APPENDIX II

Potential Impacts

Change Basis Documents Referenced in Appendix I

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Potential Impact Number: 74 **SRP Section Number:** 6.3

Document: RG 1.157 **Revision:** 0
Title: BEST-ESTIMATE CALCULATIONS OF EMERGENCY CORE COOLING SYSTEM PERFORMANCE

Summary:

Provides guidance on how to use best-estimate calculations to satisfy 10CFR50.46 requirements involving evaluation models.

Potential Impact Number: 76 **SRP Section Number:** 6.3

Document: RG 1.154 **Revision:** 0
Title: FORMAT AND CONTENT OF PLANT-SPECIFIC PRESSURIZED THERMAL SHOCK SAFETY ANALYSIS REPORTS FOR PRESSURIZED WATER REACTORS

Summary:

The ECCS is considered to be a system that is pertinent to Pressurized Thermal Shock (PTS).

Plant Modifications that result in the increased temperature of emergency cooling water and emergency feedwater should be evaluated for pressurized thermal shock (PTS), as well as for non-PTS events.

Potential Impact Number: 84 **SRP Section Number:** 9.5.4

Document: RG 1.137 **Revision:** 1
Title: FUEL-OIL SYSTEMS FOR STANDBY DIESEL GENERATORS

Summary:

The requirements for the design of fuel-oil systems for diesel generators that provide standby electrical power for a nuclear power plant that are included in ANSI

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N195-1976, "Fuel Oil Systems for Standby Diesel-Generators," provide a method acceptable to the NRC staff for complying with the pertinent requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50, subject to the qualification contained in Reg. Guide 1.137.

For proper operation of the standby diesel generators, it is necessary to ensure the proper quality of the fuel-oil. Appendix B to ANSI N195-1976 addresses the recommended fuel-oil practices. Although not a mandatory part of the standard, the staff believes Appendix B can serve as an acceptable basis for a program to maintain the quality of fuel-oil, as supplemented by regulatory position 2 of the guide.

ANSI N195-1976 should be used in conjunction with Reg. Guide 1.28 for design, construction and maintenance of the fuel oil system.

This PI supercedes the following PIs: 85, 87, 88, 89, 92, 93, 94, 96, 98, 100, 102, 1573, 1575, 1576, 1578, 1579, 1580, 1583, 1584, 1590, 1592, 1594, 1597, and 1598.

Potential Impact Number: 99 **SRP Section Number:** 6.3

Document: RG 1.82 **Revision:** 1
Title: WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT

Summary:

Description for how to evaluate ECCS sump performance.

Potential Impact Number: 183 **SRP Section Number:** 6.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix K

Summary:

On September 16, 1988 the NRC amended the requirements of 10 CFR 50 Appendix K, "ECCS Evaluation Models," to permit licensees or applicants to use either an

acceptable evaluation model or Appendix K methods. Appendix K contains the required and acceptable features of evaluation models.

Potential Impact Number: 193 **SRP Section Number:** 6.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.46 (a)(1)

Summary:

On September 16, 1988 the NRC amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation models." 10 CFR 50.46 (a)(1) now permits licensees or applicants to use either an acceptable evaluation model or Appendix K.

The ECCS must be designed such that its calculated cooling performance following postulated loss-of-coolant accidents is calculated in accordance with an acceptable evaluation model and must be calculated with a number of different LOCAs.

Potential Impact Number: 195 **SRP Section Number:** 6.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.55a, (b)(2)(iv)

Summary:

Pressure retaining welds in ASME Code Class 2 piping in the emergency core cooling system shall be examined. This is a Section XI issue.

Potential Impact Number: 270 **SRP Section Number:** 9.2.6

Document: RG 1.143 **Revision:** 1

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Title: DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

Summary:

States condensate storage tanks should have provisions to monitor liquid levels and handle overflows.

Potential Impact Number: 271 **SRP Section Number:** 9.2.6

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

C.3.2.2: The capability of all systems and components necessary to provide core cooling and decay heat removal following a station blackout should be determined, including station battery capacity, condensate storage tank capacity, compressed air capacity, and instrumentation and control requirements.

C.3.3.2: If the capacity of the condensate storage tank is not sufficient to remove decay heat for the station blackout duration recommended in Regulatory Position 3.1, a system meeting the requirements of Regulatory Position 3.5 to resupply the tank from an alternative water source is an acceptable means to increase its capacity provided any power source necessary to provide additional water is independent of both the offsite and the blacked-out unit's onsite emergency ac power systems.

C.3.5: Appendices A and B provide guidance on quality assurance (QA) activities and specifications respectively for non-safety-related equipment used to meet the requirements of 50.63 and not already covered by existing QA requirements in Appendix B or R of Part 50. This QA guidance is applicable to non-safety systems and equipment for meeting the requirements of 50.63 of 10 CFR Part 50.

Potential Impact Number: 307 **SRP Section Number:** 5.3.1

Document: CODE OF FED. REGS 10CFR50

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Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.60

Summary:

LWRs must meet the fracture toughness and material surveillance requirements for reactor coolant pressure boundary set forth in 10CFR50 Appendix G and H, except proposed alternatives to the requirements in Appendices G and H may be used when an exemption is granted by the NRC under Section 50.12.

Potential Impact Number: 349 **SRP Section Number:** 5.3.1

Document: RG 1.65 **Revision:** 0
Title: MATERIALS AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS

Summary:

Materials specifications for reactor-vessel closure studs are defined. Regulatory Guide 1.65, Position C.1.b.(2) references requirements of 10 CFR 50 Appendix G, paragraph IV.A.4. As a result of amendments to Appendix G since the issuance of the Regulatory Guide, paragraph IV.A.4 is no longer an appropriate reference with regard to the Regulatory Guide and SRP Section 5.3.1.

Potential Impact Number: 401 **SRP Section Number:** 5.3.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix G

Summary:

Reactor vessel beltline materials must have Charpy upper-shelf energy of no less than 75 ft-lb initially and must maintain upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of upper-shelf energy will

provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code.

Potential Impact Number: 404 **SRP Section Number:** 5.3.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix H

Summary:

Fracture toughness surveillance program and test-results report are described for the reactor vessel belt-line region.

Potential Impact Number: 428 **SRP Section Number:** 5.3.1

Document: RG 1.99 **Revision:** 2
Title: RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS

Summary:

Procedures are defined that are acceptable for calculating neutron irradiation embrittlement for pressure vessels.

Potential Impact Number: 445 **SRP Section Number:** 5.3.1

Document: RG 1.37 **Revision:** 0
Title: QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER COOLED NUCLEAR POWER PLANTS

Summary:

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This guide describes an acceptable method of complying with the Commission's regulations with regard to quality assurance requirements for on-site cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water-cooled nuclear power plant fluid systems. The requirements and recommendations for on-site cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water-cooled nuclear power plant fluid systems that are included in ANSI N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of 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 regulatory positions (which include application of the standard to the operations phase).

Potential Impact Number: 448 **SRP Section Number:** 5.3.1

Document: RG 1.44 **Revision:** 0
Title: CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL

Summary:

This guide describes acceptable methods of implementing the Commission's requirements/regulations with regard to control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking. This guide applies to light-water-cooled reactors. Unstabilized, austenitic stainless steel of the AISI Type 3XX series used for components that are part of (1) the reactor coolant pressure boundary, (2) systems required for reactor shutdown, (3) systems required for emergency core cooling, and (4) reactor vessel internals that are relied upon to permit adequate core cooling for any mode of normal operation or under credible postulated accident conditions should meet the criteria provided in the regulatory positions.

Potential Impact Number: 463 **SRP Section Number:** 5.3.1

Document: RG 1.154 **Revision:** 0
Title: FORMAT AND CONTENT OF PLANT-SPECIFIC PRESSURIZED THERMAL SHOCK SAFETY ANALYSIS REPORTS FOR PRESSURIZED WATER REACTORS

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Summary:

Describes format and content for plant specific pressurized thermal shock safety analysis and acceptance criteria.

Potential Impact Number: 983 **SRP Section Number:** 5.3.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

There have been numerous reports of failed bolts or closure studs. Many of these bolts are related to the primary pressure boundary. Degradation or failure of these studs and bolts reduce the integrity of the reactor coolant pressure boundary. Visual examination is currently the only reliable detection method. A major accident such as a LOCA could occur due to undetected bolting failure. This issue has a high priority ranking.

Potential Impact Number: 986 **SRP Section Number:** 3.11

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

There have been numerous reports of failed bolts or closure studs. Many of these bolts are related to the primary pressure boundary. Degradation or failure of these studs and bolts reduce the integrity of the reactor coolant pressure boundary. Visual examination is currently the only reliable detection method. A major accident such as a lock could occur due to undetected bolting failure. This issue has a high priority ranking.

Potential Impact Number: 987 **SRP Section Number:** 5.2.4

Document: NUREG 0933

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Title: A Prioritization of Generic Safety Issues

Summary:

There have been numerous reports of failed bolts or closure studs. Many of these bolts are related to the primary pressure boundary. Degradation or failure of these studs and bolts reduce the integrity of the reactor coolant pressure boundary.

In resolving the issue, the staff took into the consideration previous actions taken by the NRC to address the concerns related to threaded fasteners: Bulletin Nos. 82-02, 87-02, and 89-02; Information Notice Nos. 86-25, 89-22, 89-56 and 89-70; and Generic Letter Nos. 87-02 and 88-05.

The staff's regulatory analysis, NUREG-1445, proved to be inconclusive regarding a mandatory program on safety-related bolting for operating plants. The staff's technical findings were documented in NUREG-1339 which endorsed the recommendations of independent studies performed by the industry Joint Task Group on Bolting. This group was set up by AIF, EPRI, and the Materials Properties Council and its studies resulted in EPRI NP-5769, "Good Bolting Practices," and three EPRI video training tapes on "Pressure Boundary Bolting Problems."

The staff concluded that leakage of bolted pressure joints was possible, but catastrophic RCPB joint failure that could lead to significant accident sequences was highly unlikely. This conclusion was based on: (1) operating experience; (2) actions taken through bulletins, generic letters, and information notices; and (3) proposed industry actions. Generic Letter No. 91-17 was issued to licensees to: (1) implement the industry bolting integrity program, as presented in the EPRI report and video tapes; and (2) continue actions in accordance with commitments made in response to NRC generic letters and bulletins. Thus, this issue was RESOLVED and no new requirements were established. However, in order to improve the review of future plants and significant modifications to operating plants, the staff recommended that a new SRP Section be developed to codify existing guidance and industry recommendations.

Potential Impact Number: 995 **SRP Section Number:** 3.9.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

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Potential Impacts

There have been numerous reports of failed bolts or closure studs. Many of these bolts are related to the primary pressure boundary. Degradation or failure of these studs and bolts reduce the integrity of the reactor coolant pressure boundary. Visual examination is currently the only reliable detection method. A major accident such as a lock could occur due to undetected bolting failure. This issue has a high priority ranking.

Potential Impact Number: 999 **SRP Section Number:** 3.10

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

There have been numerous reports of failed bolts or closure studs. Many of these bolts are related to the primary pressure boundary. Degradation or failure of these studs and bolts reduce the integrity of the reactor coolant pressure boundary. Visual examination is currently the only reliable detection method. A major accident such as a lock could occur due to undetected bolting failure. This issue has a high priority ranking.

Potential Impact Number: 1000 **SRP Section Number:** 4.5.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

There have been numerous reports of failed bolts or closure studs. Many of these bolts are related to the primary pressure boundary. Degradation or failure of these studs and bolts reduce the integrity of the reactor coolant pressure boundary. Visual examination is currently the only reliable detection method. A major accident such as a LOCA could occur due to undetected bolting failure. This issue has a high priority ranking.

Potential Impact Number: 1006 **SRP Section Number:** 5.2.3

Document: NUREG 0933

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Title: A Prioritization of Generic Safety Issues

Summary:

There have been numerous reports of failed bolts or closure studs. Many of these bolts are related to the primary pressure boundary. Degradation or failure of these studs and bolts reduce the integrity of the reactor coolant pressure boundary. Visual examination is currently the only reliable detection method. A major accident such as a LOCA could occur due to undetected bolting failure.

