section 2.4.2 cites ANSI N170. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI N170-1976 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI N170 to cite the 1976 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22835/C&S: ANSI N170

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**Integrated Impact Number:** 1424      **SRP Section Number:** 2.4.3

#### **Suggested Changes to the SRP Section:**

SRP Section 2.4.3 cites ANSI N170. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI N170-1976 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI N170 to cite the 1976 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22837/C&S: ANSI N170

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**Integrated Impact Number:** 1425      **SRP Section Number:** 3.13

#### **Suggested Changes to the SRP Section:**

One of the recommendations in NUREG-1339, "Resolution of Generic Safety Issue 29: 'Bolting Degradation of Failure in Nuclear Power Plants,'" was that a new Standard Review Plan section on bolting be prepared. The staff endorsed the resolution of GSI 29, as described in NUREG-1339, in Generic Letter 91-17 of October 17, 1991.

The staff should consider preparing a new Standard Review Plan section to cover its review of threaded fastener applications in nuclear power plants.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25859/NRC GENERIC LETTER 91-17; 25860/NRC GENERIC LETTER 91-17; 25861/NRC BULLETIN 82-02; 25862/NUREG 1339; 25863/NRC GENERIC LETTER 91-17; 25864/NUREG 1339; 25865/NRC GENERIC LETTER 91-17; 25866/NUREG 1339

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**Integrated Impact Number:** 1426      **SRP Section Number:** 3.13

#### **Suggested Changes to the SRP Section:**

Consider citing 10 CFR 50.55a in new SRP Section 3.13 to apply the specifications and standards described in Regulatory Guide 1.26 to threaded fasteners.

10 CFR 50.55a, "Codes and Standards," requires (in part) that:

- a. Structures, systems, and components (SSCs) must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed;
- b. Components of the reactor coolant pressure boundary (RCPB) must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) Code;
- c. Quality Group B components covered in Regulatory Guide 1.26 must meet the requirements for Class 2 components in Section III of the ASME Code;
- d. Quality Group C components covered in Regulatory Guide 1.26 must meet the requirements for Class 3 components in Section III of the ASME Code; and



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e. ASME Code Class 1, 2, and 3 components must be designed and be provided with access to facilitate the performance of inservice testing.

Threaded fasteners must meet these requirements, in accordance with the particular application.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25867/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1427      **SRP Section Number:** 3.13

#### **Suggested Changes to the SRP Section:**

Consider citing 10 CFR 50.60 and Appendix G to 10 CFR 50 in new SRP Section 3.13 to apply the requirements described therein to threaded fasteners.

10 CFR 50.60, "Acceptance Criteria For Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," references Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," for fasteners in pressure-retaining components of the reactor coolant pressure boundary with yield strengths not over 896 MPa (130,000 psi). Appendix G describes tests and other requirements applicable to these items.

Threaded fasteners must meet these requirements, in accordance with the particular application.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25868/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1428      **SRP Section Number:** 3.13

#### **Suggested Changes to the SRP Section:**

Consider citing General Design Criterion (GDC) 1 (Appendix A of 10 CFR 50) and Appendix B of 10 CFR 50 in new SRP Section 3.13 to apply the provisions described therein to threaded fasteners.

General Design Criterion 1 (GDC 1), "Quality Standards and Records," requires (in part) that:

a. SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed;

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- b. A quality assurance (QA) program shall be established and implemented to provide adequate assurance that these SSCs will satisfactorily perform their safety functions; and
- c. Appropriate records for the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by the licensee.

Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Facilities," includes requirements applicable to threaded fasteners (reviewed under SRP Chapter 17) in the following subsections:

- a. Part VIII, "Identification and Control of Materials, Parts, and Components";
- b. Part X, "Inspection";
- c. Part XI, "Test Control"; and
- d. Part XV, "Nonconforming Materials, Parts, or Components."

These provisions are directly applicable to components containing threaded fasteners and to the fasteners themselves.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25869/CODE OF FED. REGS 10CFR50

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<b>Integrated Impact Number:</b> 1429	<b>SRP Section Number:</b> 3.13
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#### **Suggested Changes to the SRP Section:**

Consider citing GDC 2 and GDC 4 in new SRP Section 3.13 for the necessary design bases.

General Design Criterion 2 (GDC 2), "Design Bases for Protection Against Natural Phenomena," requires (in part) that SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquake, tornado, hurricane, tsunami, and seiche without loss of capability to perform their safety functions.

General Design Criterion 4 (GDC 4), "Environmental and Dynamic Effects Design Bases," requires (in part) that SSCs important to safety be designed to accommodate the effects of, and be compatible with, environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs).

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Threaded fasteners, as parts of particular SSCs, must meet these requirements.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25870/CODE OF FED. REGS 10CFR50; 25871/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1430      **SRP Section Number:** 3.13

#### **Suggested Changes to the SRP Section:**

Consider citing GDC 14, GDC 15, GDC 30, GDC 31, and GDC 32 in new SRP Section 3.13 to apply the provisions described therein to threaded fasteners.

General Design Criterion 14 (GDC 14), "Reactor Coolant Pressure Boundary," requires (in part) that the RCPB be designed, fabricated, erected, and tested in a manner that provides assurance of an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture.

General Design Criterion 15 (GDC 15), "Reactor Coolant System Design," requires (in part) that the reactor coolant system (RCS) and associated auxiliary, control, and protection systems be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences.

General Design Criterion 30 (GDC 30), "Quality of Reactor Coolant Pressure Boundary," requires (in part) that:

- a. Components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical; and
- b. Means be provided for detecting and, to the extent practical, identifying the location of reactor coolant leakage.

General Design Criterion 31 (GDC 31), "Fracture Prevention of Reactor Coolant Pressure Boundary," requires (in part) that:

- a. The RCPB be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized; and
- b. The design reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions, including the

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uncertainties associated with determining:

- (1) Material properties;
- (2) The effects of irradiation on material properties;
- (3) Residual, steady state, and transient stresses; and
- (4) Size of flaws.

General Design Criterion 32 (GDC 32), "Inspection of Reactor Coolant Pressure Boundary," requires (in part) that components that are part of the RCPB be designed to permit:

- a. Periodic inspection and testing of important areas and features to assess their structural leaktight integrity, and
- b. An appropriate material surveillance program for the reactor pressure vessel.

Threaded fasteners must meet these requirements for applications in the reactor coolant system or within the reactor coolant pressure boundary.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25872/CODE OF FED. REGS 10CFR50; 25873/CODE OF FED. REGS 10CFR50;  
25874/CODE OF FED. REGS 10CFR50; 25875/CODE OF FED. REGS 10CFR50;  
25876/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1431      **SRP Section Number:** 9.3.5

#### **Suggested Changes to the SRP Section:**

Revise Review Procedures for review of Standby Liquid Control System (SLCS) inservice testability features to incorporate current staff positions for new applications.

SRP Section 9.3.5 includes Review Procedures verifying that design provisions have been made that permit appropriate inservice inspection and functional testing of the system, that the SAR information delineates a testing and inspection program (reviewed in detail in other SRP sections), and that the system drawings show the connections and special piping and equipment required by this program.

In SECY 90-016 and SECY 93-087, the staff stated new positions related to inservice testing

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programs and testing of all safety-related pumps and valves which were approved as supplemented by the Commission. Key testability design features covered in the new positions include provisions to test motor-operated valves under design-basis differential pressure and piping design provisions for full flow testing (maximum design flow) of pumps and check valves. These issues were clarified in SECY 93-087 such that design features should support full flow testing at maximum design flow with analysis to extrapolate to design pressure if it is not practicable to conduct inservice pump testing at design flow and pressure and inservice valve tests under the maximum practicable differential pressure and flow when it is not practicable to achieve design-basis differential pressure during an inservice test.

Consideration should be given to revising Review Procedures for review of SLCS testability features to incorporate evaluation of provisions for system testability against the above discussed positions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23009/SECY 90-016

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**Integrated Impact Number:** 1433      **SRP Section Number:** 6.2.4

#### **Suggested Changes to the SRP Section:**

SRP Section 6.2.4 (Rev. 2 - July 1981) states in paragraph II.h that Regulatory Guide 1.141 "will contain" guidance on the classification of piping systems as essential or non-essential for purposes of containment isolation.

NUREG-0737 Item II.E.4.2, Clarification (3), page II.E.4.2-2, states that Revision 2 to Regulatory Guide 1.141 "is due to be issued by June 1981." This revision has not been issued. The current version of Regulatory Guide 1.141 is the original issue of April 1978.

10 CFR 50.34(f)(2)(xiv) requires that all non-essential systems be isolated automatically by the containment isolation system, and that each non-essential penetration (except instrument lines) have two isolation barriers in series.

Consider revising SRP Section 6.2.4 to indicate the current source of information on essential and non-essential systems (NUREG-0737, Item II.E.4.2). Consider revising Regulatory Guide 1.141 to include regulatory guidance on the classification of systems as essential and non-essential for purposes of containment isolation as shown on IPD 7.0 form 6.2.4-2.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

4095/RG 1.141; 4096/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1434      **SRP Section Number:** 6.2.6

#### **Suggested Changes to the SRP Section:**

Revise SRP Section 6.2.6 to address revision of time interval for Type C leakage rate testing approved by the Commission in SRM dated June 21, 1993.

Appendix J to 10 CFR Part 50 specifies that Type C leakage rate tests on containment isolation valves be performed during each reactor shutdown for refueling, but in no case at intervals greater than 2 years. The EPRI Evolutionary Requirements Document addresses increasing the time interval between Type C tests to 30 months by stating that the time interval between leakage rate tests will be evaluated during its review of an individual application for a standard design certification.

In SECY 93-087, Issue II.H, the staff recommended that the interval for Type C leakage tests be changed to 30 months from 24 months. The Commission endorsed this recommendation in the SRM dated July 21, 1993.

Revise SRP Section 6.2.6 to reflect the Commission approved revision of the time interval for Type C leakage rate testing.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25657/SECY 93-087; 25658/FINAL SER EPRI CH 5

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**Integrated Impact Number:** 1435      **SRP Section Number:** 9.1.4

#### **Suggested Changes to the SRP Section:**

SRP Section 9.1.4 cites ANS 57.1/ANSI N208. The citation is non-date-specific with regard to the applicable standard version.

Standard ANS 57.1/ANSI N208-1980 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANS 57.1/ANSI N208 to cite the 1980 version.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24010/C&S: ANS 57.1

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**Integrated Impact Number:** 1436      **SRP Section Number:** 9.1.5

#### **Suggested Changes to the SRP Section:**

SRP Section 9.1.5 cites ANS 57.1/ANSI N208. The citation is non-date-specific with regard to the applicable standard version.

Standard ANS 57.1/ANSI N208-1980 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANS 57.1/ANSI N208 to cite the 1980 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24012/C&S: ANS 57.1

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**Integrated Impact Number:** 1437      **SRP Section Number:** 9.1.5

#### **Suggested Changes to the SRP Section:**

SRP Section 9.1.5 cites ANS 57.2/ANSI N210. The citation is non-date-specific with regard to the applicable standard version.

Standard ANS 57.2/ANSI N210-1976 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANS 57.2/ANSI N210 to cite the 1976 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24011/C&S: ANS 57.2

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**Integrated Impact Number:** 1439      **SRP Section Number:** 6.2.1

#### **Suggested Changes to the SRP Section:**

Add a new introductory paragraph incorporating staff guidance on containment analyses that

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should be performed to develop sufficient bases for shutdown operations.

Shutdown and low-power operations have presented challenges to safe plant operation and the NRC staff has concerns about the overall safety of operations during shutdown and low-power conditions. In the staff's evaluation, shutdown and low-power operations encompasses operation when the reactor is in a subcritical state or is in transition between subcriticality and power operation up to 5% of rated power and addresses only conditions for which there is fuel in the reactor vessel. The NRC staff is considering the most appropriate regulatory approach to improve safety during shutdown and low-power operations. NUREG 1449 documents the staff's evaluation and recommendations for new regulatory guidance or requirements in five key areas: (1) Outage Planning and Control; (2) Fire Protection; (3) Operations, Training, Procedures, and Other Contingency Plans; (4) Technical Specifications; and (5) Instrumentation. The ABWR FSER and the CE80+ FSER addressed shutdown and low-power risk by requiring documentation of an assessment of the risks and the design features utilized to minimize the risks. Generic Letter 88-17 required PWRs to conduct analyses to supplement existing information and develop sufficient bases for shutdown operation.