This issue was resolved and no new requirements were established. However, in order to improve the review of future plants and significant modifications to operating plants, the staff recommended that a new SRP Section be developed to codify existing guidance and industry recommendations.

Potential Impact Number: 1009 **SRP Section Number:** 6.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

There have been numerous reports of failed bolts or closure studs. Many of these bolts are related to the primary pressure boundary. Degradation or failure of these studs and bolts reduce the integrity of the reactor coolant pressure boundary. Visual examination is currently the only reliable detection method. A major accident such as a lock could occur due to undetected bolting failure. This issue has a high priority ranking.

Potential Impact Number: 1081 **SRP Section Number:** 4.5.1

Document: RG 1.37 **Revision:** 0
Title: QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER COOLED NUCLEAR POWER PLANTS

Summary:

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Potential Impacts

This guide describes an acceptable method of complying with the Commission's regulations with regard to quality assurance requirements for on-site cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water-cooled nuclear power plant fluid systems. The requirements and recommendations for on-site cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water- cooled nuclear power plant fluid systems that are included in ANSI N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of 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 regulatory positions (which include application of the standard to the operations phase).

Potential Impact Number: 1083 **SRP Section Number:** 4.5.1

Document: RG 1.31 **Revision:** 3
Title: CONTROL OF FERRITE CONTENT IN STAINLESS STEEL WELD METAL

Summary:

This guide describes a method acceptable to the NRC staff for implementing NRC requirements/regulations with regard to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems in light-water-cooled nuclear power plants.

Potential Impact Number: 1086 **SRP Section Number:** 4.5.1

Document: RG 1.44 **Revision:** 0
Title: CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL

Summary:

This guide describes acceptable methods of implementing the Commission's requirements/regulations with regard to control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking. This guide applies to light-water-cooled reactors. Unstabilized, austenitic stainless steel of the AISI Type 3XX series used for components that are part of (1) the reactor coolant

pressure boundary, (2) systems required for reactor shutdown, (3) systems required for emergency core cooling, and (4) reactor vessel internals that are relied upon to permit adequate core cooling for any mode of normal operation or under credible postulated accident conditions should meet the criteria provided in the regulatory positions.

Potential Impact Number: 1109 **SRP Section Number:** 9.2.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue addresses the problem of flow blockage by Asiatic Clams (Corbicula) in redundant safety-related cooling water systems. This issue has been combined and evaluated with Issue 51 'proposed requirements for improving the reliability of open cycle service water systems.'

Potential Impact Number: 1200 **SRP Section Number:** 5.3.1

Document: RG 1.31 **Revision:** 3
Title: CONTROL OF FERRITE CONTENT IN STAINLESS STEEL WELD METAL

Summary:

This guide describes a method acceptable to the NRC staff for implementing NRC requirements/regulations with regard to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems in light-water-cooled nuclear power plants.

Potential Impact Number: 1250 **SRP Section Number:** 5.2.2

Document: RG 1.153 **Revision:** 0
Title: CRITERIA FOR POWER, INSTRUMENTATION, AND CONTROL

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Summary:

IEEE STD 603-1980 is being implemented for construction permits after November 1985. IEEE STD 603-1980 may be used for current reactors, and is broader in scope than IEEE STD 279-1971. Applies to power instrumentation and control of safety systems.

Potential Impact Number: 1269 **SRP Section Number:** 6.3

Document: NRC GENERIC LETTER 85-22
Title: POTENTIAL FOR LOSS OF POST-LOCA RECIRCULATION CAPABILITY DUE TO INSULATION DEBRIS BLOCKAGE

Summary:

The 50% screen blockage assumption identified for ECCS recirculation sumps should be replaced with a plant specific assessment of the quantity of debris that will accumulate at the sump screens. See the regulatory requirement in Rev. 1 of R.G. 1.82.

Potential Impact Number: 1284 **SRP Section Number:** 6.3

Document: NRC BULLETIN 88-08
Title: THERMAL STRESSES IN PIPING CONNECTED TO REACTOR COOLANT SYSTEMS

Summary:

Cracks in ECCS piping where it connects to the RCS. These cracks were discovered in sections of pipe that could not be isolated.

Potential Impact Number: 1288 **SRP Section Number:** 6.3

Document: NRC BULLETIN 88-04

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Title: POTENTIAL SAFETY-RELATED PUMP LOSS

Summary:

This Bulletin addressed the potential for dead-heading of ECCS pumps that have a common mini-flow line. In addition, questions regarding the installed mini-flow line capacity were also addressed.

Potential Impact Number: 1326 **SRP Section Number:** 6.3

Document: NRC BULLETIN 79-17
Title: PIPE CRACKS IN STAGNANT BORATED WATER SYSTEMS AT PWR PLANTS

Summary:

Stainless steel pipe cracks occurring in stagnant oxygenated borated water. Inspect ECCS piping for problem.

Potential Impact Number: 1328 **SRP Section Number:** 6.3

Document: NRC BULLETIN 80-18
Title: MAINTENANCE OF ADEQUATE MINIMUM FLOW THRU CENTRIFUGAL CHARGING PUMPS FOLLOWING SECONDARY SIDE HIGH ENERGY LINE RUPTURE

Summary:

This Bulletin addressed a concern regarding the potential loss of centrifugal charging pumps if the mini-flow valves are shut and RCS pressure increases above the CCP shutoff head.

Potential Impact Number: 1331 **SRP Section Number:** 6.3

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Document: NRC BULLETIN 86-03
Title: POTENTIAL FAILURE OF MULTIPLE ECCS PUMPS DUE TO SINGLE FAILURE OF AIR-OPERATED VALVE IN MINIMUM FLOW RECIRCULATION LINE

Summary:

This Bulletin addressed a concern regarding single failure of air-operated ECCS pump mini-flow isolation valves that could result in loss of the ECCS pumps.

Potential Impact Number: 1351 **SRP Section Number:** 9.3.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue concerns the reliability of the compressed air system. Air systems actuate or control safety related systems. However air systems are not safety grade systems at most plants. Some systems become degraded or fail on loss of control air.

The staff issued Information Notice No. 87-28, Supplement 1, to inform OLs and CPs of the publication of NUREG-1275, Volume 2. This report 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. The report also indicated that anticipated transient and system recovery procedures were frequently inadequate and that operators were not well-trained for coping with loss of instrument air conditions.

Generic Letter 88-14 was issued to request that each licensee/applicant review NUREG-1275, Volume 2, and perform a design and operations verification of instrument air systems. In addition, all licensees/applicants were requested to provide a discussion of their program for maintaining proper instrument air quality. Thus, this issue was RESOLVED and requirements were established.

Potential Impact Number: 1424 **SRP Section Number:** 9.2.1

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Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue concerns the effects on safety of open cycle water system fouling by mussels, clams, oysters, etc. This issue was given a medium priority ranking. Recommended solution was the implementation of a baseline fouling program. This issue is resolved and new requirements were established.

Potential Impact Number: 1430 **SRP Section Number:** 9.2.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue concerns the problem of flow blockage by blue mussels (Mytilus Edilus) in the reactor building closed cooling water system at Pilgrim. This issue has been combined and evaluated with Issue 51.

Potential Impact Number: 1463 **SRP Section Number:** 3.9.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Issue determined by NRC to have generic implications for all PWRs. Issue resolved in Jan. 1981 with the publication of NUREG-0609. MPA-10 established for operating PWRs. Generic letter 84-04 also issued.

Potential Impact Number: 1465 **SRP Section Number:** 3.9.5

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Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

In a postulated event of reactor coolant pipe rupture at the vessel nozzle, asymmetric loss-of-cooling-accident (LOCA) loading could result from forces induced on the reactor internals by transient differential pressures across the core barrel and by forces on the vessel due to transient differential pressures in the reactor cavity. This USI was resolved in January 1981, with the publication of NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems.

Potential Impact Number: 1474 **SRP Section Number:** 5.4.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Issues A-3, A-4 and A-5 were resolved and requirements were established as stated in SECY-88-272 (Sept. 1988). The current regulations (CFRs) were determined sufficient to ensure that licensees are implementing steam generator tube integrity programs which are adequate. The NRC staff may impose additional requirements as necessary.

Potential Impact Number: 1480 **SRP Section Number:** 5.4.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Issues A-3, A-4, and A-5 were resolved and requirements were established as stated in SECY-88-272 (Sept. 1988). The current regulations (CFRs) were determined sufficient to ensure that licensees are implementing steam generator tube integrity programs which are adequate. The NRC staff may impose additional requirements as necessary.

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Potential Impact Number: 1486 **SRP Section Number:** 5.4.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Issues A-3, A-4 and A-5 were resolved and requirements were established as stated in SECY-88-272 (Sept. 1988). The current regulations (CFRs) were determined sufficient to ensure that licensees are implementing steam generator tube integrity programs which are adequate. The NRC staff may impose additional requirements as necessary.

Potential Impact Number: 1499 **SRP Section Number:** 15.8

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

The NRC staff's technical findings on this issue were published in Volume 4 of NUREG-0460. This GSI was resolved on June 26, 1984 with the publication final rule in Federal Register Vol. 49, No. 124 pp. 26036-26045.

Potential Impact Number: 1503 **SRP Section Number:** 4.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

The NRC staff's technical findings on this issue were published in Volume 4 of NUREG-0460. This GSI was resolved on June 26, 1984 with the publication final rule in Federal Register Vol. 49, No. 124, pp. 26036-26045.

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Potential Impact Number: 1540 **SRP Section Number:** 5.3.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Prevention of reactor vessel failure requires maintaining the fracture toughness of the vessel materials above brittle failure levels. Plants with long service times have reduced fracture toughness levels and safety margins. GSI resolved with the issuance of NUREG-0744, Rev. 1 and Generic Letter 82-86 (Note 3a).

Potential Impact Number: 1545 **SRP Section Number:** 5.3.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Prevention of reactor vessel failure requires maintaining the fracture toughness of the vessel materials above brittle failure levels. Plants with long service times have reduced fracture toughness levels and safety margins. GSI resolved with the issuance of NUREG-0744, Rev. 1 and Generic Letter 82-86 (Note 3a).

Potential Impact Number: 1637 **SRP Section Number:** 9.5.5

Document: NRC GENERIC LETTER 89-13
Title: SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

Summary:

Heat exchanger performance must be monitored for degradation. In lieu of this monitoring frequent regular maintenance can be performed.

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Potential Impact Number: 1647 **SRP Section Number:** 9.2.5

Document: RG 1.127 **Revision:** 1
Title: INSPECTION OF WATER-CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR POWER PLANTS

Summary:

This guide describes a basis acceptable to the NRC staff for developing an appropriate inservice inspection and surveillance program for dams, slopes, canals, and other water-control structures associated with emergency cooling water systems or flood protection of nuclear power plants.

Potential Impact Number: 1706 **SRP Section Number:** 10.4.7

Document: NRC BULLETIN 87-01
Title: THINNING OF PIPE WALLS IN NUCLEAR POWER PLANTS

Summary:

Required documentation of programs for monitoring the thickness of pipes in condensate and feedwater systems.

Potential Impact Number: 1711 **SRP Section Number:** 10.4.7

Document: NRC GENERIC LETTER 81-11
Title: BWR FEEDWATER NOZZLE CRACKING

Summary:

States some crack growth analysis requirements for BWRs.

Potential Impact Number: 1720 **SRP Section Number:** 10.4.7

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Document: NRC GENERIC LETTER 89-08
Title: EROSION/CORROSION-INDUCED PIPE WALL THINNING

Summary:

Provides information related to pipe thinning in feedwater and condensate piping associated with erosion/corrosion.

Potential Impact Number: 1723 **SRP Section Number:** 10.4.7

Document: NRC GENERIC LETTER 89-19
Title: REQUEST FOR ACTION RELATED TO RESOLUTION OF UNRESOLVED SAFETY ISSUE A-47 'SAFETY IMPLICATION OF CONTROL SYSTEMS IN LWR NUCLEAR POWER PLANTS'

Summary:

As a result of the technical resolution of USI A-47, Safety implications of Control Systems in, LWR Nuclear Power Plants, the NRC has concluded that protection should be provided for certain control system failures and that selected emergency procedures should be modified to assure that plant transients resulting from control system failures do not compromise public safety.

Potential Impact Number: 1820 **SRP Section Number:** 4.5.2

Document: NRC BULLETIN 87-02
Title: FASTENER TESTING TO DETERMINE CONFORMANCE WITH APPLICABLE MATERIAL SPECIFICATIONS

Summary:

Licensees are requested to provide information on receipt, inspection, and internal control procedures for fasteners and the results of testing.

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Potential Impact Number: 1836 **SRP Section Number:** 4.5.2

Document: RG 1.31 **Revision:** 3
Title: CONTROL OF FERRITE CONTENT IN STAINLESS STEEL WELD METAL

Summary:

This guide describes a method acceptable to the NRC staff for implementing NRC requirements/regulations with regard to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems in light-water-cooled nuclear power plants.

Potential Impact Number: 1839 **SRP Section Number:** 4.5.2

Document: RG 1.37 **Revision:** 0
Title: QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER COOLED NUCLEAR POWER PLANTS

Summary:

This guide describes an acceptable method of complying with the Commission's regulations with regard to quality assurance requirements for on-site cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water-cooled nuclear power plant fluid systems. The requirements and recommendations for on-site cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water- cooled nuclear power plant fluid systems that are included in ANSI N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of 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 regulatory positions (which include application of the standard to the operations phase).

Potential Impact Number: 1841 **SRP Section Number:** 5.2.3

Document: NRC BULLETIN 82-02

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Title: DEGRADATION OF THREADED FASTENERS IN THE REACTOR COOLANT PRESSURE BOUNDARY OF PWR PLANTS

Summary:

Notifies/requests actions on incidents of severe degradation of threaded fasteners in closures in the RCPB.

Potential Impact Number: 1842 **SRP Section Number:** 4.5.2

Document: RG 1.44 **Revision:** 0
Title: CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL

Summary:

This guide describes acceptable methods of implementing the Commission's requirements/regulations with regard to control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking. This guide applies to light-water-cooled reactors. Unstabilized, austenitic stainless steel of the AISI Type 3XX series used for components that are part of (1) the reactor coolant pressure boundary, (2) systems required for reactor shutdown, (3) systems required for emergency core cooling, and (4) reactor vessel internals that are relied upon to permit adequate core cooling for any mode of normal operation or under credible postulated accident conditions should meet the criteria provided in the regulatory positions.

Potential Impact Number: 1855 **SRP Section Number:** 5.2.3

Document: NRC GENERIC LETTER 88-01
Title: NRC POSITION ON IGSCC IN BWR AUSTENITIC STAINLESS STEEL PIPING

Summary:

The purpose of this Generic Letter is to seek information regarding implementation of new staff positions covering technical areas related to intergranular stress corrosion cracking in BWR austenitic stainless steel piping. Attachment A to the letter defines staff postions on numerous subjects relating to SRP 5.2.3, including: Staff Positions on

Materials; Staff Position on Processes: Staff Position on Water Chemistry; and Staff Position on Weld Overlay Reinforcement.

The technical bases for these postions are detailed in NUREG-0313, Rev. 2, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." This revision to NUREG-0313 was a major task in the staff long range plan to deal with BWR pipe cracking that was presented to the Commission in SECY 84-301. This revision includes the relevant recommedations of the Piping Review Committee Task Group on Pipe Cracking issued as NUREG-1061, Vol. 1, "Report of USNRC Piping Review Committee," and consideration of public comments on that document. NUREG-0313, Rev. 2, describes the technical bases for the staff positions on materials, processes, and primary coolant chemistry to minimize and control IGSCC problems.

Potential Impact Number: 1971 **SRP Section Number:** 3.4.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

The conclusion section of the discussion of Unresolved Safety Issue A-17 documents a staff recommendation to revise the SRP to include guidance regarding protection from internal flooding and water intrusion events.

Potential Impact Number: 2004 **SRP Section Number:** 5.2.3

Document: RG 1.31 **Revision:** 3
Title: CONTROL OF FERRITE CONTENT IN STAINLESS STEEL WELD METAL

Summary:

This guide describes a method acceptable to the NRC staff for implementing NRC requirements/regulations with regard to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems in light-water-cooled nuclear power plants.

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Potential Impact Number: 2009 **SRP Section Number:** 5.2.3

Document: RG 1.44 **Revision:** 0
Title: CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL

Summary:

This guide describes acceptable methods of implementing the Commission's requirements/regulations with regard to control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking. This guide applies to light-water-cooled reactors. Unstabilized, austenitic stainless steel of the AISI Type 3XX series used for components that are part of (1) the reactor coolant pressure boundary, (2) systems required for reactor shutdown, (3) systems required for emergency core cooling, and (4) reactor vessel internals that are relied upon to permit adequate core cooling for any mode of normal operation or under credible postulated accident conditions should meet the criteria provided in the regulatory positions.

Potential Impact Number: 2029 **SRP Section Number:** 9.3.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.62(c)(4)

Summary:

Each BWR must have a standby liquid control system capable of injecting a borated water solution equivalent to 86 GPM of 13 weight percent sodium pentaborate solution at natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel.

Automatic initiation is required for all facilities granted a construction permit after July 26, 1984, and to facilities granted a construction permit prior to July 26, 1984, that have been designed to include this feature. The SLCS must perform its function in a reliable manner.

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Potential Impact Number: 2073 **SRP Section Number:** 9.3.5

Document: NRC GENERIC LETTER 85-03
Title: CLARIFICATION OF EQUIVALENT CONTROL CAPACITY FOR STANDBY LIQUID CONTROL SYSTEMS

Summary:

Guidance is provided to interpret the requirements set forth in 10CFR50.62 (c)(4) regarding the boron injection capability of the SLCS.

Potential Impact Number: 2263 **SRP Section Number:** 9.2.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50, Appendix A, Criterion 5

Summary:

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Potential Impact Number: 2277 **SRP Section Number:** 9.2.1

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 10CFR52.47.(b), 10CFR52.79.(b)

Summary:

Service water intake structure and ultimate heat sink design do not have to be complete for a certification of a nuclear power plant design.

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Potential Impact Number: 2279 **SRP Section Number:** 9.2.1

Document: NRC POLICY STATEMENT 52 FR 34884
Title: NUCLEAR POWER PLANT STANDARDIZATION

Summary:

Features of the design which are dependent on the site, such a cooling water supply, are reviewed at the time of an actual application.

Potential Impact Number: 2288 **SRP Section Number:** 9.2.1

Document: RG 1.68 **Revision:** 2
Title: INITIAL TEST PROGRAMS FOR WATER-COOLED NUCLEAR POWER PLANTS

Summary:

Service water systems should be tested.

Potential Impact Number: 2289 **SRP Section Number:** 9.2.1

Document: RG 1.68.2 **Revision:** 1
Title: INITIAL STARTUP TEST PROGRAM TO DEMONSTRATE REMOTE SHUTDOWN CAPABILITY FOR WATER-COOLED NUCLEAR POWER PLANTS

Summary:

A heat transfer path to the ultimate heat sink should be demonstrated.

Potential Impact Number: 2331 **SRP Section Number:** 9.2.1

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Document: RG 5.65 **Revision:** 0
Title: VITAL AREA ACCESS CONTROLS, PROTECTION OF PHYSICAL SECURITY EQUIPMENT, AND KEY AND LOCK CONTROLS

Summary:

Intake structures present a complication for security.

Potential Impact Number: 2402 **SRP Section Number:** 5.3.1

Document: NRC GENERIC LETTER 82-26
Title: NUREG-0744 REV. 1; PRESSURE VESSEL MATERIAL FRACTURE TOUGHNESS

Summary:

To provide guidance to licensees who may be required to submit a fracture analysis to justify continued operation of a pressure vessel, the USNRC has developed an elastic-plastic fracture mechanics analytical procedure. This analytical methodology is presented in NUREG-0744, Revision 1 (Office of Management and Budget Clearance Number 3150-0011), a copy of which is enclosed with this letter.

This document presents one methodology which is acceptable to the USNRC staff for meeting the requirement for a fracture analysis with sufficient accuracy and reliability to demonstrate adequate margins for continued operation in accordance with Section V.C.3 of Appendix G to 10 CFR 50. Licensees, however, need not employ this specific fracture analysis methodology and may submit an analysis based on other techniques for NRC review and approval.

Potential Impact Number: 2412 **SRP Section Number:** 9.2.1

Document: NRC BULLETIN 81-03
Title: FLOW BLOCKAGE OF COOLING WATER TO SAFETY SYSTEM COMPONENTS BY CORBICULA SP. (ASIATIC CLAM) AND MYTILUS SP. (MUSSEL)

Summary:

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On September 3, 1980, Arkansas Nuclear One (ANO), Unit 2, was shut down after the NRC Resident Inspector discovered that Unit 2 had failed to meet the technical specification requirements for minimum service water flow rate through the containment cooling units (CCUs). After plant shutdown it was determined that the inadequate flow was due to extensive plugging of the CCUs by Asiatic clams (Corbicula species, a non-native fresh water bivalve mollusk). The licensee disassembled the service water piping at the coolers. Clams were found in the 3-inch diameter supply piping at the inlet to the CCUs and in the cooler inlet water boxes. The service water, which is taken from the Dardanelle Reservoir, is filtered before it is pumped through the system. The strainers on the service water pump discharges were examined and found to be intact. Since these strainers have a 3/16-inch mesh, much smaller than some of the shells found, it appears that clams had been growing in the system.

Potential Impact Number: 2520 **SRP Section Number:** 9.2.1

Document: NRC GENERIC LETTER 89-13
Title: SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

Summary:

This Generic Letter is being issued to require licensees and applicants to supply information about their respective service water systems to assure the NRC of such compliance and to confirm that the safety functions of their respective service water systems are being met. Main topic of this Generic Letter is service water system problems.

Potential Impact Number: 2521 **SRP Section Number:** 9.2.1

Document: NRC GENERIC LETTER 89-13 Sup 1
Title: SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

Summary:

Main topic is service water system problems.

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This supplement contains the questions and answers read into the transcripts during the workshops, except for the following changes. Questions received in the general, Action I, and Action II categories have been grouped according to topic. In addition, the NRC staff modified some answers after the workshops with the aim of furnishing additional guidance.

Potential Impact Number: 2546 **SRP Section Number:** 9.3.3

Document: RG 1.120 **Revision:** 1
Title: FIRE PROTECTION GUIDELINES FOR NUCLEAR POWER PLANTS

Summary:

Drainage system requirements with respect to fire protection are stated.

Of particular concern is adequate drainage of water used to extinguish fires and prevention of loss of non-water fire suppressants.

Potential Impact Number: 2655 **SRP Section Number:** 5.3.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.61

Summary:

Licensee shall submit projected values of RT [subscript] PTS of Reactor vessel beltline materials and other information related to fracture toughness.

Potential Impact Number: 2691 **SRP Section Number:** 5.3.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix G

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Summary:

Specifies minimum requirements for principal design criteria similar in design to those for which construction permits have been issued. APPENDIX G specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

Potential Impact Number: 2692 **SRP Section Number:** 5.3.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix H

Summary:

APPENDIX H -- REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM REQUIREMENTS
The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel

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Potential Impact Number: 2816 **SRP Section Number:** 11.3

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION
Locations: 10CFR20.1302, 10CFR20.1302, 10CFR20.1301

Summary:

Limits on doses to members of the public. A licensee may show compliance with the average annual dose limits by demonstrating that annual average concentration of

radioactive material released in gaseous and liquid effluents at the boundary of unrestricted areas do not exceed the values specified in Table 2 of Appendix B to 10 CFR 20.1001-20.2401. Adjustments are allowed by 10 CFR 20.1302 (c) based upon the physical and chemical characteristics of the effluents.