Consideration should be given to adding an introductory paragraph to address containment analyses that may be required to support shutdown operations.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25754/SECY 93-087; 25755/FINAL SER ABWR CH 19; 25756/NRC GENERIC LETTER 88-17; 25757/NUREG 1449; 25889/FINAL SER CE80 CH 20

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**Integrated Impact Number:** 1440      **SRP Section Number:** 2.5.2

#### **Suggested Changes to the SRP Section:**

SRP Section 2.5.2 currently reviews vibratory ground motion for design basis earthquakes.

10 CFR 52 provides requirements for site parameter envelopes that are to be included in applications for design certifications and manufacturing licenses for standard plant designs. Applications which reference standard plant designs approved under 10 CFR 52 are required to address the conformance of site-specific parameters with the site parameter envelope for the approved, certified standard design. In SERs documenting staff review of evolutionary plant applications for design certification, the staff addressed the requirements related to the site parameter envelope in Section 2.6.

Consideration should be given to revising existing SRP Sections for review of site-specific parameters to reflect the site parameter-related requirements of 10 CFR 52, for applications



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referencing a standard plant design.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25736/FINAL SER CE80 CH 2; 25738/FINAL SER ABWR CH 2; 25883/CODE OF FED. REGS 10CFR52; 25884/CODE OF FED. REGS 10CFR52

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**Integrated Impact Number:** 1441      **SRP Section Number:** 13.5.1.1

#### **Suggested Changes to the SRP Section:**

SRP Section 13.5.1 cites ANS 3.2. The citation is non-date-specific with regard to the applicable standard version.

Standard ANS 3.2-1976 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANS 3.2 to cite the 1976 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24031/C&S: ANS 3.2

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**Integrated Impact Number:** 1443      **SRP Section Number:** 15.2.7

#### **Suggested Changes to the SRP Section:**

Update the references for approved models for accident analyses methods.

SRP Section 15.2.7, Revision 1 Acceptance Criteria, states that the models which have been approved by the NRC are identified in References 2 through 8. References 2 and 3 are the March 1977 GESSAR-251 and GESSAR-238 SERs. In Section 15.1 of the ABWR FSER, the staff discusses it's acceptance review of the ODYNA and REDYA computer codes used in the ABWR transient analyses.

Consider updating References 2 and 3 to reflect more recent staff approved analyses models.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25892/FINAL SER ABWR CH 15; 25893/FINAL SER ABWR CH 15

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**Integrated Impact Number:** 1444      **SRP Section Number:** 15.2.7

#### **Suggested Changes to the SRP Section:**

Update the references for approved models for accident analyses methods.

SRP Section 15.2.7, Revision 1 Acceptance Criteria, states that the models which have been approved by the NRC are identified in References 2 through 8. Reference 7 is the December 1974 CESSAR System 80 SER. In the FSER for the system 80+ design, Section 15.1 lists the analytical methods used for the transient analysis, and the associated NRC approval letters, in tables 15.3 and 15.4.

Consider updating Reference 7 to reflect more recent staff approved analyses models

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25890/FINAL SER ABWR CH 15; 25891/FINAL SER CE80 CH 15

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**Integrated Impact Number:** 1445      **SRP Section Number:** 6.2.5

#### **Suggested Changes to the SRP Section:**

Provide Acceptance Criteria and Review Procedures, applicable to evolutionary plants, for the purpose of establishing current guidance with regard to addressing combustible gas control systems in the containment.

SRP 6.2.5 currently cites 10 CFR 50.44, discusses hydrogen recombiners and containment penetrations for them, discusses BWR Mark III containments, and discusses reactors that must rely on purge/repressurization as a primary means for hydrogen control in the containment. These examples were all necessary adjustments for established older plants (prior to early 1980s) for controlling combustible gasses within the containment such as excessive hydrogen generated from a zirconium/water reaction following a LOCA. 10 CFR 50.44 contains dated requirements that have since been completed.

Currently, Regulatory Guide 1.7 requires that containment purge/repressurization not be relied upon for combustible gas control in the containment.

In SECY-90-016, the staff recommended to the Commission that the hydrogen control requirements for evolutionary plants be identical to those stated in 10 CFR 50.34(f). The Commission approved the staff's recommendation in a staff requirements memorandum dated June 26, 1990.

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Paragraphs (1)(xii), (2)(ix), and (3)(v) of section 50.34 (f), pertaining to hydrogen control measures, must be met by all the applicants listed in 50.34(f). 10 CFR 50.34(f)(2)(ix) requires a hydrogen control system based on a 100-percent fuel-cladding metal-water reaction and a hydrogen concentration limit of 10 percent on uniformly distributed hydrogen in the containment or a post accident atmosphere that will not support hydrogen combustion.

Consider providing Acceptance Criteria and Review Procedures applicable to evolutionary plants establishing a design bases from 10 CFR 50.34(f) with regard to control of combustible gases in containment and deleting outdated information that was previously necessary as a fix for existing plants.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

4284/NUREG 0933; 22676/DRAFT SER CE80 CH 20; 22678/FINAL SER EPRI CH 5; 22679/DRAFT FINAL SER ABWR CH 6; 22910/SECY 90-016; 22911/CODE OF FED. REGS 10CFR52; 23825/CODE OF FED. REGS 10CFR50; 23826/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1446      **SRP Section Number:** 6.1.1

#### **Suggested Changes to the SRP Section:**

SRP Section 6.1.1 cites ASTM A262. The citation is non-date-specific with regard to the applicable standard version.

Standard ASTM A262-1970 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ASTM A262 to cite the 1970 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22249/C&S: ASTM A262

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**Integrated Impact Number:** 1447      **SRP Section Number:** 6.1.1

#### **Suggested Changes to the SRP Section:**

SRP Section 6.1.1 cites AWS D1.1. The citation is non-date-specific with regard to the applicable standard version.

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Standard AWS D1.1-1981 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of AWS D1.1 to cite the 1981 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22250/C&S: AWS D1.1

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**Integrated Impact Number:** 1451      **SRP Section Number:** 8.2

#### **Suggested Changes to the SRP Section:**

SRP Section 8.2 cites ANSI C37.06. The citation is non-date-specific with regard to the applicable standard version.

ANSI C37.06 provides a circuit breaker rating method for establishing the rating of high-voltage ac circuit breakers. Guidance for establishing a full set of ratings including maximum voltage; rated voltage, current, and frequency; short circuit current is provided.

Standard ANSI C37.06-1979 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI C37.06 to cite the 1979 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23661/C&S: ANSI C37.06

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**Integrated Impact Number:** 1453      **SRP Section Number:** 8.2

#### **Suggested Changes to the SRP Section:**

Add an Appendix (B) to SRP Section 8.2 that provides guidance for reviews of the adequacy of alternate ac (AAC) power sources with respect to the requirements of the Station Blackout Rule, 10 CFR 50.63.

The Nuclear Regulatory Commission has issued 10 CFR 50.63, Loss of All Alternating Current Power, and a supporting Regulatory Guide (RG), 1.155, Station Blackout. RG 1.155 describes a means acceptable to the staff for meeting the requirements of 10 CFR 50.63 including guidance for AAC power sources. RG 1.155 references NUMARC-8700, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors. The latter

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document also provides acceptable guidance for meeting the requirements of the Station Blackout Rule.

Consider adding Appendix B to SRP Section 8.2 covering review of AAC sources.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25676/CODE OF FED. REGS 10CFR50; 25677/RG 1.155; 25678/NUMARC 8700;  
26018/FINAL SER CE80 CH 8; 26019/FINAL SER ABWR CH 8

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**Integrated Impact Number:** 1454      **SRP Section Number:** 2.1.1

#### **Suggested Changes to the SRP Section:**

SRP section 2.1.1, Site Location and Description, focuses on requirements of 10 CFR 100, particularly on the boundaries of the exclusion area and on transportation corridors through the exclusion area. Regulatory Guide 1.70, Section 2.1.1.3 requires the applicant to provide information on the restricted area, including boundaries, radiological effluent release points, and measures which the applicant will take to control access to the restricted area. The restricted area is a requirement of 10 CFR 20. The material identified in Regulatory Guide 1.70 must be addressed under the corresponding SRP section.

In preparing the draft update to the SRP section, LITCO inserted guidance for review of the restricted area description and included compliance with 10 CFR 20 as the acceptance criterion. Major revisions to part 20 were made in 1991. The current version of Part 20 was utilized in doing this. However, consideration of Part 20 was limited to Subpart D. Consideration of other portions of the revised regulation may show that the updated SRP section is adequate.

The regulatory guide is based on outdated Part 20 requirements. Most notably, the 1991 revisions reorganized and renumbered sections of Part 20 even where no substantive changes were made. An IPD 7.0 form is attached recommending consideration of revisions to Regulatory Guide 1.70 based on the 1991 revisions to 10 CFR 20.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25899/CODE OF FED. REGS 10CFR20

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**Integrated Impact Number:** 1455      **SRP Section Number:** 3.4.2

**Suggested Changes to the SRP Section:**

The acceptance criteria rely on the third edition of U.S. Army Coastal Engineering Research Center "Shore Protection Manual," 1977. Another edition of this manual was published in 1984. The staff should evaluate the latest edition of the manual and update SRP Section 3.4.2 as warranted.

**Potential Impacts/Documents supporting the Suggested Changes:**

25900/C&S: US ARMY SPM-1977

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**Integrated Impact Number:** 1456      **SRP Section Number:** 3.6.1

**Suggested Changes to the SRP Section:**

There is a need to review the ASME Boiler and Pressure Vessel Code Section III to determine their acceptability for use in the SRP Section 3.6.1.

Appendix B to BTP SPLB 3-1 specifies that S[SUB]A[sub] can be calculated according to the rules of NC-3600 of the ASME Code Section III, Winter 1972 Addenda. Review of NUREG/CR-5973, Rev. 1, disclosed that the latest edition of the ASME Code is 1992. Review of this document is needed to determine its acceptability for use in the SRP Section 3.6.1.

An IPD 7.0 Form (Research/Regulatory Action Needs Form) has been developed to address the need to review the latest version of the ASME Boiler and Pressure Vessel Code Section III to determine its compliance with the current positions of the staff.

**Potential Impacts/Documents supporting the Suggested Changes:**

25901/C&S: ASME III

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**Integrated Impact Number:** 1457      **SRP Section Number:** 13.3

**Suggested Changes to the SRP Section:**

Add a discussion of special emergency planning and preparedness requirements applicable for issuance of a license for fuel loading and low power testing.

10 CFR 50.47(d) sets forth specific provisions which would allow a license to be issued for fuel

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loading and low power testing without all of the requirements of 10 CFR 50.47(a) and (b) having been met.

Consider modifying Areas of Review, Acceptance Criteria and Review Procedures to address the 10 CFR 50.47(d) provisions.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25902/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1458      **SRP Section Number:** 13.3

#### **Suggested Changes to the SRP Section:**

Add a discussion of 10 CFR Part 52 requirements for applicants for Combined Licenses and Standard Design Certifications.

10 CFR 52.79, et seq., contains specific provisions regarding emergency planning requirements for a combined license applicant. 10 CFR 52, Appendix O, contains specific provisions regarding emergency planning requirements for a standard design certification applicant. These Part 52 provisions complement the contents of Part 50 for these special applicants.

Consider modifying Areas of Review, Acceptance Criteria, and Review Procedures to include reference to the guidance provided in Title 10 CFR Part 52.

(NOTE: This ROC has been changed. The reference to early site permits has been removed since the SRP does not address the review of early site applications.)

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25903/CODE OF FED. REGS 10CFR52

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**Integrated Impact Number:** 1459      **SRP Section Number:** 13.3

#### **Suggested Changes to the SRP Section:**

Add NUREG 0396 to provide guidance for developing emergency planning zones and a discussion of the MOU between the NRC and FEMA defining the principles by which each organization will operate with respect to the assessment of emergency planning.

A footnote in 10 CFR 50.33 states that emergency planning zones (EPZs) are discussed in

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NUREG-0396 (EPA 520/1-78-016).

10 CFR 50.47 specifies the involvement of FEMA in the review of State and local plans. Several substantive amendments have been made to regulations pertaining to FEMA review of State and local emergency plans subsequent to issuance of SRP Section 13.3, Revision 2. Amendments have been made to the NRC/FEMA Memorandum of Understanding and to the FEMA regulations for review of State and local plans. Specific inclusion of 50.47 as an acceptance criterion will provide a clearer basis for establishing compliance with the ensuing regulations.

Consider modifying Acceptance Criteria and Review Procedures to include NUREG 0396 and a discussion relating to the MOU established between the NRC and FEMA.

(NOTE: This ROC has been revised.)

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25904/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1461      **SRP Section Number:** 3.8.3

#### **Suggested Changes to the SRP Section:**

SRP Section 3.8.3 cites ACI 318. The citation is non-date-specific with regard to the applicable standard version.

Standard ACI 318-1977 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ACI 318 to cite the 1977 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23966/C&S: ACI 318

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**Integrated Impact Number:** 1462      **SRP Section Number:** 3.8.3

#### **Suggested Changes to the SRP Section:**

SRP Section 3.8.3 cites ANSI N45.2.5. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI N45.2.5-1974 has been determined to be the version applicable to the SRP



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citation.

Consider updating the citation of ANSI N45.2.5 to cite the 1974 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23967/C&S: ANSI N45.2.5

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**Integrated Impact Number:** 1464      **SRP Section Number:** 3.8.5

#### **Suggested Changes to the SRP Section:**

SRP Section 3.8.5 cites ACI 349. The citation is non-date-specific with regard to the applicable standard version.

Standard ACI 349-1976 (S79) has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ACI 349 to cite the 1976 (S79) version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23975/C&S: ACI 349

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**Integrated Impact Number:** 1465      **SRP Section Number:** 3.8.5

#### **Suggested Changes to the SRP Section:**

SRP Section 3.8.5 cites AISC. The citation is non-date-specific with regard to the applicable standard version.

Standard AISC N690-1969 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of AISC to cite the 1969 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23974/C&S: AISC Specifications

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**Integrated Impact Number:** 1466      **SRP Section Number:** 2.4.4

**Suggested Changes to the SRP Section:**

SRP Section 2.4.4 cites ANSI N170. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI N170-1976 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI N170 to cite the 1976 version.

**Potential Impacts/Documents supporting the Suggested Changes:**

22853/C&S: ANSI N170

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**Integrated Impact Number:** 1467      **SRP Section Number:** 2.4.8

**Suggested Changes to the SRP Section:**

SRP Section 2.4.8 cites ANSI N170. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI N170-1976 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI N170 to cite the 1976 version.

**Potential Impacts/Documents supporting the Suggested Changes:**

23959/C&S: ANSI N170

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**Integrated Impact Number:** 1468      **SRP Section Number:** 2.4.10

**Suggested Changes to the SRP Section:**

SRP Section 2.4.10 cites ANSI N170. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI N170-1976 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI N170 to cite the 1976 version.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

22817/C&S: ANSI N170

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**Integrated Impact Number:** 1469      **SRP Section Number:** 3.3.1

#### **Suggested Changes to the SRP Section:**

SRP Section 3.3.1 cites ANSI A58.1. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI A58.1-1972 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI A58.1 to cite the 1972 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22648/C&S: ANSI A58.1

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**Integrated Impact Number:** 1470      **SRP Section Number:** 3.3.2

#### **Suggested Changes to the SRP Section:**

SRP Section 3.3.2 cites ANSI A58.1. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI A58.1-1972 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI A58.1 to cite the 1972 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22649/C&S: ANSI A58.1

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**Integrated Impact Number:** 1472      **SRP Section Number:** 15.0

#### **Suggested Changes to the SRP Section:**

Incorporate a new staff position requiring accident and transient analyses to consider a loss of

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offsite power (LOOP) in addition to any limiting single failure (i.e., LOOP is not to be considered as a single failure).

In the CE80+ FSER, the staff stated that, in accordance with the requirements of GDC 17, a LOOP should not be considered as a single-failure event and should be assumed in the analysis for each event without changing the event category. ABB-CE was required to discuss each of the transient and accident analyses in the CESSAR-DC to justify that the analyses conform to the GDC 17 requirements given above. If the existing analyses did not conform to the GDC 17 requirements, ABB-CE should have reanalyzed the transient and accident analyses in accordance with GDC 17 and should submit the analyses for the staff to review.

Consider reflecting the new staff position regarding LOOP and single failures assumed in accident and transient analyses.

PIPB Comment:

SRXB confirmed that this staff position was new to the CE 80+ SER and not previously addressed. GDC 17 requires consideration of a LOOP in transient analysis and LOOP should be considered in all events. Lacking specific guidance in the SRP, GDC 17 prevails as an acceptance criterion and should be added to those sections where turbine trip results from the transient.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23118/FINAL SER CE80 CH 15

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**Integrated Impact Number:** 1473      **SRP Section Number:** 15.0

#### **Suggested Changes to the SRP Section:**

Incorporate the staff position on use of more recent source term assumptions than those in TID 14844.

Potential offsite doses due to accidents involving fuel damage have traditionally been predicted using the approach outlined in TID 14844. TID 14844 is identified in 10 CFR 100.11 as guidance for dose calculations. The assumptions of TID 14844 have also been used in development of dose assessment guidance such as Regulatory Guides 1.3, 1.4 and 1.77. 10 CFR 100.11 is identified as Acceptance Criteria and various Regulatory Guides using TID 14844 methodology are identified as guidance in SRP sections that evaluate the radiological consequences of accidents.

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10 CFR 50.34(f)(2)(xxviii) requires the evaluation of pathways that may lead to control room habitability problems "under accident conditions resulting in TID 14844 source term release." Similar wording appears in subparagraphs (vii), (viii), and (xxvi). These regulations have been interpreted as requiring use of TID 14844.

In SECY 90-016, the staff recommended that deviations from past methodologies used to calculate 10 CFR Part 100 doses be evaluated on a case-by-case basis, while ensuring that the requirements of Part 100 are met. In the staff's Final Safety Evaluation Report for ABB-CE System 80+ design certification, they accepted an exemption from the requirements of 10 CFR 50.34(f)(2) to evaluate release pathways under accident conditions resulting in a TID 14844 source term release and allowed ABB-CE to implement the new source term technology summarized in Draft NUREG-1465. Consistent with this exemption and the position stated in SECY 90-016, the staff reviewed the applicant's analyses of the radiological consequences of accidents using the assumptions specified in draft NUREG-1465.

Consider revising SRP section 15.0 to acknowledge use of newer source term guidance other than TID 14844.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25910/CODE OF FED. REGS 10CFR100; 25911/CODE OF FED. REGS 10CFR50;  
25912/FINAL SER CE80 CH 15; 25913/SECY 90-016; 25914/SECY 93-087

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**Integrated Impact Number:** 1474      **SRP Section Number:** 6.2.4

#### **Suggested Changes to the SRP Section:**

Add a Review Procedure incorporating staff guidance on containment closure during shutdown operations.

Shutdown and low-power operations have presented challenges to safe plant operation and the NRC staff has concerns about the overall safety of operations during shutdown and low-power conditions. In the staff's evaluation, shutdown and low-power operations encompasses operation when the reactor is in a subcritical state or is in transition between subcriticality and power operation up to 5% of rated power and addresses only conditions for which there is fuel in the reactor vessel. The NRC staff is considering the most appropriate regulatory approach to improve safety during shutdown and low-power operations. NUREG 1449 documents the staff's evaluation and recommendations for new regulatory guidance or requirements in five key areas: (1) Outage Planning and Control; (2) Fire Protection; (3) Operations, Training, Procedures, and Other Contingency Plans; (4) Technical Specifications; and (5) Instrumentation. The ABWR FSER and the CE80+ FSER addressed shutdown and low-power risk by requiring

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documentation of an assessment of the risks and the design features utilized to minimize the risks. Generic Letter 88-17 required PWRs to implement procedures and administrative controls to reasonably assure containment closure can be achieved during shutdown operations.

Consideration should be given to adding a Review Procedure to address the necessary controls and procedures for containment closure during shutdown operations.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25745/SECY 93-087; 25751/FINAL SER ABWR CH 19; 25752/NRC GENERIC LETTER 88-17; 25753/NUREG 1449

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**Integrated Impact Number:** 1475      **SRP Section Number:** 13.5.2.1

#### **Suggested Changes to the SRP Section:**

Revise the Acceptance Criteria, Review Procedures and Appendix A to cite staff guidance regarding emergency operating procedure (EOP) development.

In NUREG-1358 and Supplement 1 thereto, the staff provides summaries of the special inspection program for EOPs conducted from 1988 to 1991. Neither of the NUREGs establishes any new staff positions or requirements, but were provided to licensees for guidance in improving and updating their EOPs.

In Appendix J of the FSER for the GE ABWR design, the staff cites NUREG-1358 as providing guidance to be used in the development of plant procedures, including EOPs.

Consider revising the Acceptance Criteria, Review Procedures, and Appendix A to include NUREG-1358 and Supplement 1 thereto.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25915/NUREG 1358; 25916/NUREG 1358 SUP 1; 25917/FINAL SER ABWR APPENDIX J

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**Integrated Impact Number:** 1476      **SRP Section Number:** 6.2.6

#### **Suggested Changes to the SRP Section:**

Modify review procedures to describe the Appendix J "Type A Testing" exemption approved by the staff in the System 80+ FSER.

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In their SAR for the System 80+ design ABB-CE committed to containment leakage testing with the exception that Type A testing would not be halted due to leaks that occurred during the test. ABB-CE proposed that such leaks would be isolated, the Type A test would continue on to completion, and a "pre-maintenance" local leak rate test would be conducted on the leaking component. If the results of this local leak rate test, when added to the the results from the Type A test, are in conformance with Appendix J, the Type A test would be considered satisfactory. Otherwise, the leaking component would be repaired, a "post-maintenance" local leak rate test would be performed on the component, and, if the results of this test when added to the results of the Type A test meet Appendix J, then the Type A test would be considered satisfactory. The staff stated that this exemption to Appendix J was justified since it meets the intent of the regulation.

Consider modifying Review Procedures to describe the Appendix J exemption approved by the staff in the System 80+ FSER.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25656/FINAL SER CE80 CH 6

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**Integrated Impact Number:** 1479      **SRP Section Number:** 13.5.2.2

#### **Suggested Changes to the SRP Section:**

Update current discussions of review applicability to address the review of DC and COL applications.

The standard plant licensing process of 10 CFR 52 has provisions for determining that certain aspects of a design may be determined to be outside the scope of design certification (DC) and consequently must be addressed by a combined license (COL) applicant referencing the certified design. The ABWR and CE 80+ FSERs indicate that the staff has determined that development of detailed procedures and associated training materials may be beyond the scope of design certification and the responsibility of a COL applicant referencing the certified design. SRP Section 13.5.2 currently describes differences in the review of plant procedures for the cases of applications for construction permits or operating licenses.

Consideration should be given to revising SRP 13.5.2 to indicate that development of detailed procedures and associated training material may be beyond the scope of design certification and in such a case, would be the responsibility of a COL applicant referencing the certified design.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25925/FINAL SER CE80 CH 13; 25926/FINAL SER ABWR CH 13

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**Integrated Impact Number:** 1481      **SRP Section Number:** 13.5.2.2

#### **Suggested Changes to the SRP Section:**

Revise Acceptance Criteria and add Review Procedures regarding operating and maintenance procedures other than those used by licensed operators in the control room.

NUREG-0933 item HF4.4, "Guidelines for Upgrading Other Procedures"; TMI Item I.C.1, "Short-term accident and procedures review"; TMI Item I.C.5, "Feedback of Operating Experience"; and TMI Item I.C.9, "Long Term Plan for Upgrading Procedures," provide criteria regarding development, verification, and validation, implementation, maintenance, and revision of plant procedures. 10 CFR 34(f)(2)(ii), related to I.C.9 requires that applicants establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures.