Potential Impact Number: 2817 **SRP Section Number:** 11.3

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION
Location: 10CFR20 (20.1-20.601) Appendix B

Summary:

Provides legal limits on average annual activity attributable to concentrations of radionuclides in effluents.

Potential Impact Number: 2822 **SRP Section Number:** 11.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.36a

Summary:

Requires technical specifications and operating procedures governing, controlling, and minimizing radioactivity released in effluents.

Potential Impact Number: 2837 **SRP Section Number:** 11.3

Document: RG 1.70 **Revision:** 3
Title: STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS

Summary:

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Any "realistic analysis" provided will help quantify the margins that are inherent in the design basis approach. A "realistic analysis" need not include a consequence assessment and may be limited to a presentation of assumptions that are more likely to be obtained than those used for purposes of design. The parameters and assumptions used for these analyses, as well as the results, should be presented in tabular form. Table 15-3 provides a representative list of these items. Table 15-4 summarizes additional items that should be provided when dealing with specific types of accidents. When possible, the summary tabulation should provide the necessary quantitative information. If, however, a particular assumption cannot be simply or clearly stated in the table, the table should reference a section or an appendix that adequately discusses the information. Judgment should be used in eliminating unnecessary parameters from the summary table or in adding parameters of significance that do not appear in Table 15-3 or 15-4.

Table 15-4, Additional Parameters and Information to be Provided or Referenced in the Summary Tabulation for Specific Design Basis Accidents, includes item 2. Waste Gas System Failure (Section 15.7.1)

Potential Impact Number: 2919 **SRP Section Number:** 5.3.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, IV Fluid Systems, Criterion 31

Summary:

The RCPB shall be designed to assure that the boundary materials behave in a non-brittle manner and so that the probability of rapid fracture propagation is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Potential Impact Number: 2920 **SRP Section Number:** 5.3.1

Document: CODE OF FED. REGS 10CFR50

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Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, IV Fluid Systems, Criterion 32

Summary:

Components that are part of the RCPB shall be designed to (1) permit inspection and testing to assess structure and leak-tight integrity and (2) assure an appropriate material surveillance program is in place for the reactor pressure vessel.

Potential Impact Number: 2990 **SRP Section Number:** 5.4.2.1

Document: NRC BULLETIN 89-01 **Revision:** 0 , 5/15/89
Title: FAILURE OF WESTINGHOUSE STEAM GENERATOR TUBE MECHANICAL PLUGS

Summary:

Identifies Westinghouse steam generator tube mechanical plugs made from Inconel 600 that may fail due to PWSCC.

Potential Impact Number: 3004 **SRP Section Number:** 5.4.2.1

Document: NRC BULLETIN 82-02
Title: DEGRADATION OF THREADED FASTENERS IN THE REACTOR COOLANT PRESSURE BOUNDARY OF PWR PLANTS

Summary:

Discusses the failure of steam generator manway studs. The failure was caused by stress corrosion cracking. Studs failed by the joint action of stud preload, material conditions and a corrosive environment generated by primary coolant leakage.

Potential Impact Number: 3051 **SRP Section Number:** 5.4.2.1

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Document: RG 1.31 **Revision:** 3
Title: CONTROL OF FERRITE CONTENT IN STAINLESS STEEL WELD METAL

Summary:

This guide describes a method acceptable to the NRC staff for implementing NRC requirements/regulations with regard to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems in light-water-cooled nuclear power plants.

This guide provides limits of 5-20% for ferrite content in austenitic stainless steel weld metal to prevent the formation of microfissures.

Potential Impact Number: 3056 **SRP Section Number:** 5.4.2.1

Document: RG 1.37 **Revision:** 0
Title: QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER COOLED NUCLEAR POWER PLANTS

Summary:

This guide describes an acceptable method of complying with the Commission's regulations with regard to quality assurance requirements for on-site cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water-cooled nuclear power plant fluid systems. The requirements and recommendations for on-site cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water- cooled nuclear power plant fluid systems that are included in ANSI N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of 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 regulatory positions (which include application of the standard to the operations phase).

Potential Impact Number: 3057 **SRP Section Number:** 5.4.2.1

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Document: NRC GENERIC LETTER 88-01
Title: NRC POSITION ON IGSCC IN BWR AUSTENITIC STAINLESS STEEL PIPING

Summary:

The purpose of this Generic Letter is to seek information regarding implementation of new staff positions covering technical areas related to intergranular stress corrosion cracking in BWR austenitic stainless steel piping. Attachment A to the letter defines staff postions on numerous subjects, including: Staff Positions on Materials; Staff Position on Processes: Staff Position on Water Chemistry; and Staff Position on Weld Overlay Reinforcement.

The technical bases for these postions are detailed in NUREG-0313, Rev. 2, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." This revision to NUREG-0313 was a major task in the staff long range plan to deal with BWR pipe cracking that was presented to the Commission in SECY 84-301. This revision includes the relevant recommedations of the Piping Review Committee Task Group on Pipe Cracking issued as NUREG-1061, Vol. 1, "Report of USNRC Piping Review Committee," and consideration of public comments on that document. NUREG-0313, Rev. 2, describes the technical bases for the staff positions on materials, processes, and primary coolant chemistry to minimize and control IGSCC problems.

Potential Impact Number: 3060 **SRP Section Number:** 5.4.2.1

Document: RG 1.44 **Revision:** 0
Title: CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL

Summary:

This guide describes acceptable methods of implementing the Commission's requirements/regulations with regard to control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking. This guide applies to light-water-cooled reactors. Unstabilized, austenitic stainless steel of the AISI Type 3XX series used for components that are part of (1) the reactor coolant pressure boundary, (2) systems required for reactor shutdown, (3) systems required for emergency core cooling, and (4) reactor vessel internals that are relied upon to permit adequate core cooling for any mode of normal operation or under credible postulated accident conditions should meet the criteria provided in the regulatory positions.

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Potential Impact Number: 3667 **SRP Section Number:** 9.5.1

Document: NRC BULLETIN 81-03
Title: FLOW BLOCKAGE OF COOLING WATER TO SAFETY SYSTEM COMPONENTS BY CORBICULA SP. (ASIATIC CLAM) AND MYTILUS SP. (MUSSEL)

Summary:

Licensees were requested to review fire water supplies and lines to determine whether biofouling could have degraded system performance.

Potential Impact Number: 3817 **SRP Section Number:** 9.5.1

Document: NRC GENERIC LETTER 81-12
Title: FIRE PROTECTION RULE (45 FR 76602, NOVEMBER 19, 1980)

Summary:

This generic letter requested responses from licensed plants with regard to compliance with the newly issued fire protection rule (10 CFR 50.48). Included with the letter is a staff position on safe shutdown capability.

Potential Impact Number: 3823 **SRP Section Number:** 9.5.1

Document: NRC GENERIC LETTER 83-33
Title: NRC POSITIONS ON CERTAIN REQUIREMENTS OF APPENDIX R TO 10 CFR 50

Summary:

This letter provides NRC staff positions on certain requirements of Appendix R to 10 CFR 50.

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Potential Impact Number: 3842 **SRP Section Number:** 9.5.1

Document: NRC GENERIC LETTER 86-10
Title: IMPLEMENTATION OF FIRE PROTECTION REQUIREMENTS

Summary:

The guidance on implementation of Appendix R, as approved by the Commission, is appended to this letter. This package represents recent staff assessment of these issues and provides guidance as to acceptable methods of satisfying Commission regulatory requirements. Other methods proposed by licensees for complying with Commission regulations may also be satisfactory and will be considered on their own merits. To the extent that this guidance may be inconsistent with prior guidance (including Generic Letter 83-33), it is intended that the current letter takes precedence.

Potential Impact Number: 3898 **SRP Section Number:** 10.3

Document: NRC GENERIC LETTER 86-09
Title: TECHNICAL RESOLUTION OF GENERIC ISSUE NO. B-59-(N-1) LOOP OPERATION IN BWRs AND PWRs

Summary:

The staff recently completed a safety evaluation report (SER) for the request by Beaver Valley Unit 1 for (N-1) loop operation. Based on that SER, it is expected that Beaver Valley Unit 1 will be authorized to operate with (N-1) loops when the Technical Specifications are revised and updated appropriately. The review by the staff of the Beaver Valley Unit 1 submittal has not identified any unacceptable consequences associated with (N-1) loop operation. Therefore, other PWR owners may wish to evaluate the merits of (N-1) loop submittals for their plants based on the Beaver Valley 1 experience or on the approval of (N-1) loop operation for Millstone 3 (Docket No. 50-423), including technical specifications, in conjunction with its operating license review. However, the specific design characteristics of each plant must be reviewed in the same detail as Beaver Valley 1 and Millstone 3 to assure that all safety considerations relative to (N-1) loop operation are evaluated. Examples of PWR considerations which are highly plant specific in nature include effects of the down loop on safety systems such as steam to turbine- driven safety pumps.

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Potential Impact Number: 3941 **SRP Section Number:** 6.2.4

Document: NRC GENERIC LETTER 83-02
Title: NUREG-0737 Technical Specifications

Summary:

This generic letter identifies TMI action plan items for which Technical Specifications are required to promote assurance of acceptable facility operation following implementation.

Certain positions pertain to containment isolation and containment purge and vent valves.

Potential Impact Number: 4059 **SRP Section Number:** 11.2

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION
Location: 20.1301

Summary:

Subpart D of 10 CFR Part 20, Sections 20.1301 and 20.1302, provide radiation dose limits for individual members of the public that were effective for implementation on 20 June, 1991 and became mandatory on January 1, 1994.

The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year.

The annual average concentration of radioactive material released in liquid effluents at the boundary of the unrestricted area do not exceed values specified in Table 2, Appendix B to 20.1001-20.2401.

Potential Impact Number: 4087 **SRP Section Number:** 6.2.4

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Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

Appropriate containment integrity should be addressed.

Potential Impact Number: 4094 **SRP Section Number:** 10.4.9

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.34 (f)(1)(ii)

Summary:

Licensee shall include following information for construction permit: Perform an evaluation of proposed AFW system to include a simplified reliability analysis using event-tree and fault-tree logic techniques, a design review of AFWS, and evaluation of AFWS flow design bases and criteria.

Potential Impact Number: 4095 **SRP Section Number:** 6.2.4

Document: RG 1.141 **Revision:** 0
Title: CONTAINMENT ISOLATION PROVISIONS FOR FLUID SYSTEMS

Summary:

Documents containment isolation with provision for fluid systems.

Potential Impact Number: 4096 **SRP Section Number:** 6.2.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.34.(f).(2).(xiv)

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Summary:

Provide containment isolation systems that: (A) Ensure all non-essential systems are isolated automatically by the containment isolation system, (B) For each non-essential penetration (except instrument lines) have two isolation barriers in series, (C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal, (D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation, (E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

Potential Impact Number: 4098 **SRP Section Number:** 10.4.9

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.62 (c)(1)

Summary:

Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically actuate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to final actuation device) from the existing reactor trip system.

Potential Impact Number: 4138 **SRP Section Number:** 10.4.9

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

If a system is required for primary coolant charging and makeup, reactor coolant pump seal cooling or injection, decay heat removal, or maintaining appropriate containment integrity specifically to meet the station blackout duration recommended by Regulatory Position 3.1, criteria provided in the RG should be met.

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Potential Impact Number: 4155 **SRP Section Number:** 10.4.9

Document: NRC BULLETIN 85-01
Title: STEAM BINDING OF AUXILIARY FEEDWATER PUMPS

Summary:

Requires licensee to develop procedures for monitoring fluid conditions within the AFW system on a regular basis during times when the system is required to be operable until hardware modifications are implemented to resolve GSI-93. Develop procedures for recognizing steam binding and restoring the system to operable status should steam binding occur. Documents suggests increased frequency of AFW temperature monitoring from once per month to once per shift.