The CE80+ and ABWR FSERs indicated that the staff included the considerations of I.C.1, I.C.5, and I.C.9 in their review of the methods and criteria for the development, V&V, implementation, maintenance, and revision of plant procedures. SRP Section 13.5.2 currently describes separate Acceptance Criteria for procedures used by licensed operators in the control room and other operating and maintenance procedures. The Acceptance Criteria for other operating and maintenance procedures does not identify HF-4.4, I.C.1, I.C.5, and I.C. 0.

Consideration should be given to revising Acceptance Criteria and adding Review Procedures regarding operating and maintenance procedures other than those used in the control room to address the considerations identified above.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25923/CODE OF FED. REGS 10CFR50; 25924/NUREG 0933; 25930/FINAL SER CE80 CH 13; 25931/FINAL SER ABWR CH 13

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**Integrated Impact Number:** 1482      **SRP Section Number:** 13.5.2.2

#### **Suggested Changes to the SRP Section:**

10 CFR 50, Appendix B, criteria V and VI, establish requirements for the development, approval, and control of procedures for all activities affecting quality. 10 CFR 50, Appendix B,



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is identified in the regulatory basis for Regulatory Guide 1.33 currently cited in SRP Section 13.5.2. These requirements are not currently identified in SRP Section 13.5.2 Acceptance Criteria.

Consideration should be given to adding citations of 10 CFR 50, Appendix B, criterion V and criterion VI, to SRP Acceptance Criteria.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25932/CODE OF FED. REGS 10CFR50; 25933/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1487      **SRP Section Number:** 6.2.1.1.A

#### **Suggested Changes to the SRP Section:**

Add a new Review Procedure incorporating staff guidance on containment analyses that should be performed to develop sufficient bases for shutdown operations.

Shutdown and low-power operations have presented challenges to safe plant operation and the NRC staff has concerns about the overall safety of operations during shutdown and low-power conditions. In the staff's evaluation, shutdown and low-power operations encompasses operation when the reactor is in a subcritical state or is in transition between subcriticality and power operation up to 5% of rated power and addresses only conditions for which there is fuel in the reactor vessel. The NRC staff is considering the most appropriate regulatory approach to improve safety during shutdown and low-power operations. NUREG 1449 documents the staff's evaluation and recommendations for new regulatory guidance or requirements in five key areas: (1) Outage Planning and Control; (2) Fire Protection; (3) Operations, Training, Procedures, and Other Contingency Plans; (4) Technical Specifications; and (5) Instrumentation. The CE80+ FSER addressed shutdown and low-power risk by requiring documentation of an assessment of the risks and the design features utilized to minimize the risks. Generic Letter 88-17 required PWRs to conduct analyses to supplement existing information and develop sufficient bases for shutdown operation.

Consideration should be given to adding a review procedure to address containment analyses that may be required to support shutdown operations.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25758/SECY 93-087; 25759/FINAL SER CE80 CH 19; 25760/NRC GENERIC LETTER 88-17; 25761/NUREG 1449

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**Integrated Impact Number:** 1488      **SRP Section Number:** 15.5.1 - 15.5.2

#### **Suggested Changes to the SRP Section:**

Add GDC 17 as an Acceptance Criterion and incorporate staff position from CE 80+ into Review Procedures.

In the CE 80+ FSER, the staff stated that, in accordance with the requirements of GDC 17, the loss-of-offsite power (LOOP) should not be considered as a single failure event, but should be assumed in the analysis for each event without changing the event category.

Consider modifying Acceptance Criteria to include GDC 17 and revising Review Procedures to incorporate staff guidance for the assumption of LOOP, in addition to the limiting single failure event, for the analysis of reactor coolant pump rotor seizure and reactor coolant pump shaft breaks.

#### **PIPB Comment:**

SRXB confirmed that this staff position was new to the CE 80+ SER and not previously addressed. GDC 17 requires consideration of a LOOP in transient analysis and LOOP should be considered in all events. Lacking specific guidance in the SRP, GDC 17 prevails as an acceptance criterion and should be added to those sections where turbine trip results from the transient.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25743/FINAL SER CE80 CH 15; 25744/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1489      **SRP Section Number:** 15.5.1 - 15.5.2

#### **Suggested Changes to the SRP Section:**

Include a discussion of the unique acceptance criterion accepted in the ABWR FSER for the postulated pressure regulator downscale failure. The transient is caused by a postulated common-mode software failure of the three independent pressure regulators. The NRC staff classified this postulated event in a special category of anticipated transients involving a common-mode software failure and established a special acceptance criterion for the radiological dose calculation.

Normally, such transients are treated as anticipated operational occurrences (AOOs) for which the acceptance criteria is that specified acceptable fuel design limits should not be exceeded.

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For this special case, the NRC staff concluded that the transient was uncertain to occur during the plant lifetime (not an AOs, but classified as anticipated transients involving a common-mode software failure), and therefore established a special acceptance criterion for the radiological dose calculation: the resulting dose, accounting for fuel failures, should not exceed 10 percent of 10 CFR Part 100, which the staff considered appropriate for an event of such postulated frequency. This is due to the unique design features of the ABWR instrumentation and control systems, which should reduce the frequency of such events and therefore allow these events to be recategorized as special cases.

Consideration should be given to discussing this unique acceptance criterion for the ABWR's postulated downscale failure of pressure regulators.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25742/FINAL SER ABWR CH 15

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**Integrated Impact Number:** 1490      **SRP Section Number:** 3.10

#### **Suggested Changes to the SRP Section:**

SRP Section 3.10 cites ANSI N41.6 DRAFT. The citation is non-date-specific with regard to the applicable standard version.

Standard IEEE 382-1972 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI N41.6 (draft) to cite the IEEE 382-1972 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23214/C&S: IEEE 382

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**Integrated Impact Number:** 1491      **SRP Section Number:** 3.11

#### **Suggested Changes to the SRP Section:**

Incorporate Regulatory Guide 1.151 as guidance in Acceptance Criteria of this SRP section.

Regulatory Guide 1.151 states, in part, that the requirements of ISA-S67.02, "Nuclear Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants", provide an acceptable basis to the staff for the design and installation of safety related instrument sensing lines in nuclear power plants subject to the regulatory positions and

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recommendations as specified in the subject guide.

Regulatory Positions C.4 and C.5 of Regulatory Guide 1.151 include staff positions regarding potential freezing of condensate or liquid in instrument sensing lines, based on connection to safety related systems and the safety significance of the connected instrumentation. Additionally, monitoring and alarming systems are recommended at a level of redundancy commensurate with the safety significance of the system potentially affected by occasional freezing conditions. Although this issue is not part of the qualification program for harsh environments, it is proper to address it in this SRP section, under the "mild environment" portion.

Consideration should be given to including the guidance of Regulatory Guide 1.151, with regard to the potential for freezing in instrument sensing lines, in Acceptance Criteria and Evaluation Findings.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25650/RG 1.151

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**Integrated Impact Number:** 1492      **SRP Section Number:** 5.2.4

#### **Suggested Changes to the SRP Section:**

Incorporate a discussion of augmented inspections of CRD and feedwater nozzles in BWRs. NUREG-0619 details the staff's efforts in resolving Generic Technical Issue A-10, "BWR Nozzle Cracking," and contains the solutions recommended by the staff with respect to the nozzles, spargers, cladding, leakage, operating procedures, and inservice testing. Generic Letter 80-95 forwarded NUREG-0619 for implementation. Generic Letter 81-11 provides clarification to portions of NUREG-0619.

Consider incorporating in Review Procedures a discussion on augmented inspections of CRD and feedwater nozzles as required by NUREG-0619.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25942/NUREG 0619; 25943/NRC GENERIC LETTER 80-95; 25944/NRC GENERIC LETTER 81-11

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**Integrated Impact Number:** 1493      **SRP Section Number:** 5.4.7

**Suggested Changes to the SRP Section:**

SRP Section 5.4.7 cites IEEE 338. The citation is non-date-specific with regard to the applicable standard version.

Standard IEEE 338-1977 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of IEEE 338 to cite the 1977 version.

**Potential Impacts/Documents supporting the Suggested Changes:**

23991/C&S: IEEE 338

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**Integrated Impact Number:** 1494      **SRP Section Number:** 3.11

**Suggested Changes to the SRP Section:**

SRP Section 3.11 cites IEEE 334. The citation is non-date-specific with regard to the applicable standard version.

Standard IEEE 334-1971 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of IEEE 334 to cite the 1971 version.

**Potential Impacts/Documents supporting the Suggested Changes:**

6434/RG 1.40

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**Integrated Impact Number:** 1495      **SRP Section Number:** 3.11

**Suggested Changes to the SRP Section:**

SRP Section 3.11 cites IEEE 382. The citation is non-date-specific with regard to the applicable standard version.

Standard IEEE 382-1972 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of IEEE 382 to cite the 1972 version.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24189/C&S: IEEE 382-72

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**Integrated Impact Number:** 1496      **SRP Section Number:** 3.11

#### **Suggested Changes to the SRP Section:**

SRP Section 3.11 cites IEEE 649. The citation is non-date-specific with regard to the applicable standard version.

Standard IEEE 649-1980 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of IEEE 649 to cite the 1980 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23981/C&S: IEEE 649

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**Integrated Impact Number:** 1497      **SRP Section Number:** 3.11

#### **Suggested Changes to the SRP Section:**

SRP Section 3.11 cites IEEE 650. The citation is non-date-specific with regard to the applicable standard version.

Standard IEEE 650-1979 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of IEEE 650 to cite the 1979 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23980/C&S: IEEE 650

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**Integrated Impact Number:** 1498      **SRP Section Number:** 5.4.2.2

#### **Suggested Changes to the SRP Section:**

Incorporate guidance for application of alternate repair criteria for outside diameter stress corrosion cracking (ODSCC) at the tube-to-tube support plate intersections in Westinghouse-designed steam generators having drilled-hole tube support plates (TSPs) and

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alloy 600 steam generator tubing.

Generic Letter 95-05 offers guidance on the implementation of an alternate repair criterion to be applied to predominantly axially oriented ODSCC at TSP locations. This criterion does not set limits on the depth of ODSCC indications to ensure tube integrity margins; instead, it relies on correlating the eddy current voltage amplitude from a bobbin coil probe with the more specific measurement of burst pressure and leak rate. The staff recognizes that although total margin may be reduced following application of the voltage-based repair guidance of this generic letter, this guidance does ensure that structural and leakage integrity continues to be maintained with an acceptable level of margin consistent with the applicable GDCs of 10 CFR Part 50, Appendix A and the limits of 10 CFR Part 100. Since the voltage-based repair criteria do not incorporate minimum wall thickness requirements, there is a possibility that tubes with up to 100-percent through-wall cracks can remain in service. Because of the increased likelihood of through-wall cracks, the staff has included provisions for augmented steam generator tube inspections and more restrictive operational leakage limits in this generic letter guidance.

Consider adding Review Procedure guidance for application of alternate repair criteria for ODSCC at the tube-to-tube support plate intersections in Westinghouse-designed steam generators having drilled-hole TSPs and alloy 600 steam generator tubing.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25945/NRC GENERIC LETTER 95-05

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**Integrated Impact Number:** 1499      **SRP Section Number:** 9.1.1

#### **Suggested Changes to the SRP Section:**

SRP Section 9.1.1 cites ANS 57.1. The citation is non-date-specific with regard to the applicable standard version.

Standard ANS 57.1-1980 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANS 57.1 to cite the 1980 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22841/C&S: ANS 57.1

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**Integrated Impact Number:** 1500      **SRP Section Number:** 9.1.1

#### **Suggested Changes to the SRP Section:**

SRP Section 9.1.1 cites ANS 57.3 . The citation is non-date-specific with regard to the applicable standard version.

Standard ANS 57.3-1981 (Draft) has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANS 57.3 to cite the 1981 draft version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

22842/C&S: ANS 57.3

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**Integrated Impact Number:** 1501      **SRP Section Number:** 6.2.1.1.B

#### **Suggested Changes to the SRP Section:**

Add a new Review Procedure incorporating staff guidance on containment analyses that should be performed to develop sufficient bases for shutdown operations.

Shutdown and low-power operations have presented challenges to safe plant operation and the NRC staff has concerns about the overall safety of operations during shutdown and low-power conditions. In the staff's evaluation, shutdown and low-power operations encompasses operation when the reactor is in a subcritical state or is in transition between subcriticality and power operation up to 5% of rated power and addresses only conditions for which there is fuel in the reactor vessel. The NRC staff is considering the most appropriate regulatory approach to improve safety during shutdown and low-power operations. NUREG 1449 documents the staff's evaluation and recommendations for new regulatory guidance or requirements in five key areas: (1) Outage Planning and Control; (2) Fire Protection; (3) Operations, Training, Procedures, and Other Contingency Plans; (4) Technical Specifications; and (5) Instrumentation. The CE80+ FSER addressed shutdown and low-power risk by requiring documentation of an assessment of the risks and the design features utilized to minimize the risks. Generic Letter 88-17 required PWRs to conduct analyses to supplement existing information and develop sufficient bases for shutdown operation.

Consideration should be given to adding a Review Procedure to address containment analyses that may be required to support shutdown operations.



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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25762/SECY 93-087; 25763/FINAL SER CE80 CH 19; 25764/NRC GENERIC LETTER 88-17; 25765/NUREG 1449

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**Integrated Impact Number:** 1502      **SRP Section Number:** 6.2.1.1.C

#### **Suggested Changes to the SRP Section:**

Add a new Review Procedure incorporating staff guidance on containment analyses that should be performed to develop sufficient bases for shutdown operations.

Shutdown and low-power operations have presented challenges to safe plant operation and the NRC staff has concerns about the overall safety of operations during shutdown and low-power conditions. In the staff's evaluation, shutdown and low-power operations encompasses operation when the reactor is in a subcritical state or is in transition between subcriticality and power operation up to 5% of rated power and addresses only conditions for which there is fuel in the reactor vessel. The NRC staff is considering the most appropriate regulatory approach to improve safety during shutdown and low-power operations. NUREG 1449 documents the staff's evaluation and recommendations for new regulatory guidance or requirements in five key areas: (1) Outage Planning and Control; (2) Fire Protection; (3) Operations, Training, Procedures, and Other Contingency Plans; (4) Technical Specifications; and (5) Instrumentation. The ABWR FSER addressed shutdown and low-power risk by requiring documentation of an assessment of the risks and the design features utilized to minimize the risks.

Consideration should be given to adding a Review Procedure to address containment analyses that may be required to support shutdown operations.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25769/SECY 93-087; 25770/FINAL SER ABWR CH 19; 25771/NUREG 1449

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**Integrated Impact Number:** 1504      **SRP Section Number:** 10.2.3

#### **Suggested Changes to the SRP Section:**

SRP Section 10.2.3 cites ASTM A370. The citation is non-date-specific with regard to the applicable standard version.

Standard ASTM A370-1972 has been determined to be the version applicable to the SRP citation.

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Consider updating the citation of ASTM A370 to cite the 1972 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23726/C&S: ASTM A370

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**Integrated Impact Number:** 1505      **SRP Section Number:** 10.2.3

#### **Suggested Changes to the SRP Section:**

SRP Section 10.2.3 cites ASTM E208. The citation is non-date-specific with regard to the applicable standard version.

Standard ASTM E208-1969 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ASTM E208 to cite the 1969 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23725/C&S: ASTM E208

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**Integrated Impact Number:** 1506      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

SRP Section 17.1 cites ANSI N45.2.9. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI N45.2.9-1974 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI N45.2.9 to cite the 1974 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25661/C&S: ANSI N45.2.9

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**Integrated Impact Number:** 1507      **SRP Section Number:** 17.1

**Suggested Changes to the SRP Section:**

SRP Section 17.1 cites NFPA 232. The citation is non-date-specific with regard to the applicable standard version.

Standard NFPA 232-1980 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of NFPA 232 to cite the 1980 version.

**Potential Impacts/Documents supporting the Suggested Changes:**

24039/C&S: NFPA 232

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**Integrated Impact Number:** 1509      **SRP Section Number:** 13.1.1

**Suggested Changes to the SRP Section:**

SRP Section 13.1.1 cites ANSI N18.1. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI N18.1-1971 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI N18.1 to cite the 1971 version.

**Potential Impacts/Documents supporting the Suggested Changes:**

22573/C&S: ANSI N18.1

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**Integrated Impact Number:** 1510      **SRP Section Number:** 13.1.2 - 13.1.3

**Suggested Changes to the SRP Section:**

SRP Section 13.1.2 - 13.1.3 cites ANSI N18.1. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI N18.1-1971 has been determined to be the version applicable to the SRP citation.

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Consider updating the citation of ANSI N18.1 to cite the 1971 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24022/C&S: ANSI N18.1

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**Integrated Impact Number:** 1511      **SRP Section Number:** 13.1.2 - 13.1.3

#### **Suggested Changes to the SRP Section:**

SRP Section 13.1.2 - 13.1.3 cites ANSI N18.7. The citation is non-date-specific with regard to the applicable standard version.

Standard ANSI N18.7-1976/ANS-3.2 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANSI N18.7 to cite the 1976 version and include the ANS-3.2 designation.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24023/C&S: ANSI N18.7

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**Integrated Impact Number:** 1512      **SRP Section Number:** 15.3.1 - 15.3.2

#### **Suggested Changes to the SRP Section:**

Add GDC 17 as an Acceptance Criterion and incorporate staff position from CE 80+ into Review Procedures.

In the CE 80+ FSER, the staff stated that, in accordance with the requirements of GDC 17, the loss-of-offsite power (LOOP) should not be considered as a single failure event, but should be assumed in the analysis for each event without changing the event category.

Consider modifying Acceptance Criteria to include GDC 17 and revising Review Procedures to incorporate staff guidance for the assumption of LOOP, in addition to the limiting single failure event, for the analysis of reactor coolant pump rotor seizure and reactor coolant pump shaft breaks.

PIPB Comment:

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SRXB confirmed that this staff position was new to the CE 80+ SER and not previously addressed. GDC 17 requires consideration of a LOOP in transient analysis and LOOP should be considered in all events. Lacking specific guidance in the SRP, GDC 17 prevails as an acceptance criterion and should be added to those sections where turbine trip results from the transient.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25740/FINAL SER CE80 CH 15; 25741/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1513      **SRP Section Number:** 15.3.3 - 15.3.4

#### **Suggested Changes to the SRP Section:**

Add GDC 17 as an Acceptance Criterion and incorporate staff position from CE 80+ into Review Procedures.

In the CE 80+ FSER, the staff stated that, in accordance with the requirements of GDC 17, the loss-of-offsite power (LOOP) should not be considered as a single failure event, but should be assumed in the analysis for each event without changing the event category.

Consider modifying Acceptance Criteria to include GDC 17 and revising Review Procedures to incorporate staff guidance for the assumption of LOOP, in addition to the limiting single failure event, for the analysis of reactor coolant pump rotor seizure and reactor coolant pump shaft breaks.

#### **PIPB Comment:**

SRXB confirmed that this staff position was new to the CE 80+ SER and not previously addressed. GDC 17 requires consideration of a LOOP in transient analysis and LOOP should be considered in all events. Lacking specific guidance in the SRP, GDC 17 prevails as an acceptance criterion and should be added to those sections where turbine trip results from the transient.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25734/CODE OF FED. REGS 10CFR50; 25735/FINAL SER CE80 CH 15

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**Integrated Impact Number:** 1514      **SRP Section Number:** 13.3

#### **Suggested Changes to the SRP Section:**

Include NUREG 1394 and Generic Letter 91-014 as guidance documents relating to requirements associated with the Emergency Response Data System and Emergency Telecommunications.

Appendix E of 10 CFR Part 50, Section VI, requires that licensees implement the Emergency Response Data System (ERDS), a direct near-real-time electronic data link between the licensee's onsite computer system and the NRC's Operations Center that provides for the automated transmission of a limited data set of selected parameters. Additional regulations presented in 10 CFR 50.47(b) (6) and 10 CFR 50, Appendix E, IV.E.9d requires provisions be made for communications by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility.

NUREG 1394 was developed to provide guidance that the Staff believes should be followed to meet the requirements of Appendix E of 10 CFR Part 50, Section VI. Generic Letter 91-014 discussed a requirement that licensees convert to the Federal Telecommunications System (FTS) 2000 network as a necessary step to maintain assured and reliable communications during an emergency as well as for licensee reporting of events during normal operations. Conversion to the FTS 2000 network allows compliance with 10 CFR 50.47(b)(6) and 10 CFR Part 50, Appendix E, IV.E.9d.

Consideration should be given to incorporating NUREG 1394 and Generic Letter 91-014 as guidance documents relating to the compliance of emergency telecommunication systems with established requirements.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24485/NRC GENERIC LETTER 91-14; 24486/NRC GENERIC LETTER 93-01; 25953/NRC GENERIC LETTER 89-15; 25954/CODE OF FED. REGS 10CFR50; 25955/NUREG 1394

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**Integrated Impact Number:** 1515      **SRP Section Number:** 17.4

#### **Suggested Changes to the SRP Section:**

Commission guidance relative to a Reliability Assurance Program (RAP) was provided in the Staff Requirements Memo (SRM) dated June 30, 1994 and in SECY-95-132, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems (RTNSS) in

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Passive Plant Designs," dated May 22, 1995.

A Rulemaking is proposed which would add Appendix A, "Design Certification Rule for the GE Advanced Boiling Water Reactor Design," and Appendix B, "Design Certification Rule for the ABB CE System 80+ Design," to 10 CFR Part 52.

In SECY-94-084, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994, the staff gave its final position on the RAP for both evolutionary and passive ALWRs. For design certification of all ALWRs, a RAP applicable to design certification (D-RAP) would be required, and for a Combined License (COL) application that references a certified design, a RAP plan (augmented D-RAP and O-RAP) and inspections, tests, analyses, and acceptance criteria (ITAAC) would be required. The COL applicant would be required to augment the design certification D-RAP with a RAP plan that integrates the design certification D-RAP, site-specific design information, the procurement process, and the O-RAP.

In the Staff Requirements Memo (SRM) of June 30, 1994, the Commission approved the D-RAP approach subject to implementation of the D-RAP using the ITAAC process, and disapproved the O-RAP requirements.

In SECY-95-132, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," dated May 22, 1995, the staff modified its approach to the RAP plan in accordance with the guidance in the June 30, 1994 SRM. The staff retained the two stage approach. The first stage (D-RAP) would apply to prior to the initial fuel load. Second stage activities can be integrated into existing programs (e.g., maintenance, surveillance testing, inservice inspection, inservice testing, and quality assurance.) Reliability performance goals for risk- significant SSCs would be established consistent with the existing maintenance and quality assurance processes on the basis of information from the D-RAP. The D-RAP would be verified using the ITAAC process.

The COL applicant would establish performance and condition monitoring requirements to provide reasonable assurance that risk-significant SSCs do not degrade to an unacceptable level during plant operations. Most of the operational reliability assurance activities could be incorporated into the requirements of the maintenance rule, 10 CFR 50.65. Implementation of the maintenance rule following the guidance contained in RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," could meet the objective of reliability assurance for monitoring and correcting degradation in SSC reliability or availability associated with maintenance. The remaining operational reliability assurance activities for risk significant SSCs should be incorporated using existing regulatory programs (e.g., quality assurance, inservice inspections, inservice testing and surveillance testing.) For nonsafety-related, risk-significant SSC failures caused by design deficiencies or operational errors not related to

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maintenance, cause determinations and corrective actions should be performed as part of operational reliability assurance process. The cause determination and corrective action process does not have to be as stringent as the root cause analysis and corrective action process under the quality assurance program. However, a COL's normal work order process should be able to incorporate this operational reliability assurance objective for nonsafety-related, risk significant SSCs.

Consider developing SRP Section 17.4 to reflect the Commission approved approach to NRC staff review of RAP requirements.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25956/SECY 95-132; 25957/SECY 94-084; 25958/CODE OF FED. REGS 10CFR50;  
25959/RG 1.160

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**Integrated Impact Number:** 1516      **SRP Section Number:** 13.4

#### **Suggested Changes to the SRP Section:**

SRP Section 13.4 cites ANS 3.1 version 1978.

The latest version of this standard is 1993.

Consideration should be given to updating the citation of ANS 3.1 to cite the 1993 version.