Potential Impact Number: 4158 **SRP Section Number:** 9.5.1

Document: NRC GENERIC LETTER 89-13
Title: SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

Summary:

Systems that use raw service water as a source, such as some fire protection systems, should be chlorinated or equally effectively treated before layup to help prevent microbiologically influenced corrosion.

Potential Impact Number: 4159 **SRP Section Number:** 9.5.1

Document: NRC GENERIC LETTER 89-13 Sup 1
Title: SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

Summary:

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Guidance is provided regarding biofouling protection for fire protection water systems.

Potential Impact Number: 4167 **SRP Section Number:** 6.5.1

Document: RG 1.52 **Revision:** 2
Title: DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR POSTACCIDENT ENGINEERED-SAFETY-FEATURE ATMOSPHERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

Summary:

System Design criteria are contained in this reg guide. ESF atmosphere cleanup systems that are designed and installed for the purpose of mitigating accident doses should be redundant. The systems should consist of the following sequential components: (1) demisters, (2) prefilters (demisters may serve this function), (3) HEPA filters before the adsorbers, (4) iodine adsorbers (impregnated activated carbon or equivalent adsorbent such as metal zeolites), (5) HEPA filters after the adsorbers, (6) ducts and valves, (7) fans, and (8) related instrumentation. Heaters or cooling coils used in conjunction with heaters should be used when the humidity is to be controlled before filtration.

Potential Impact Number: 4168 **SRP Section Number:** 14.2

Document: RG 1.68.3 **Revision:** 0
Title: PREOPERATIONAL TESTING OF INSTRUMENT AND CONTROL AIR SYSTEMS

Summary:

This guide, which is a revision of and replaces Regulatory Guide 1.80, describes a method acceptable to the NRC staff for complying with the Commission's regulations with respect to preoperational testing to verify that instrument and control air systems and the loads they supply will operate properly at normal system pressures and to ensure the operability of functions important to safety in the event that system pressure is lost, reduced below normal operating level, or increased above the design pressure of the air system components to the upstream safety valve accumulation pressure.

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The recommendations of this guide should also be applied to (1) compressed gas systems that, although not designated as instrument and control air systems, supply loads that could affect the overall safety and performance of the plant, (2) compressed gas systems that supply loads only to specific systems that are important to safety, and (3) systems important to safety that use compressed gases other than air.

Potential Impact Number: 4212 **SRP Section Number:** 9.3.4

Document: NRC BULLETIN 80-18
Title: MAINTENANCE OF ADEQUATE MINIMUM FLOW THRU CENTRIFUGAL CHARGING PUMPS FOLLOWING SECONDARY SIDE HIGH ENERGY LINE RUPTURE

Summary:

The bulletin identifies that under certain conditions the centrifugal charging pumps could be damaged due to lack of minimum flow. The particular circumstances that could result in damage vary somewhat from plant to plant, but involve unavailability of the pressurizer power operated relief valves, with operation of one or more CCPs repressurizing the reactor during SI following a secondary system high energy line break.

Potential Impact Number: 4283 **SRP Section Number:** 14.2

Document: RG 1.68.1 **Revision:** 1
Title: PREOPERATIONAL AND INITIAL STARTUP TESTING OF FEEDWATER AND CONDENSATE SYSTEMS FOR BOILING WATER REACTOR POWER PLANTS

Summary:

Regulatory Guide 1.68, 'Initial Test Programs for Water-Cooled Reactor Power Plants,' identifies tests acceptable to the NRC staff for preoperational and startup testing of nuclear power plant structures, systems, and components. 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. This guide

describes in more detail the type and nature of BWR feedwater and condensate system tests that are acceptable to the staff.

Potential Impact Number: 4284 **SRP Section Number:** 6.2.5

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue (A-48) was resolved with an update to 10CFR50.44 (see SECY-89-122). The following rules were implemented with regard to hydrogen control in containments: (1) BWR Mark I and II containment must use methods for producing an inert containment atmosphere (using nitrogen gas for example), (2) Mark III and PWR ice condenser containments should provide means for controlling the hydrogen produced from a 75% fuel-cladding metal-water reaction, and (3) hydrogen control in large dry PWR containments was determined not to be a concern.

Potential Impact Number: 4298 **SRP Section Number:** 5.2.1.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: 50.55a (c)(3), 50.55a (c)(3), 50.55a (d)(2), 50.55a (e)(2)

Summary:

In 50.55a, subparagraphs (c)(3), (d)(2), (e)(2) all state that the method for determining the applicable Code Edition, Addenda, and optional Code Cases to be applied to relevant components/systems must be determined by the provisions of paragraph NCA-1140, Subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code.

In addition, footnote 6 states, ASME Code cases that have been determined suitable for use by the commission staff are listed in: NRC Regulatory Guide 1.84, "Design and Code Case Acceptability --ASME Section III Division 1," NRC Regulatory Guide 1.85, "Materials Code Case Acceptability --ASME Section III Division 1," and NRC Regulatory

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Guide 1.147, "Inservice Inspection Code Case Acceptability --ASME Section XI Division 1."

Potential Impact Number: 4321 **SRP Section Number:** 3.7.4

Document: RG 1.12 **Revision:** 1
Title: INSTRUMENTATION FOR EARTHQUAKES

Summary:

This guide describes acceptable seismic instrumentation for satisfying the requirements of Appendix A to 10CFR100.

Potential Impact Number: 4396 **SRP Section Number:** 9.3.2

Document: NRC GENERIC LETTER 83-36
Title: NUREG-0737 TECHNICAL SPECIFICATIONS

Summary:

Provides staff position (related to TMI Item II.B.3) that a Technical Specification administrative program should ensure the capability to obtain and analyze reactor coolant and containment atmosphere samples under accident conditions. The program should include: training, procedures for sampling/analysis, and provisions for maintenance of sample/analysis equipment.

Also states that it is acceptable to the staff, if the licensee elects to reference this program in the administrative controls section of the Technical Specifications and include a detailed description of the program in the plant operation manuals. A copy of the program should be readily available to the operating staff during accident and transient conditions.

Potential Impact Number: 4397 **SRP Section Number:** 9.3.4

Document: NRC BULLETIN 80-05

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Title: VACUUM CONDITION RESULTING IN DAMAGE TO CHEMICAL VOLUME CONTROL SYSTEM (CVCS) HOLDUP TANKS (SOMETIMES CALLED 'CLEAN WASTE RECEIVER TANKS')

Summary:

As specified by the required actions licensees should consider installing vacuum protection in their plants, appropriately designed to preclude collapse of the tanks. Any proposed or already installed vacuum protection system must consider the necessity for precluding adverse effects due to operation or misoperation of the system (for example, creation of an explosive gas mixture if hydrogen gas is present in the HUT). Any proposed or already installed system must also include consideration of the following: (a) tanks with a cover gas must be able to admit the cover gas fast enough to keep up with the maximum rate of liquid removal from the tank; (b) vacuum relief valves must be covered by an acceptable surveillance program; (c) tanks that are located outside must have adequate freeze protection for the tank and for the vacuum relief system.

Potential Impact Number: 4435 **SRP Section Number:** 9.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.34 (f)(2)(viii)

Summary:

(2) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage.....

(viii) Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID 14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 75 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)

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This regulation was subsequently amended resulting in the following exposure limits: "... without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities." Reference PIs 22547 and 23692.

Potential Impact Number: 4489 **SRP Section Number:** 3.6.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 4

Summary:

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Potential Impact Number: 4575 **SRP Section Number:** 14.2

Document: NRC GENERIC LETTER 83-24
Title: TMI Task Action Plan Item I.G.1, 'Special Low Power Testing and Training,' Recommendations for BWRs

Summary:

The BWR Owners' Group initial response to TMI Item I.G.1 was that it should not apply to BWRs as there are no additional tests analogous to the PWR tests which would provide meaningful technical information and supplemental operator training. In a letter from D. B. Waters to D. G. Eisenhut dated February 4, 1981, the BWR Owners' Group subsequently proposed that BWR applicants meet the I.G.1 requirement by augmenting

reactor operator participation in the initial test program and by some additional preoperational tests. After review of this response, it was the staff's position that, to ensure compliance with I.G.I, the BWR applicants be required to do some additional startup testing beyond that called for by Regulatory Guide 1.68 (and in addition to some "new" tests proposed by the BWR Owners' Group). The staff subsequently requested BWR applicants to commit to the recommendations of the Owners' Group and, in addition, to perform a "Simulated Loss of All A-C Power" (Station Blackout-SBO) test. The objective of the SBO test was to determine the temperature, pressure, and level responses and associated time constants of the reactor, drywell, containment, and vital spaces in event of a loss of heating, ventilation, and air conditioning (HVAC) and cooling water, with decay heat being rejected to the suppression pool via the safety-relief valves. Decay heat was to be simulated by nuclear heat produced at low power, or the test could be postponed until later in the fuel cycle when sufficient decay heat was available.

Upon further review of the basis for the requirement, the practicalities and value of such a test, and the proposed augmented Owners' Group program, we conclude that the SBO test does not provide significant new information to justify its performance. Furthermore, since one of the original criteria for I.G.1 special tests (as stated in the Sequoyah SER) is that the test must not pose a hazard to plant equipment, the staff now recommends that the SBO test be deleted from the BWR I.G.1 staff position.

Potential Impact Number: 4752 **SRP Section Number:** 6.1.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

The following rules were implemented with regard to hydrogen control in containments: (1) BWR Mark I and II containment must use methods for producing an inert containment atmosphere (using nitrogen gas for example), (2) Mark III and PWR ice condenser containments should provide means for controlling the hydrogen produced from a 75% fuel-cladding metal-water reaction, and (3) hydrogen control in large dry PWR containments was determined not to be a concern. This issue was resolved with an update to 10CFR50.44 (see SECY-89-122).

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Potential Impact Number: 4953 **SRP Section Number:** 14.2

Document: RG 1.68.2 **Revision:** 1
Title: INITIAL STARTUP TEST PROGRAM TO DEMONSTRATE REMOTE SHUTDOWN CAPABILITY FOR WATER-COOLED NUCLEAR POWER PLANTS

Summary:

This guide 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. This guide is applicable to watercooled nuclear power plants.

Regulatory Guide 1.68 lists as one of its initial startup tests the demonstration of 'shutdown from outside the control room,' the capability for which is required by GDC 19. Experience in the application of Regulatory Guide 1.68 has shown that amplification of guidance for initial test programs in this area is desirable.

Potential Impact Number: 5002 **SRP Section Number:** 14.2

Document: RG 1.139 **Revision:** 0
Title: GUIDANCE FOR RESIDUAL HEAT REMOVAL

Summary:

SUMMARY: RHR System testing requirements specified.

The isolation valve operability and interlock circuits should be designed to permit on-line testing when operating in the RHR mode. System testing should meet the requirements of IEEE Standard 338 and the recommendations of Regulatory Guide 1.22. The preoperational and initial startup test program should be in conformance with Regulatory Guide 1.68. The programs for PWRs should include tests with supporting analysis to confirm (a) that adequate mixing of borated water added prior to or during cooldown can be achieved under natural circulation conditions and permit estimation of the times required to achieve such mixing and (b) that the cooldown under natural circulation conditions can be achieved within the limits specified in the emergency operating procedures. Comparison with the performance of previously tested plants of similar design may be substituted for these tests.

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Potential Impact Number: 5055 **SRP Section Number:** 11.4

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION
Locations: Subpart K, Subpart K, 20.2001

Summary:

A licensee shall dispose of licensed material only by decay in storage; or by release in effluents within the limits in 20.1301.

A person must be specifically licensed to receive waste containing licensed material from other persons for treatment or disposal by incineration; or disposal at a land disposal facility licensed under part 61; or disposal at a geologic repository under part 60 of this chapter.

Each application shall include (a) A description of the waste containing licensed material to be disposed of, including the physical and chemical properties important to risk evaluation, and the proposed manner and conditions of waste disposal; and (b) An analysis and evaluation of pertinent information on the nature of the environment; and (c) The nature and location of other potentially affected licensed and unlicensed facilities; and (d) Analyses and procedures to ensure that doses are maintained ALARA and within the dose limits in this part.