This ROC is a placeholder and will not be processed further at this time.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24028/C&S: ANS 3.1-78

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**Integrated Impact Number:** 1517      **SRP Section Number:** 13.2.2

#### **Suggested Changes to the SRP Section:**

SRP Section 13.2.2 cites ANS 3.1 . The citation is non-date-specific with regard to the applicable standard version.

Standard ANS 3.1-1981 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of ANS 3.1 to cite the 1981 version.



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#### **Potential Impacts/Documents supporting the Suggested Changes:**

24026/C&S: ANS 3.1

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**Integrated Impact Number: 1518      SRP Section Number: 8.1**

#### **Suggested Changes to the SRP Section:**

SRP Section 8.1 cites IEEE 308. The citation is non-date-specific with regard to the applicable standard version.

Standard IEEE 308-1974 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of IEEE 308 to cite the 1974 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24206/C&S: IEEE 308-80

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**Integrated Impact Number: 1519      SRP Section Number: 8.1**

#### **Suggested Changes to the SRP Section:**

SRP Section 8.1 cites IEEE 317. The citation is non-date-specific with regard to the applicable standard version.

Standard IEEE 317-1983 (R92) has been determined to be the version applicable to the SRP citation.

Consider updating the citation of IEEE 317 to cite the 1983 (R92) version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

23994/C&S: IEEE 317; 25675/RG 1.63

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**Integrated Impact Number: 1520      SRP Section Number: 8.1**

#### **Suggested Changes to the SRP Section:**

SRP Section 8.1 cites IEEE 338. The citation is non-date-specific with regard to the applicable standard version.

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Standard IEEE 338-1987 (R94) has been determined to be the version applicable to the SRP citation.

Consider updating the citation of IEEE 338 to cite the 1987 (R94) version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

18806/RG 1.118; 25637/RG 1.118

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**Integrated Impact Number:** 1521      **SRP Section Number:** 8.1

#### **Suggested Changes to the SRP Section:**

SRP Section 8.1 cites IEEE 387. The citation is non-date-specific with regard to the applicable standard version.

Standard IEEE 387-1984 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of IEEE 387 to cite the 1984 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25662/C&S: IEEE 387

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**Integrated Impact Number:** 1522      **SRP Section Number:** 8.1

#### **Suggested Changes to the SRP Section:**

SRP Section 8.1 cites IEEE 450. The citation is non-date-specific with regard to the applicable standard version.

Standard IEEE 450-1975 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of IEEE 450 to cite the 1975 version.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

24202/C&S: IEEE 450-75

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**Integrated Impact Number:** 1523      **SRP Section Number:** 8.1

**Suggested Changes to the SRP Section:**

SRP Section 8.1 cites IEEE 384. The citation is non-date-specific with regard to the applicable standard version.

Standard IEEE 384-1974 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of IEEE 384 to cite the 1974 version.

**Potential Impacts/Documents supporting the Suggested Changes:**

18814/RG 1.75

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**Integrated Impact Number:** 1524      **SRP Section Number:** 8.1

**Suggested Changes to the SRP Section:**

SRP Section 8.1 cites IEEE 484. The citation is non-date-specific with regard to the applicable standard version.

Standard IEEE 484-1975 has been determined to be the version applicable to the SRP citation.

Consider updating the citation of IEEE 484 to cite the 1975 version.

**Potential Impacts/Documents supporting the Suggested Changes:**

18807/RG 1.128; 24201/C&S: IEEE 484-75

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**Integrated Impact Number:** 1525      **SRP Section Number:** 17.3

**Suggested Changes to the SRP Section:**

Add acceptance criteria for quality assurance guidance for accident monitoring instrumentation.

Regulatory Guide 1.97 in Table 1 provides design and qualification criteria for instrumentation to assess plant and environs conditions during and following an accident. Paragraph 5 of Table 1 specifies the quality assurance guidance for this type of instrumentation.

Consider adding acceptance criteria on the quality assurance guidance for accident monitoring

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instrumentation.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20237/RG 1.97

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**Integrated Impact Number:** 1526      **SRP Section Number:** 17.3

#### **Suggested Changes to the SRP Section:**

Regulatory Guide 1.155, Regulatory Position C.3.5, provides quality assurance guidance for station blackout equipment. Existing quality assurance requirements of Appendix B and Appendix R of Part 50 continue to apply. For nonsafety-related equipment that are used to meet the requirements of 10 CFR 50.63 and that are not already covered by existing QA requirements in Appendix B or R, the guidance provided in Appendices A and B of RG 1.155 apply as specified in position C.3.5.

Consider adding acceptance criteria on the quality assurance guidance for station blackout equipment.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

20203/RG 1.155

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**Integrated Impact Number:** 1527      **SRP Section Number:** 17.3

#### **Suggested Changes to the SRP Section:**

Update SRP Section 17.3 to address the unique requirements of optical disk storage systems.

Appendix B of 10 CFR 50, under criterion XVII, "Quality Assurance Records," establishes requirements for a record keeping system. The purpose of Generic Letter 88-18 is to inform addressees that the staff approves the use of optical disk document imaging systems for the storage and retrieval of record copies of quality assurance records when appropriate quality assurance controls are applied.

The generic letter lists appropriate quality controls for an optical disk document imaging system which could have an effect on the QA program review criteria of SRP Section 17.3.

Consider revising SRP Section 17.3 to address the unique requirements of optical disk storage systems and the provisions of Generic Letter 88-18.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25964/NRC GENERIC LETTER 88-18

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**Integrated Impact Number:** 1528      **SRP Section Number:** 8-A

#### **Suggested Changes to the SRP Section:**

Consider revising BTP ICSB-4 (PSB) to permit compliance with Regulatory Guide 1.153 guidance as an acceptable alternative and also consider eventually deleting the BTP as superseded by Regulatory Guide 1.153.

SRP Appendix 8A, Branch Technical Position (BTP) ICSB-4 (PSB) provides staff positions relative to meeting IEEE 279 (no date specified) "operating bypass" requirements for motor-operated isolation valves in ECCS accumulator lines.

10 CFR 50.55a, "Codes and Standards" requires in paragraph (h) that protection systems meet the requirements set forth in IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," or in the edition or revision in effect at the time a construction permit application is docketed. Regulatory Guide 1.153 describes a method acceptable to the NRC staff for complying with the 10 CFR 50.55a requirements and other Commission regulations with respect to the design, reliability, qualification, and testability of the power, instrumentation, and control portions of safety-related systems. Regulatory Guide 1.153, Rev. 0 endorses IEEE 603-1980 (a revision of IEEE 279) as supplemented by its regulatory positions. The NRC staff is also in the process of revising Regulatory Guide 1.153 (See Draft Regulatory Guide 1042) to update its endorsement from IEEE 603-1980 to IEEE 603-1991.

The staff does not state any exceptions to IEEE 603 operating bypass provisions in its endorsement in Regulatory Guide 1.153. Thus, the criteria of IEEE 603 represents an acceptable method for addressing operating bypasses for safety-related systems, including ECCS motor-operated isolation valves, and should therefore be referenced as an alternative to the positions stated in Section B of BTP ICSB 4 (PSB).

Consideration should therefore be given to revising BTP ICSB 4 (PSB) to reflect Regulatory Guide 1.153 as an acceptable alternative.

Consideration should also be given to performing a further evaluation of whether BTP ICSB 4 (PSB) may be deleted as superseded by Regulatory Guide 1.153. Action in this regard is tracked by IPD 7.0 Form 8-A-1.

It should be noted that BTP ICSB 4 also appears in Appendix A of Chapter 7 of the SRP and

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should be revised or deleted in both Chapter 7 and Chapter 8.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25960/RG 1.153; 25961/C&S: IEEE 279; 25962/C&S: IEEE 603

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**Integrated Impact Number:** 1530      **SRP Section Number:** 8-A

#### **Suggested Changes to the SRP Section:**

Consider revising BTP ICSB-21 (PSB) to acknowledge/reflect Regulatory Guide 1.153 guidance and also consider withdrawing Regulatory Guide 1.47 as superseded by Regulatory Guide 1.153.

SRP Appendix 8A, Branch Technical Position (BTP) ICSB-21 (PSB) provides staff positions relative to application of Regulatory Guide 1.47 guidance for bypass and inoperable status indication.

Regulatory Guide 1.153 describes a method acceptable to the NRC staff for complying with the requirements of Commission regulations with respect to the design, reliability, qualification, and testability of the power, instrumentation, and control portions of safety-related systems. Regulatory Guide 1.153, Rev. 0 endorses IEEE 603-1980 (a revision of IEEE 279) as supplemented by its regulatory positions. The NRC staff is also in the process of revising Regulatory Guide 1.153 (See Draft Regulatory Guide 1042) to update its endorsement from IEEE 603-1980 to IEEE 603-1991.

Based on the endorsement in Regulatory Guide 1.153, IEEE 603 is a method acceptable for complying with requirements for bypass and inoperable status indication for safety-related systems and the staff does not state any exceptions to IEEE 603 Section 5.8.3 in its endorsement. Thus, the guidance of Regulatory Guide 1.153 should be acknowledged/reflected in BTP ICSB-21 (PSB) as at least of equivalent importance to the guidance of Regulatory Guide 1.47.

Consideration should therefore be given to revising BTP ICSB-21 (PSB) to acknowledge/reflect Regulatory Guide 1.153 guidance.

Consideration should also be given to withdrawing Regulatory Guide 1.47 as superseded by Regulatory Guide 1.153. Action in this regard is tracked by IPD 7.0 Form 8-A-2.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25960/RG 1.153; 25962/C&S: IEEE 603; 25967/RG 1.47

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**Integrated Impact Number:** 1531      **SRP Section Number:** 8-A

#### **Suggested Changes to the SRP Section:**

Consider revising BTP PSB-1 to reflect Regulatory Guide 1.153 guidance.

SRP Appendix 8A, Branch Technical Position (BTP) PSB-1 provides staff positions relative to station electric distribution voltages including positions to provide a second level of undervoltage protection (commonly referred to as "degraded voltage protection") for Class 1E busses. Position B.1.c) addressed the voltage sensors for such protection schemes and requires that the sensors be designed to satisfy requirements characterized as "derived from IEEE Std 279-1971."

Regulatory Guide 1.153 describes a method acceptable to the NRC staff for complying with the requirements of Commission regulations with respect to the design, reliability, qualification, and testability of the power, instrumentation, and control portions of safety-related systems. Regulatory Guide 1.153, Rev. 0 endorses IEEE 603-1980 (a revision of IEEE 279) as supplemented by its regulatory positions. The NRC staff is also in the process of revising Regulatory Guide 1.153 (See Draft Regulatory Guide 1042) to update its endorsement from IEEE 603-1980 to IEEE 603-1991.

The guidance of Regulatory Guide 1.153 represents an acceptable method for addressing requirements related to IEEE Std 279 and therefore should be discussed in conjunction with discussions of the requirements "derived from IEEE Std 279."

Consideration should therefore be given to revising BTP PSB-1 to reflect Regulatory Guide 1.153 guidance.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25960/RG 1.153; 25961/C&S: IEEE 279; 25962/C&S: IEEE 603

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**Integrated Impact Number:** 1532      **SRP Section Number:** 8-A

#### **Suggested Changes to the SRP Section:**

Consider revising BTP PSB-2 to reflect the relevant guidance of Regulatory Guides 1.9, Rev. 3 and 1.153.

SRP Appendix 8A, Branch Technical Position (BTP) PSB-2 provides staff positions relative to bypass and inoperable status indication for diesel-generator units.

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Regulatory Guide 1.153 describes a method acceptable to the NRC staff for complying with the requirements of Commission regulations with respect to the design, reliability, qualification, and testability of the power, instrumentation, and control portions of safety-related systems. Regulatory Guide 1.153, Rev. 0 endorses IEEE 603-1980 (a revision of IEEE 279) as supplemented by its regulatory positions. The NRC staff is also in the process of revising Regulatory Guide 1.153 (See Draft Regulatory Guide 1042) to update its endorsement from IEEE 603-1980 to IEEE 603-1991. IEEE 603 includes comprehensive guidance for bypass and inoperable status indication that is acceptable for all safety-related systems.

Regulatory Guide 1.9, Rev. 3 provides guidance acceptable to the NRC staff for complying with the Commission's requirements that diesel generator units intended for use as onsite emergency power sources in nuclear power plants be selected with sufficient capacity, be qualified, and have the necessary reliability and availability for station blackout and design basis accidents. The Guide endorses IEEE 387-1984 with exceptions and includes positions that appear relevant to evaluation of bypass and inoperable status indication for diesel-generator units.

Consideration should therefore be given to revising BTP PSB-2 to reflect the relevant guidance of Regulatory Guides 1.9, Rev. 3 and 1.153.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25960/RG 1.153; 25962/C&S: IEEE 603; 25966/RG 1.9

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**Integrated Impact Number:** 1533      **SRP Section Number:** 14.3

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of Certified Design Material (CDM) and proposed inspections, tests, analyses and acceptance criteria (ITAAC).

The requirements for design certification applicants to submit material are contained in 10 CFR 52.47, and the requirements for ITAAC are provided in 10 CFR 52.97.

Following review and final design approval (FDA) by the NRC staff, standardized reactor designs are certified in rules issued as appendices to 10 CFR Part 52. Each appendix incorporates by reference a Design Control Document (DCD) that contains the design certification information for the respective design. An applicant for design certification is required to submit a DCD to the staff for review and approval. A combined license applicant or licensee referencing a standard design must comply with both the rule certifying the design and the DCD. The DCD includes the Tier 1 information that is certified by the rule and the Tier 2 information that is approved by the rule.



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The Tier 1 portion of the DCD is referred to as the CDM, and is derived from the Tier 2 information, which generally consists of the standard safety analysis report (SSAR) for the design. The CDM consists of an introduction; definitions, legends, and general provisions; design descriptions and ITAAC; interface requirements; and site parameters for the design. The purpose of the ITAAC is to verify that a facility that references the design certification has been built and will operate in accordance with the design certification and applicable regulations.

Consider developing Acceptance Criteria and Review Procedures to describe the review of the CDM portion of an application submitted under 10 CFR 52, including the associated ITAAC.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25968/CODE OF FED. REGS 10CFR52

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**Integrated Impact Number:** 1534      **SRP Section Number:** 14.3.1

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of certified design material (CDM), including associated inspections, tests, analyses, and acceptance criteria (ITACC) for site parameters.

The site parameters used in the design must be specified in both the CDM and Chapter 2 of the SSAR. The site parameters specified in the CDM are the top-level bounding site parameters used for in the selection of a suitable site for a facility referencing the certified design. Because they were used in bounding evaluations of the certified design, they define the requirements for the design that must be met by a site. This ensures that a facility built on the site remains in conformance with the design certification. Appropriate values for site parameters should be selected that make the design suitable for many sites. The site parameters specified in the SSAR Chapter 2 should be consistent with those in the CDM.

The site parameters should include a requirement that liquefaction not occur underneath structures, systems, and components resulting from the site-specific SSE. In addition, although the design for the sites should be based on the 0.3g RG 1.60 spectra, the evaluation of the sites for liquefaction potential should use the site-specific SSE with acceptance criteria demonstrating adequate margin for no liquefaction.

Consider adding Acceptance Criteria and Review Procedures for CDM and ITAAC related to review of site parameters.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25969/CODE OF FED. REGS 10CFR52; 25989/RG 1.60

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**Integrated Impact Number:** 1535      **SRP Section Number:** 14.3.2

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of certified design material (CDM), including associated inspections, tests, analyses and acceptance criteria (ITAAC) for plant structural and system engineering.

10 CFR Parts 52.47 and 52.97 provide the Commission's requirements with regard to the content of applications for design certification, and the issuance of combined licenses, respectively. The ITAAC must be reviewed and verified to provide assurance that compliance with the ITAAC ensures that a plant, that references a certified design and/or applies for a combined license, will be constructed and operated in accordance with the design certification and/or the requirements of the license.

10 CFR 52.47 and 52.97 require, in part, that applications for design certification and/or a combined license include evidence that the application is in accordance with Commissions regulations, including those in 10 CFR 50 as appropriate. Appendix A to 10 CFR 50 provides the General Design Criterion applicable to review of nuclear plant design. The primary criterion applicable to review of structural and systems engineering include GDCs 1, 2, 4, 14, 16, and 50, as well as the ASME Boiler and Pressure Vessel Code as endorsed in 10 CFR 50.55.

Consider adding Acceptance Criteria and Review Procedures for review of CDM and related ITAAC for plant structural and system engineering.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25970/CODE OF FED. REGS 10CFR52; 25979/CODE OF FED. REGS 10CFR50;  
25980/CODE OF FED. REGS 10CFR50; 25981/CODE OF FED. REGS 10CFR50;  
25982/CODE OF FED. REGS 10CFR50; 25983/CODE OF FED. REGS 10CFR50;  
25984/CODE OF FED. REGS 10CFR50; 25985/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1536      **SRP Section Number:** 14.3.3

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of certified design material

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(CDM), including associated inspections, tests, analyses and acceptance criteria (ITAAC) for plant piping and system components.

10 CFR Parts 52.47 and 52.97 provide the Commission's requirements with regard to the content of applications for design certification, and the issuance of combined licenses, respectively. The ITAAC must be reviewed and verified to provide assurance that compliance with the ITAAC ensures that a plant, that references a certified design and/or applies for a combined license, will be constructed and operated in accordance with the design certification and/or the requirements of the license.

10 CFR 50.55a and General Design Criteria 1, 14 and 30 provide requirements relative to the standards to be applied piping systems and components, specifically with regard to the reactor coolant pressure boundary.

For design certification applications, the application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted.

Consider adding Acceptance Criteria and Review Procedures for review of CDM and related ITAAC for plant piping and system components.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25971/CODE OF FED. REGS 10CFR52; 26035/CODE OF FED. REGS 10CFR50;  
26036/CODE OF FED. REGS 10CFR50; 26037/CODE OF FED. REGS 10CFR50;  
26038/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1537      **SRP Section Number:** 14.3.4

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of certified design material (CDM), including associated inspections, tests, analyses and acceptance criteria (ITAAC) for reactor systems

10 CFR Parts 52.47 and 52.97 provide the Commission's requirements with regard to the content of applications for design certification, and the issuance of combined licenses, respectively. The ITAAC must be reviewed and verified to provide assurance that compliance with the ITAAC ensures that a plant, that references a certified design and/or applies for a combined license, will be constructed and operated in accordance with the design certification and/or the requirements

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of the license.

For design certification applications, the application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted.

Important insights and design features from these PRA and severe accident analyses should be incorporated into the CDM. For the severe accident analyses in particular, the basis for the staff's review for the evolutionary standard designs was the Commission guidance related to SECYs 90-016 and 93-087, later included in the design certification rules for these designs. For both PRA and severe accident analyses, although large uncertainties and unknowns may be associated with the event phenomena, design features important for severe accident prevention and mitigation resulting from these analyses should be selected for treatment in the CDM.

Consider adding Acceptance Criteria and Review Procedures for review of CDM and related ITAAC for reactor systems.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25972/CODE OF FED. REGS 10CFR52; 25986/SECY 90-016; 25987/SECY 93-087

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**Integrated Impact Number:** 1538      **SRP Section Number:** 14.3.5

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of certified design material (CDM), including associated inspections, tests, analyses and acceptance criteria (ITAAC) for plant instrumentation and controls.

10 CFR Parts 52.47 and 52.97 provide the Commission's requirements with regard to the content of applications for design certification, and the issuance of combined licenses, respectively. The ITAAC must be reviewed and verified to provide assurance that compliance with the ITAAC ensures that a plant, that references a certified design and/or applies for a combined license, will be constructed and operated in accordance with the design certification and/or the requirements of the license.

10 CFR 52.47 and 52.97 require, in part, that applications for design certification and/or a combined license include evidence that the application is in accordance with Commissions regulations, including those in 10 CFR 50 as appropriate. Appendix A to 10 CFR 50 provides the General Design Criterion applicable to review of nuclear plant design. The primary criterion

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applicable to review of instrumentation and controls include GDCs 1, 2, 4, 13, 19, 20, 21, 22, 23, 25, and 29, as well as IEEE Std 279-1971 as endorsed in 10 CFR 50.55a.

Consider adding Acceptance Criteria and Review Procedures for review of CDM and related ITAAC for plant instrumentation and controls.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25973/CODE OF FED. REGS 10CFR52; 25990/CODE OF FED. REGS 10CFR50;  
25991/CODE OF FED. REGS 10CFR50; 25992/CODE OF FED. REGS 10CFR50;  
25993/CODE OF FED. REGS 10CFR50; 25994/CODE OF FED. REGS 10CFR50;  
25995/CODE OF FED. REGS 10CFR50; 25996/CODE OF FED. REGS 10CFR50;  
25997/CODE OF FED. REGS 10CFR50; 25998/CODE OF FED. REGS 10CFR50;  
25999/CODE OF FED. REGS 10CFR50; 26000/CODE OF FED. REGS 10CFR50;  
26001/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1539      **SRP Section Number:** 14.3.6

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of certified design material (CDM), including associated inspections, tests, analyses and acceptance criteria (ITAAC) for plant electrical systems.

10 CFR Parts 52.47 and 52.97 provide the Commission's requirements with regard to the content of applications for design certification, and the issuance of combined licenses, respectively. The ITAAC must be reviewed and verified to provide assurance that compliance with the ITAAC ensures that a plant, that references a certified design and/or applies for a combined license, will be constructed and operated in accordance with the design certification and/or the requirements of the license.

10 CFR 52.47 and 52.97 require, in part, that applications for design certification and/or a combined license include evidence that the application is in accordance with Commissions regulations, including those in 10 CFR 50 as appropriate. The NRC's regulations relevant to electrical design include 10 CFR Parts 50.49, 50.63, and 10 CFR 50, Appendix A, General Design Criterion (GDC) 17. IEEE Std. 308 also provides guidance acceptable to the staff for the design of Class 1E power systems.

Consider adding Acceptance Criteria and Review Procedures for review of CDM and related ITAAC for plant electrical systems.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

25974/CODE OF FED. REGS 10CFR52; 26003/CODE OF FED. REGS 10CFR50;  
26004/CODE OF FED. REGS 10CFR50; 26005/CODE OF FED. REGS 10CFR50;  
26006/C&S: IEEE 308

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**Integrated Impact Number:** 1540      **SRP Section Number:** 14.3.7

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of certified design material (CDM), including associated inspections, tests, analyses and acceptance criteria (ITAAC) for plant systems.

10 CFR Parts 52.47 and 52.97 provide the Commission's requirements with regard to the content of applications for design certification, and the issuance of combined licenses, respectively. The ITAAC must be reviewed and verified to provide assurance that compliance with the ITAAC ensures that a plant, that references a certified design and/or applies for a combined license, will be constructed and operated in accordance with the design certification and/or the requirements of the license.

10 CFR 52.47 and 52.97 require, in part, that applications for design certification and/or a combined license include evidence that the application is in accordance with Commissions regulations, including those in 10 CFR 50 as appropriate. Regulations applicable to plant systems include 10 CFR 50.48 and 50.49, which establish fire protection and environmental qualification requirements, respectively, and the guidance of Regulatory Guide 1.97, which establishes design and qualification requirements for instrumentation required for monitoring numerous plant conditions and variables for accident and post-accident conditions.

Consider adding Acceptance Criteria and Review Procedures for review of CDM and related ITAAC for plant systems.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25975/CODE OF FED. REGS 10CFR52; 26007/CODE OF FED. REGS 10CFR50; 26008/RG 1.97; 26039/CODE OF FED. REGS 10CFR50

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**Integrated Impact Number:** 1541      **SRP Section Number:** 14.3.8

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of certified design material (CDM), including associated inspections, tests, analyses and acceptance criteria (ITAAC) for radiation protection and emergency preparedness.

10 CFR Parts 52.47 and 52.97 provide the Commission's requirements with regard to the content of applications for design certification, and the issuance of combined licenses, respectively. The ITAAC must be reviewed and verified to provide assurance that compliance with the ITAAC ensures that a plant, that references a certified design and/or applies for a combined license, will be constructed and operated in accordance with the design certification and/or the requirements of the license.