Potential Impact Number: 5087 **SRP Section Number:** 11.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 61

Summary:

Criterion 61 -- Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that

reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Potential Impact Number: 5152 **SRP Section Number:** 3.10

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: 10CFR100 Appendix A Section VI (a) (2)

Summary:

The Operating Basis Earthquake shall be defined by response spectra. All structures, systems, and components of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public shall be designed to remain functional and within applicable stress and deformation limits when subjected to the effects of the vibratory motion of the Operating Basis Earthquake in combination with normal operating loads. The engineering method used to insure that these structures, systems, and components are capable of withstanding the effects of the Operating Basis Earthquake shall involve the use of either a suitable dynamic analysis or a suitable qualification test to demonstrate that the structures, systems and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism. The analysis or test shall take into account soil-structure interaction effects and the expected duration of vibratory motion.

Potential Impact Number: 5173 **SRP Section Number:** 3.10

Document: RG 1.97 **Revision:** 3
Title: INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT

Summary:

The instrumentation should be qualified in accordance with Regulatory Guide 1.89, Qualification of Class 1E Equipment for Nuclear Power Plants, and the methodology

described in NUREG- 0588, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment. Instrumentation whose ranges are required to extend beyond those ranges calculated in the most severe design basis accident event for a given variable should be qualified using the guidance provided in paragraph 6.3.6 of ANS-4.5. Qualification applies to the complete instrumentation channel from sensor to display where the display is a direct-indicating meter or recording device. If the instrumentation channel signal is to be used in a computer-based display, recording, or diagnostic program, qualification applies from the sensor up to and including the channel isolation device. The seismic portion of qualification should be in accordance with Regulatory Guide 1.100, Seismic Qualification of Electric Equipment for Nuclear Power Plants. Instrumentation should continue to read within the required accuracy following, but not necessarily during, a safe shutdown earthquake.

Potential Impact Number: 5174 **SRP Section Number:** 3.10

Document: RG 1.97 **Revision:** 3
Title: INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT

Summary:

It is desirable that accident-monitoring instrumentation components and their mounts that cannot be located in seismically qualified buildings be designed to continue to function, to the extent feasible, following seismic events. An acceptable method for enhancing the seismic resistance of this instrumentation would be to design it to meet the seismic criteria applicable to like instrumentation installed in seismically qualified locations although a lesser overall qualification results.

Potential Impact Number: 5219 **SRP Section Number:** 14.2

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

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Item I.C.7 requires NSSS vendor review of the low power and the power ascension test program test procedures.

Potential Impact Number: 5298 **SRP Section Number:** 15.4.6

Document: NRC NOTICE 82-19
Title: LOSS OF HIGH HEAD SAFETY INJECTION EMERGENCY BORATION AND REACTOR COOLANT MAKEUP CAPABILITY

Summary:

Describes incident where a plant experienced a loss of high head safety injection emergency boration and reactor coolant makeup capability when hydrogen from the positive displacement pump suction dampener entered the common suction of the charging system causing the CCPs and PDP to become inoperable.

Potential Impact Number: 5305 **SRP Section Number:** 15.4.6

Document: NRC NOTICE 87-57
Title: LOSS OF EMERGENCY BORATION CAPABILITY DUE TO NITROGEN GAS INTRUSION

Summary:

Describes incident where a plant experienced a loss of the ability to borate and emergency borate affecting the ability to maintain required shutdown margin. The cause was in leakage of nitrogen gas from a boric acid pump seal cover gas system which caused the necessary pumps for boration to become air bound.

Potential Impact Number: 5477 **SRP Section Number:** 3.8.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50, Appendix A, GDC 4

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Summary:

Basis for dynamic loading requirements.

Potential Impact Number: 5524 **SRP Section Number:** 3.8.4

Document: RG 1.142 **Revision:** 1
Title: SAFETY-RELATED CONCRETE STRUCTURES FOR NUCLEAR POWER PLANTS (OTHER THAN REACTOR VESSELS AND CONTAINMENTS)

Summary:

Structural design requirements for concrete structures. Discussion on NRC accepted Industry Code and Standards.

Potential Impact Number: 5744 **SRP Section Number:** 5.4.6

Document: NRC GENERIC LETTER 83-02
Title: NUREG-0737 Technical Specifications

Summary:

Requires licensee action to review and revise the facility technical specifications for 1) RCIC restart on low water level after termination on high water level and 2) automatic RCIC suction switch over from storage tank to suppression pool at low condensate storage tank level. (II.K.3.13 and II.K.3.22).

Potential Impact Number: 5748 **SRP Section Number:** 5.4.6

Document: NRC GENERIC LETTER 83-02
Title: NUREG-0737 Technical Specifications

Summary:

Requires addition of time delay modification for RCIC pipe break (isolation) circuitry into the technical specifications. (III.K.3.15).

Potential Impact Number: 5954 **SRP Section Number:** 3.9.2

Document: RG 1.29 **Revision:** 3
Title: SEISMIC DESIGN CLASSIFICATION

Summary:

C.1.a,b,c - The reactor coolant pressure boundary, reactor core and internals, the emergency core cooling system are seismic category I systems.

C.2 - Those portions of structures, systems, or components whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items 1.a through 1.q above to an unacceptable safety level or could result in incapacitating injury to occupants of the control room should be designed and constructed so that the SSE would not cause such failure.

C.3 - Seismic Category I design requirements should extend to the first seismic restraint beyond the defined boundaries. Those portions of structures, systems, or components that form interfaces between Seismic Category I and non-Seismic Category I features should be designed to Seismic Category I requirements.

Potential Impact Number: 6065 **SRP Section Number:** 5.3.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Neutron irradiation of reactor vessel welds and plate materials reduces their fracture toughness, thus increasing the likelihood of crack formation in these materials if the vessel should be overcooled. This issue was resolved with the approval of 10 CFR 50.61 and the publication of Reg. Guide 1.154.

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Potential Impact Number: 6068 **SRP Section Number:** 5.3.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Neutron irradiation of reactor vessel welds and plate materials reduces their fracture toughness, thus increasing the likelihood of crack formation in these materials if the vessel should be overcooled. This issue was resolved with the approval of 10CFR50.61 and the publication of Reg. Guide 1.154.

Potential Impact Number: 6215 **SRP Section Number:** 10.4.2

Document: RG 1.123 **Revision:** 1
Title: QUALITY ASSURANCE REQUIREMENTS FOR CONTROL OF PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR POWER PLANTS

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to quality assurance requirements for control of procurement of items/1/ and services during the design, construction, and operations phases of nuclear power plants.

Potential Impact Number: 6301 **SRP Section Number:** 10.4.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 4

Summary:

Environmental and dynamic design bases

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Potential Impact Number: 6331 **SRP Section Number:** 2.5.1

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: Appendix A.I

Summary:

Purpose is to set forth the principal seismic geologic conditions to guide the Commission in evaluation of suitability of proposed sites.

Potential Impact Number: 6332 **SRP Section Number:** 2.5.1

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: Appendix A.II

Summary:

Describes the scope of investigations, procedures and activities based upon investigation results required for the development of design bases with respect to geologic and seismic considerations.

Potential Impact Number: 6333 **SRP Section Number:** 2.5.1

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: Appendix A.II

Summary:

Each applicant for a construction permit shall investigate all seismic and geologic factors that may affect the design and operation of the proposed nuclear power plant

irrespective of whether such factors are explicitly included in these criteria (in 10 CFR 100 Appendix A).

Potential Impact Number: 6364 **SRP Section Number:** 9.2.4

Document: RG 1.70 **Revision:** 3
Title: STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS

Summary:

A description of the potable and sanitary water systems should be provided. System design criteria should provide for prevention of connections to systems having the potential for containing radioactive material. An evaluation of radiological contamination, including accidental, and safety implications of sharing (for multi-unit facilities) should be described.

Potential Impact Number: 6434 **SRP Section Number:** 3.11

Document: RG 1.40 **Revision:** 0
Title: QUALIFICATION TESTS OF CONTINUOUS-DUTY MOTORS INSTALLED INSIDE THE CONTAINMENT OF WATER-COOLED NUCLEAR POWER PLANTS

Summary:

IEEE Std 334-1971 is generally acceptable and provides an adequate basis for complying with the qualification testing to verify adequacy of design for service under the most adverse design conditions. To the extent practicable, auxiliary equipment that will be part of the installed motor assembly should also be qualified. The qualification test(s) should simulate as closely as practicable all design basis events which affect operation of the motor's auxiliary equipment.

Potential Impact Number: 6471 **SRP Section Number:** 3.11

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Document: RG 1.89 **Revision:** 1
Title: ENVIRONMENTAL QUALIFICATION OF CERTAIN ELECTRIC EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS

Summary:

This regulatory guide describes a method acceptable to the NRC staff for complying with 50.49 of 10 CFR Part 50 with regard to qualification of electric equipment important to safety for service in nuclear power plants to ensure that the equipment can perform its safety function during and after a design basis accident.

Potential Impact Number: 6488 **SRP Section Number:** 5.4.7

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue addresses the design of the containment sump or RHR suction intake so as to prevent the formation of vortices in the suction of the RHR and containment spray pumps. These vortices could lead to air ingestion and failure of these pumps. This issue is covered under USI A-43, 'Containment Sump Performance.'

Potential Impact Number: 6556 **SRP Section Number:** 2.5.1

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: Appendix A. III

Summary:

Definitions of seismic and geologic terms.

Potential Impact Number: 6560 **SRP Section Number:** 2.5.1

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Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: Appendix A.IV

Summary:

Geologic, seismic and engineering characteristics shall be investigated with scope and detail to provide assurance that they are sufficiently well understood to evaluate site.

Potential Impact Number: 6561 **SRP Section Number:** 2.5.1

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: Appendix A.IV.(a)(2)

Summary:

Determine lithologic, stratigraphic, hydrologic and structural geologic conditions including geologic history of site and region.

Potential Impact Number: 6562 **SRP Section Number:** 2.5.1

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: Appendix A.IV(a)(3)

Summary:

Requires evaluation of behavior during prior earthquakes of surficial geologic materials and substrata underlying site.

Potential Impact Number: 6563 **SRP Section Number:** 2.5.1

Document: CODE OF FED. REGS 10CFR100

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Potential Impacts

Title: REACTOR SITE CRITERIA
Location: Appendix A.IV.(a)(4)

Summary:

Determine static and dynamic engineering properties of materials underlying site--behavior of underlying materials and characteristics of underlying materials in transmitting earthquake induced motions.

Potential Impact Number: 6567 **SRP Section Number:** 2.5.1

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: Appendix A.IV.(a)(7)

Summary:

Requires Determination whether faults within 200 miles of site are capable, for use in establishing Safe Shutdown Earthquake.

Potential Impact Number: 6568 **SRP Section Number:** 2.5.1

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: Appendix A.IV.(a)(8)

Summary:

Requires determination of characteristics of capable faults during site investigations for vibratory ground motion.

Potential Impact Number: 6580 **SRP Section Number:** 2.5.1

Document: CODE OF FED. REGS 10CFR100

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Potential Impacts

Title: REACTOR SITE CRITERIA
Location: Appendix A.IV.(c)(1)(iii)

Summary:

Coastal site investigations shall include appropriate geologic and seismic evidence to provide information establishing design basis for seismically induced floods or from coastal subsidence.

Potential Impact Number: 6583 **SRP Section Number:** 2.5.1

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: Appendix A.IV.(c)(2)

Summary:

Lake and River site investigations, requirements for determining potential for seismically induced floods and water waves as from dam failure and earthslides.

Potential Impact Number: 6600 **SRP Section Number:** 2.5.1

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: Appendix A.V.(d)(1)

Summary:

Requires determination of other design bases conditions based upon site characteristics such as soil stability, slope stability, adequacy of safety related cooling water supply, and adequacy of distant (not necessarily onsite) safety related structures.