10 CFR 52.47 and 52.97 require, in part, that applications for design certification and/or a combined license include evidence that the application is in accordance with Commission's regulations. In addition to the Commission's regulations and guidance applicable to radiation protection, the EPA regulates offsite radiation exposure in accordance with 40 CFR 190.

10 CFR 52.97(b)(1) states, in part, that ITAAC be included for emergency planning. SECY 95-090 (including the attached draft Supplement 2 to NUREG-0654) provide the Commission with the staff's views with regard to emergency planning under the requirements of 10 CFR 52.

Consider adding Acceptance Criteria and Review Procedures for review of CDM and related ITAAC for radiation protection and emergency preparedness.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25976/CODE OF FED. REGS 10CFR52; 26009/CODE OF FED. REGS 40CFR190;  
26010/SECY 95-090

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**Integrated Impact Number:** 1542      **SRP Section Number:** 14.3.9

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of certified design material (CDM), including associated inspections, tests, analyses and acceptance criteria (ITAAC) for human factors engineering.

10 CFR Parts 52.47 and 52.97 provide the Commission's requirements with regard to the content

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of applications for design certification, and the issuance of combined licenses, respectively. The ITAAC must be reviewed and verified to provide assurance that compliance with the ITAAC ensures that a plant, that references a certified design and/or applies for a combined license, will be constructed and operated in accordance with the design certification and/or the requirements of the license.

10 CFR 52.47 and 52.97 require, in part, that applications for design certification and/or a combined license include evidence that the application is in accordance with Commissions regulations, including those in 10 CFR 50 as appropriate. 10 CFR 52.47(a)(1)(ii) requires the applicant to demonstrate compliance with the applicable TMI requirements of 10 CFR 50.34(f). Guidance for review of human factors engineering is contained in staff NUREG-0711.

Consider adding Acceptance Criteria and Review Procedures for review of CDM and related ITAAC for human factors engineering.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25977/CODE OF FED. REGS 10CFR52; 26011/CODE OF FED. REGS 10CFR50; 26012/RG 1.97; 26041/NUREG 0711

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**Integrated Impact Number:** 1543      **SRP Section Number:** 14.3.10

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of certified design material (CDM), including associated inspections, tests, analyses and acceptance criteria (ITAAC) for the initial test program and the Design Reliability Assurance Program (DRAP).

10 CFR Parts 52.47 and 52.97 provide the Commission's requirements with regard to the content of applications for design certification, and the issuance of combined licenses, respectively. The ITAAC must be reviewed and verified to provide assurance that compliance with the ITAAC ensures that a plant, that references a certified design and/or applies for a combined license, will be constructed and operated in accordance with the design certification and/or the requirements of the license.

10 CFR 52.47 and 52.97 require, in part, that applications for design certification and/or a combined license include evidence that the application is in accordance with Commissions regulations. Guidance acceptable to the staff for complying with the initial test program requirements is contained in Regulatory Guide 1.68.

Consider adding Acceptance Criteria and Review Procedures for review of CDM and related



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ITAAC for the initial test program and the DRAP.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

25978/CODE OF FED. REGS 10CFR52; 26013/RG 1.68

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**Integrated Impact Number:** 1544      **SRP Section Number:** 18.0

#### **Suggested Changes to the SRP Section:**

NRC guidance for the systematic, top-down evaluation of human factors engineering (HFE) was originally provided in NUREG-0700, Revision 0. This document provided a methodology for the review of existing control rooms. It recommended that for new control rooms and re-designs of existing control rooms, that additional analyses be conducted to optimize the allocation of functions to humans and machines and further examine advanced control system technologies. Appendix B to NUREG-0700, Revision 0, was provided as one source of guidance regarding these analyses.

The guidance of NUREG-0700, Revision 0, was updated in NUREG-0700, Revision 1, to reflect changes in HFE review concerns and HSI technologies. Thus, the HFE review process presented in SRP Chapter 18 should incorporate guidance from NUREG-0700, Revision 1.

NUREG-0700, Revision 1, contains important factors for NRC staff reviewers to consider during reviews. Consider incorporating relevant review considerations from NUREG-0700, Revision 1, into SRP Chapter 18 and defining appropriate interfaces with SRP Sections 13.1.1, 13.1.2-13.1.3, 13.2, and 13.5.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

26014/NUREG 0700

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**Integrated Impact Number:** 1545      **SRP Section Number:** 18.0

#### **Suggested Changes to the SRP Section:**

NUREG-0711, "Human Factors Engineering Program Review Model," published in July 1994, presents a model for performing design certification reviews of human factors engineering program elements. The NUREG contains important factors for NRC staff reviewers to consider during reviews.

NUREG-0711 addresses the integration of human factors engineering (HFE) in the design

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process and was originally developed to support NRC reviews of submittals for certification of new plant designs under 10 CFR Part 52. However, because it updates the guidance of Appendix B of NUREG-0700, Revision 0, it should be used for HFE reviews of plant designs licensed under both 10 CFR Part 50 and 10 CFR Part 52. Portions of NUREG-0711 should also be used, as appropriate, to support the NRC in its reviews of re-designs and upgrades of current control rooms. Thus, the HFE review process presented in SRP Chapter 18 should incorporate guidance from NUREG-0711.

Consider incorporating relevant review considerations from NUREG-0711, into SRP Chapter 18 and defining appropriate interfaces with SRP Sections 13.1.1, 13.1.2-13.1.3, 13.2, and 13.5.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

26015/NUREG 0711

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**Integrated Impact Number:** 1546      **SRP Section Number:** 17.1

#### **Suggested Changes to the SRP Section:**

Add Regulatory Guides 1.36, 1.54 and 1.152 as guidance for QA of non-metallic thermal insulation, protective coatings and programmable digital computer system software in safety related systems. Addition of these guides will make regulatory guidance cited in SRP 17.1 consistent with that in SRP 17.3.

Regulatory Guide 1.36 describes an acceptable method for implementing regulations with regard to the selection and use of nonmetallic thermal insulation to minimize any contamination that could promote stress-corrosion cracking in the stainless steel portions of the reactor coolant pressure boundary and other systems important to safety. This guide applies to light-water-cooled, reactors. The Reg. Guide endorses several material test standards to be used for insulation materials qualification tests. The Reg. Guide is cited in SRP Section 17.3 as programmatic QA guidance.

Regulatory Guide 1.54 describes an acceptable method of complying with the Commission's quality assurance requirements with regard to protective coatings applied to ferritic steels, aluminum, stainless steel, zinc-coated (galvanized) steel, concrete, or masonry surfaces of water-cooled nuclear power plants. The requirements and guidelines included in ANSI N101.4-1972, Quality Assurance for Protective Coatings Applied to Nuclear Facilities, for protective coatings applied to ferritic steels, aluminum, stainless steel, zinc-coated (galvanized) steel, concrete, or masonry surfaces of water-cooled nuclear power plants are generally acceptable and provide an adequate basis for complying with the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 subject to the modifications described in the

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regulatory positions. This RG is cited in SRP Section 17.3 as programmatic QA guidance.

Regulatory Guide 1.152 describes a method acceptable to the NRC staff for complying with the Commission's regulations for promoting high functional reliability for safety related systems using programmable digital computer systems in the operation of nuclear power plants. Cited regulations include Appendix B to 10 CFR Part 50, specifically Criterion III. This method is applicable to designing software, verifying software, implementing software, and validating computer systems. The requirements set forth in ANSI/IEEE-ANS-74.3.2-1982 establish a method acceptable to the NRC staff for designing software, verifying software, implementing software, and validating computer systems used in safety-related systems of nuclear power plants. This RG is cited in SRP Section 17.3 as programmatic QA guidance.

Consider adding RGs 1.36, 1.54 and 1.152 as guidance in SRP Section 17.1, consistent with applicable guidance cited in SRP 17.3.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

19812/RG 1.152; 19813/RG 1.36; 19814/RG 1.54

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**Integrated Impact Number:** 1547      **SRP Section Number:** 17.2

#### **Suggested Changes to the SRP Section:**

Add Regulatory Guides 4.15 and 7.10 to the references list as guidance applicable to QA commitments, consistent with applicable cited Regulatory Guides in SRP 17.3.

RG 4.15 provides guidance on radiological monitoring required by the NRC. This guidance does not identify separately the activities that are within the scope of Appendix B to 10 CFR Part 50. However, this guidance is intended to be consistent with the requirements of Appendixes A and B to 10 CFR Part 50 in that quality assurance requirements should be consistent with the importance of the activity.

RG 7.10 provides persons subject to the QA requirements of Part 71 with information on the essential elements needed to develop, establish, and maintain a quality assurance program acceptable to the NRC staff for packages to transport radioactive materials.

Both of these RGs are listed as references in SRP 17.3

Consider adding RGs 4.15 and 7.10 as guidance applicable to QA commitments in SRP 17.2 for consistency with the guidance cited in SRP 17.3.

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#### **Potential Impacts/Documents supporting the Suggested Changes:**

19931/RG 7.10; 26021/RG 4.15

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**Integrated Impact Number:** 1548      **SRP Section Number:** 8.1

#### **Suggested Changes to the SRP Section:**

Augment SRP Section 8.1 with the staff interpretation of GDC 17.

In connection with a loss of offsite power event at Arkansas Nuclear One, the Power Systems Branch prepared a table detailing the staff interpretation of the requirements of GDC 17 which is used in licensing reviews.

Consider adding the staff interpretation of the requirements of GDC 17 to Section 8.1.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

26022/CODE OF FED. REGS 10CFR50; 26023/NRC MEMORANDUM 12/15/80, from F. Rosa to C. Michelson

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**Integrated Impact Number:** 1549      **SRP Section Number:** 1.8

#### **Suggested Changes to the SRP Section:**

Modify Acceptance Criteria to include reference to requirements for a description, analysis and evaluation of the interfaces between the submitted design and the balance of the nuclear power plant.

Appendix O of 10 CFR 50 requires the submittal of a standard design for a nuclear power reactor to include a description, analysis and evaluation of the interfaces between the submitted design and the balance of the nuclear power plant.

10 CFR 52.47(a)(1) and Appendix O to Part 52 require an application for design certification to contain the interface requirements to be met by those portions of the plant for which the application does not seek certification. These requirements are to be sufficiently detailed to allow completion of the final safety analysis and design-specific probabilistic risk assessments. 10 CFR 52.47 also requires that the application propose and justify those inspections, tests, analyses, and acceptance criteria (ITAAC) which are necessary and sufficient to provide reasonable assurance that, if the tests, inspections and analyses are performed and the acceptance criteria met, a plant which references the design is built and will operate in accordance with the

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design certification. Part 52 supersedes the guidance of Appendix A to Regulatory Guide 1.70.

The staff's technical review of ALWR applications for certification of their standard designs was performed in accordance with Commission guidance and the requirements of 10 CFR 52.47. The staff's evaluation of the technical information required by 52.47(a)(1) was performed in accordance with the SRP and is set forth throughout the evolutionary plant SERs.

Consider modifying Acceptance Criteria to include reference to requirements for a description, analysis and evaluation of the interfaces between the submitted design and the balance of the nuclear power plant.

#### **Potential Impacts/Documents supporting the Suggested Changes:**

26030/CODE OF FED. REGS 10CFR50; 26031/CODE OF FED. REGS 10CFR52;  
26032/DRAFT SER CE80 CH 1; 26033/DRAFT FINAL SER ABWR CH 1; 26034/CODE OF  
FED. REGS 10CFR52

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**Integrated Impact Number:** 1550      **SRP Section Number:** 14.3.11

#### **Suggested Changes to the SRP Section:**

Develop Acceptance Criteria and Review Procedures for review of certified design material (CDM), including associated inspections, tests, analyses and acceptance criteria (ITAAC) for containment systems.

10 CFR Parts 52.47 and 52.97 provide the Commission's requirements with regard to the content of applications for design certification, and the issuance of combined licenses, respectively. The ITAAC must be reviewed and verified to provide assurance that compliance with the ITAAC ensures that a plant, that references a certified design and/or applies for a combined license, will be constructed and operated in accordance with the design certification and/or the requirements of the license.

For design certification applications, the application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted.

Consider adding Acceptance Criteria and Review Procedures for review of CDM and related ITAAC for containment systems.

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**Potential Impacts/Documents supporting the Suggested Changes:**

26040/CODE OF FED. REGS 10CFR52

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