Potential Impact Number: 6756 **SRP Section Number:** 13.2.1

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Document: CODE OF FED. REGS 10CFR55
Title: OPERATORS' LICENSES
Location: Section 55.45 (a)(1) through (13)

Summary:

The operating test requires the applicant to demonstrate ability to perform various operational actions, such as control manipulation and observation of status parameters, necessary for safe plant operation.

Potential Impact Number: 6797 **SRP Section Number:** 4.4

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Hydrodynamic flow instabilities may occur in a BWR when two-phase flow exists in a channel with particular dimensions and flow parameters. These instabilities may cause power oscillations which lead to local violations of the critical heat flux limits. However, the NRC staff has determined that this does not pose an immediate safety concern for continued BWR operation prior to orderly examination and possible TS changes. Therefore, this issue was resolved and no new requirements were established.

Potential Impact Number: 6912 **SRP Section Number:** 3.11

Document: RG 1.97 **Revision:** 3
Title: INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-watercooled nuclear power plant.

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Potential Impact Number: 6991 **SRP Section Number:** 5.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix G, Section IV.A.1

Summary:

Reactor vessel beltline materials must have Charpy upper-shelf energy of no less than 75 ft-lb initially and must maintain upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code.

Potential Impact Number: 7107 **SRP Section Number:** 5.3.2

Document: RG 1.154 **Revision:** 0
Title: FORMAT AND CONTENT OF PLANT-SPECIFIC PRESSURIZED THERMAL SHOCK SAFETY ANALYSIS REPORTS FOR PRESSURIZED WATER REACTORS

Summary:

This guide provides the recommended methods used to analyze the risk due to PTS events for proposed operation of plants with reactor vessel RT(PTS) above the screening criterion of 10CFR Section 50.61.

Potential Impact Number: 7133 **SRP Section Number:** 13.5.1.1

Document: NRC GENERIC LETTER 82-12
Title: NUCLEAR POWER PLANT STAFF WORKING HOURS

Summary:

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Describes revisions to earlier policy statements on staff working hours and overtime. Enough personnel should be employed to not require heavy use of overtime.

Potential Impact Number: 7170 **SRP Section Number:** 13.4

Document: RG 1.33 **Revision:** 2
Title: QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION)

Summary:

An Audit Program shall review the performance, training, and qualifications of the facility staff at least once per 12 months.

Potential Impact Number: 7213 **SRP Section Number:** 13.5.1.1

Document: NRC GENERIC LETTER 82-12
Title: NUCLEAR POWER PLANT STAFF WORKING HOURS

Summary:

Licensees should ensure that qualified plant personnel to man the operational shifts are readily available in the event of an abnormal or emergency situation.

Potential Impact Number: 7225 **SRP Section Number:** 3.11

Document: RG 1.131 **Revision:** 0
Title: QUALIFICATION TESTS OF ELECTRIC CABLES, FIELD SPLICES, AND CONNECTIONS FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

Summary:

This regulatory guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to qualification testing of electric cables, field splices, and connections for service in light-water-cooled nuclear power plants to

ensure that the cables, field splices, and connections can perform their safety- related functions.

Potential Impact Number: 7239 **SRP Section Number:** 3.11

Document: RG 1.156 **Revision:** 0
Title: ENVIRONMENTAL QUALIFICATION OF CONNECTION ASSEMBLIES

Summary:

This regulatory guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to the environmental qualification of quick-disconnect connection assemblies and environmental seals in combination with cables or wires as assemblies for service in nuclear power plants. The environmental qualification is to ensure that connection assemblies can perform their safety functions during and after a design basis event.

Potential Impact Number: 7240 **SRP Section Number:** 3.11

Document: RG 1.158 **Revision:** 0
Title: QUALIFICATION OF SAFETY-RELATED LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS

Summary:

This regulatory guide describes a method acceptable to the NRC staff for complying with Commission regulations with regard to qualification of safety related lead storage batteries for nuclear power plants.

Conformance with the requirements of IEEE Std 535-1986 is an acceptable method for qualification of safety-related lead storage batteries.

Potential Impact Number: 7271 **SRP Section Number:** 5.4.7

Document: NUREG 0933

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Title: A Prioritization of Generic Safety Issues

Summary:

This issue addresses the behavior of various kinds of pipe or component insulation under pipe break accident conditions. There is a concern that insulation debris from such accidents may impair flow through the containment sump strainer and/or damage the RHR or containment spray pumps. This issue is covered under USI A-43, 'Containment Emergency Sump Performance.'

Potential Impact Number: 7377 **SRP Section Number:** 9.5.4

Document: NRC GENERIC LETTER 83-26
Title: Clarification of Surveillance Requirements for Diesel Fuel Impurity Level Tests

Summary:

At least once per 92 days, or when adding new fuel, verify per ASTM D270-1975 and ASTM D965-1977; Water < .05 volume %, kinematic viscosity between 1.9 and 4.1 centistokes, specific gravity, impurities of less than 2 mg per 100 ml, and other properties of Reg. Guide 1.137 Rev. 1, position 2a within 14 days. At least once per 92 days (31 days if groundwater table higher than tank bottom) check, and remove accumulated water from fuel oil storage.

Potential Impact Number: 7402 **SRP Section Number:** 13.5.1.1

Document: NRC GENERIC LETTER 82-02
Title: Nuclear Power Plant Staff Working Hours

Summary:

Reiterates and clarifies the NRC's position that operation personnel not be assigned duties while in a fatigued condition; addresses staffing, overtime, and scheduling requirements.

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Potential Impact Number: 7404 **SRP Section Number:** 13.5.1.1

Document: NRC GENERIC LETTER 82-12
Title: NUCLEAR POWER PLANT STAFF WORKING HOURS

Summary:

Reiterates and clarifies the NRC's position that operation personnel not be assigned duties while in a fatigued condition; addresses staffing, overtime, and scheduling requirements.

Potential Impact Number: 7487 **SRP Section Number:** 3.9.6

Document: NRC GENERIC LETTER 89-04
Title: GUIDANCE OF DEVELOPING ACCEPTABLE INSERVICE TESTING PROGRAMS

Summary:

This letter has been written to provide guidance relative to meeting the requirements of 10 CFR 50.55a(g). Through reviews and inspections, the NRC staff has identified a number of generic deficiencies that affect plant safety and have frequently appeared as IST programmatic weaknesses. These programmatic weaknesses can be attributed to a lack of understanding in the following areas: (1) Code testing requirements, (2) technical specification requirements, and (3) acceptable alternatives to Code requirements. In order to remedy these generic IST deficiencies, to clarify the status of current programs with respect to applicable T.S. and 10 CFR 50.55a requirements, and to alleviate the problem with respect to review of program revisions, the NRC has established the guidance detailed in Attachment 1 of the letter.

Attachment 1 provides staff guidance on the following topics:
1. Full Flow Testing of Check Valves
2. Alternative to Full Flow Testing of Check Valves
3. Back Flow Testing of Check Valves
4. Pressure Isolation Valves
5. Limiting Values of Full-Stroke Times for Power Operated Valves
6. Stroke Time Measurements for Rapid-Acting Valves
7. Testing Individual Control Rod Scram Valves in BWRs
8. Starting Point for Time Period in TS ACTION Statements

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9. Pump Testing using Minimum-flow Return Line With or Without Flow Measuring Devices
10. Containment Isolation Valve Testing
11. IST Program Scope

Potential Impact Number: 7509 **SRP Section Number:** 3.4.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Backflow through floor drains could result in flooding of the compartments containing safety-related systems and equipment, thus compromising the intended function of this equipment. A permanent solution to this problem is to install check valves in all floor drain piping running from such compartments. This issue was integrated into the resolution of Task Action Plan Item A-17.

Potential Impact Number: 7642 **SRP Section Number:** 6.6

Document: NRC BULLETIN 82-02
Title: DEGRADATION OF THREADED FASTENERS IN THE REACTOR COOLANT PRESSURE BOUNDARY OF PWR PLANTS

Summary:

This bulletin discusses problems in applying Section XI ultrasonic inspections in cases of corrosion or wastage. The above concerns are further compounded by the fact that under the present ASME Code Section XI inservice inspection rules ultrasonic examination is not required on threaded fasteners in sizes 2 inches and less in diameter (e.g., Table IWB-2500-1). However, except for the reactor coolant pump stud wastage, most failures have occurred in fastener sizes 2 inches and smaller. Furthermore, experience has clearly shown that Code-specified ultrasonic testing (UT) methods are not singularly adequate to detect corrosion wastage conditions. Moreover, the present Code UT procedures are not sufficiently sensitive to detect initiation of stress corrosion cracking (SCC) but requires the use of specialized UT techniques and calibration standards based on notch reflectors simulating critical flaw

parameters to enhance reliability of detection. At the present time, visual examination (e.g., IWA 2210, VT-I) appears to be the only method to detect borated water corrosion wastage or erosion-corrosion damage and may require insulation removal and/or disassembly of the component, in some cases, in order to have direct visual access to the threaded fasteners. Therefore, degradation could go undetected when there is no clear evidence of leakage in the surrounding area. Similarly, the reliability of visual examination alone is questionable in detection of SCC initiation of threaded fasteners either in-situ or removed. Accordingly, it is necessary that a combination of nondestructive examination techniques (UT, VT- 1, MT, PT) be employed to the maximum extent practical to enhance detection of the degradation mechanisms discussed above.

Potential Impact Number: 7648 **SRP Section Number:** 6.6

Document: NRC GENERIC LETTER 81-03
Title: IMPLEMENTATION OF NUREG-0313, REV. 1, 'TECHNICAL REPORT ON MATERIAL SELECTION AND PROCESSING GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING (GENERIC TASK A-42)

Summary:

NUREG-0313, Rev. 1 sets forth the NRC staff's revised acceptable methods to reduce the intergranular stress corrosion cracking susceptibility of BWR ASME Code Class 1, 2, and 3 pressure boundary piping and safe ends. For plants that cannot fully comply with the material selection, testing and processing guidelines of this report, varying degrees of augmented Inservice Inspection and leak detection requirements are presented.

Potential Impact Number: 7662 **SRP Section Number:** 6.6

Document: NRC GENERIC LETTER 88-01
Title: NRC POSITION ON IGSCC IN BWR AUSTENITIC STAINLESS STEEL PIPING

Summary:

Attachment A to this letter contains staff positions on: 1) inspection methods and personnel; 2) inspection schedules; and, 3) sample expansion. The technical bases

for these positions are detailed in NUREG-0313, Rev. 2 "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping."

This Generic Letter applies to all BWR piping made of austenitic stainless steel that is four inches or larger in nominal diameter and contains reactor coolant at a temperature above 200°F during power operation regardless of Code classification. It also applies to reactor vessel attachments and appurtenances such as jet pump instrumentation penetration assemblies and head spray and vent components. This Generic Letter does not apply to piping made of carbon steel classified as P-1 by the ASME Boiler and Pressure Vessel Code.

Potential Impact Number: 7682 **SRP Section Number:** 6.6

Document: NRC GENERIC LETTER 89-13
Title: SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

Summary:

Nuclear power plant facilities of licensees and applicants must meet the minimum requirements of the General Design Criteria (GDC) in 10 CFR Part 50, Appendix A. In particular, 'GDC 44--Cooling Water' requires provision of a system (here called the service water system) 'to transfer heat from structures, systems, and components important to safety to an ultimate heat sink' (UHS). 'GDC 45--Inspection of Cooling Water System' requires the system design 'to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.' 'GDC 46--Testing of Cooling Water System' requires the design 'to permit appropriate periodic pressure and functional testing.'

In addition, nuclear power plant facilities of licensees and applicants must meet the minimum requirements for quality assurance in 10 CFR Part 50, Appendix B. In particular, Section XI, 'Test Control,' requires that 'a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

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Recent operating experience and studies have led the NRC to question the compliance of the service water systems in the nuclear power plants of licensees and applicants with these GDC and quality assurance requirements.

Therefore, this Generic Letter is being issued to require licensees and applicants to supply information about their respective service water systems to assure the NRC of such compliance and to confirm that the safety functions of their respective service water systems are being met.

Potential Impact Number: 8023 **SRP Section Number:** 5.4.7

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue addresses the following concerns: (1) the resistance of the containment sump suction intake to vortex formation and air ingestion to the RHR or containment spray pumps, (2) the ability of the suction strainers for these pumps to prevent the ingestion of debris which may be generated by a LOCA, and (3) the ability of the RHR and containment spray pump to resist damage due to air or debris ingestion. This issue was resolved with the revision of SRP Section 6.2.2 (Rev. 4) and Reg. Guide 1.82 (Rev. 1).

Potential Impact Number: 8025 **SRP Section Number:** 6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue addresses the following concerns: (1) the resistance of the containment sump suction intake to vortex formation and air ingestion to the RHR or containment spray pumps, (2) the ability of the suction strainers for these pumps to prevent the ingestion of debris which may be generated by a LOCA, and (3) the ability of the RHR and containment spray pump to resist damage due to air or debris ingestion. This

issue was resolved with the revision of SRP Section 6.2.2 (Rev. 4) and Reg. Guide 1.82 (Rev. 1).

Potential Impact Number: 8326 **SRP Section Number:** 15.6.5

Document: NRC GENERIC LETTER 83-10A
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

We have completed our evaluation of your analyses of LOFT Test L3-6 and conclude that the evaluations acceptably predict the test results. Therefore, we find the currently approved B&W evaluation model for small break LOCAs in continued conformance with Appendix K to 10 CFR 50 for the case of limited RCP operation after reactor trip and for the range of licensed B&W reactor designs.

We have reviewed industry analyses and performed our own analyses to determine whether RCP trip is necessary during LOCAs, and evaluated the desirability of continued RCP operation during non-LOCA transients and accidents, including steam generator tube ruptures. We have 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, some of the calculations show that for certain small break LOCAs, primarily those with only one of the two High Pressure Safety Injection (HPSI) Pumps assumed available, continued operation of the RCPs or continued operation of the RCPs followed by delayed RCP trip could lead to core damage.

In summary, we have concluded that the need for RCP trip following a transient or accident should be determined by each licensee on a case-by-case basis, considering the Owners Group input. However, the staff must 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 licensees to comply with the Commission's rules and regulations.

We recognize that possible differences exist between the requirements of 10 CFR 50.46, which assure ample core cooling capacity, and the approaches described in the enclosure which are based upon assuring proper operator/ system response under conditions that may be faced during accidents and transients. Accordingly, in such

cases, we will consider a request for exemption from specific requirements of 10 CFR 50.46 pursuant to 10 CFR 50.12.

For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual licensees, we strongly urge that licensees work collectively with owners of similar plants (i.e., owners Group) and propose setpoints and methods consistent with other licensees.

However, once the technical bases for the decision are established, we intend to conduct inspections of individual licensees led by Regional personnel. During these inspections, we will examine the translation of the 10 CFR 50, Appendix K, and RCP operation mode evaluations into plant procedures.

Potential Impact Number: 8327 **SRP Section Number:** 15.6.5

Document: NRC GENERIC LETTER 83-10B
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

This letter provides guidance for a Combustion Engineering NSSS for the development of either satisfactory setpoints for RCP trip or the technical bases for allowing continued RCP operation in the event of a SBLOCA.

Potential Impact Number: 8328 **SRP Section Number:** 15.6.5

Document: NRC GENERIC LETTER 83-10C
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS' (GENERIC LETTER NO. 83-l0c)

Summary:

This letter provides guidance for a Westinghouse NSSS for the development of either satisfactory setpoints for RCP trip or the technical bases for allowing continued RCP operation in the event of a SBLOCA.

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Potential Impact Number: 8329 **SRP Section Number:** 15.6.5

Document: NRC GENERIC LETTER 83-10D
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

We have completed our evaluation of your analyses of LOFT Test L3-6 and conclude that the evaluations acceptably predict the test results. Therefore, we find the currently approved B&W evaluation model for small break LOCAs in continued conformance with Appendix K to 10 CFR 50 for the case of limited RCP operation after reactor trip and for the range of licensed B&W reactor designs.

We have reviewed industry analyses and performed our own analyses to determine whether RCP trip is necessary during LOCAs, and evaluated the desirability of continued RCP operation during non-LOCA transients and accidents, including steam generator tube ruptures. We have 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, some of the calculations show that for certain small break LOCAs, primarily those with only one of the two High Pressure Safety Injection (HPSI) Pumps assumed available, continued operation of the RCPs or continued operation of the RCPs followed by delayed RCP trip could lead to core damage.

In summary, we have concluded that the need for RCP trip following a transient or accident should be determined by each licensee on a case-by-case basis, considering the Owners Group input. However, the staff must 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 licensees to comply with the Commission's rules and regulations.

We recognize that possible differences exist between the requirements of 10 CFR 50.46, which assure ample core cooling capacity, and the approaches described in the enclosure which are based upon assuring proper operator/ system response under conditions that may be faced during accidents and transients. Accordingly, in such cases, we will consider a request for exemption from specific requirements of 10 CFR 50.46 pursuant to 10 CFR 50.12.

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For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual licensees, we strongly urge that licensees work collectively with owners of similar plants (i.e., owners Group) and propose setpoints and methods consistent with other licensees.

However, once the technical bases for the decision are established, we intend to conduct inspections of individual licensees led by Regional personnel. During these inspections, we will examine the translation of the 10 CFR 50, Appendix K, and RCP operation mode evaluations into plant procedures.

Potential Impact Number: 8330 **SRP Section Number:** 15.6.5

Document: NRC GENERIC LETTER 83-10E
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

We have completed our evaluation of your analyses of LOFT Test L3-6 and conclude that the evaluations acceptably predict the test results. Therefore, we find the currently approved B&W evaluation model for small break LOCAs in continued conformance with Appendix K to 10 CFR 50 for the case of limited RCP operation after reactor trip and for the range of licensed B&W reactor designs.

We have reviewed industry analyses and performed our own analyses to determine whether RCP trip is necessary during LOCAs, and evaluated the desirability of continued RCP operation during non-LOCA transients and accidents, including steam generator tube ruptures. We have 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, some of the calculations show that for certain small break LOCAs, primarily those with only one of the two High Pressure Safety Injection (HPSI) Pumps assumed available, continued operation of the RCPs or continued operation of the RCPs followed by delayed RCP trip could lead to core damage.

In summary, we have concluded that the need for RCP trip following a transient or accident should be determined by each licensee on a case-by-case basis, considering the Owners Group input. However, the staff must 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 licensees to comply with the Commission's rules and regulations.

We recognize that possible differences exist between the requirements of 10 CFR 50.46, which assure ample core cooling capacity, and the approaches described in the enclosure which are based upon assuring proper operator/ system response under conditions that may be faced during accidents and transients. Accordingly, in such cases, we will consider a request for exemption from specific requirements of 10 CFR 50.46 pursuant to 10 CFR 50.12.

For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual licensees, we strongly urge that licensees work collectively with owners of similar plants (i.e., owners Group) and propose setpoints and methods consistent with other licensees.

However, once the technical bases for the decision are established, we intend to conduct inspections of individual licensees led by Regional personnel. During these inspections, we will examine the translation of the 10 CFR 50, Appendix K, and RCP operation mode evaluations into plant procedures.

Potential Impact Number: 8331 **SRP Section Number:** 15.6.5

Document: NRC GENERIC LETTER 83-10F
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

This letter provides guidance for a Babcock and Wilcox NSSS for the development of either satisfactory setpoints for RCP trip or the technical bases for allowing continued RCP operation in the event of a SBLOCA.

Potential Impact Number: 8333 **SRP Section Number:** 15.6.5

Document: NRC GENERIC LETTER 85-12
Title: IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

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Summary:

The purpose of this letter is to inform you of (1) the staff's conclusions regarding the Westinghouse Owners Group (WOG) submittals on reactor coolant pump trip in response to Generic Letters 83-10c and d. and (2) provide guidance concerning implementation of the reactor coolant pump trip criteria. Our Safety Evaluation (SE) on this subject is enclosed for your use.

With regard to the WOG submittals referenced in Section V of the enclosed SE, we conclude that the methods employed by the WOG to justify manual reactor coolant pump (RCP) trip are consistent with the guidelines and criteria provided in Generic Letters 83-10c and d. The approved Westinghouse Small Break LOCA Evaluation Model was used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

We have determined that the Information provided by the WOG in Support of the alternative RCP trip criteria is acceptable on a generic basis. A suitable reactor coolant pump trip criterion can be selected by each licensee to minimize reactor coolant pump trip during steam generator tube ruptures and non-LOCA events, while still providing for RCP trip for small break LOCAs.

In reviewing the WOG RCP trip Criteria we note that the process of criterion selection involves a number of considerations which were assigned plant-specific status by the WOG during the process of the trip criteria review. Accordingly, we request that operating reactor licensees select and implement an appropriate RCP trip criterion based upon the WOG methodology.

Those applicants and licensees who choose not to endorse the WOG methodology should submit a schedule for submittal of plant specific RCP trip criteria or justification for non-trip of RCPs within 45 days of receipt of this letter. .

Potential Impact Number: 8334 **SRP Section Number:** 15.6.5

Document: NRC GENERIC LETTER 86-05
Title: IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

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The purpose of this letter is to inform you of (1) the staff's conclusions regarding the B&W Owners Group (BWOG) submittals on reactor coolant pump trip in response to Generic Letters 83-10e and f, and (2) provide guidance concerning implementation of the reactor coolant pump trip criterion. Our Safety Evaluation (SE) on this subject is enclosed for your use.

With regard to the BWOG submittals referenced in Section V of the enclosed SE, we conclude that the methods employed by the BWOG to justify manual reactor coolant pump (RCP) trip are consistent with the guidelines and criteria provided in Generic Letters 83-10e and f. The approved B&W Small Break LOCA Evaluation Model was used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

We have determined that the information provided by the BWOG in support of the loss-of-subcooling RCP trip criterion is acceptable. The generic information presented by the BWOG, however, does not address plant specific concerns about instrumentation uncertainties, potential reactor coolant pump problems and operator training and procedures as requested in Generic Letter 83-10. This information, contained in Section IV of the SE, is now being requested to assess implementation of the RCP trip criterion. Accordingly, for those applicants and licensees who choose to endorse the BWOG methodology, we request that operating reactor licensees implement the RCP trip criterion based upon the BWOG methodology.

Those applicants and licensees who choose not to endorse the BWOG methodology should submit a schedule for submittal of plant specific RCP trip criteria or justification for non-trip of RCPs within 45 days of receipt of this letter.

Potential Impact Number: 8335 **SRP Section Number:** 15.6.5

Document: NRC GENERIC LETTER 86-06
Title: IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

The purpose of this letter is to inform you of (l) the staff's conclusions regarding the CE Owners Group (CEOG) submittals on reactor coolant pump trip in response to Generic Letters 83-10a and b, and (2) provide guidance concerning implementation of the

reactor coolant pump trip criterion. Our Safety Evaluation (SE) on this subject is enclosed for your use

With regard to the CEOG submittals referenced in Section V of the enclosed SE, we conclude that the methods employed by the CEOG to justify manual reactor coolant pump (RCP) trip are consistent with the guidelines and criteria provided in Generic Letters 83-10a and b. The approved CE Small Break LOCA Evaluation Model was used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

We have determined that the information provided by the CEOG in support of the trip-two/leave-two staggered reactor coolant pump trip criterion is acceptable. The generic information presented by the CEOG, however, does not address plant specific concerns about instrumentation uncertainties, potential reactor coolant pump problems and operator training and procedures as requested in Generic Letter 83-10. This information, contained in Section IV of the SE, is now being requested to assess implementation of the RCP trip criterion. Accordingly, for those applicants and licensees who choose to endorse the CEOG methodology, we request that operating reactor licensees implement the RCP trip criterion based upon the CEOG methodology.

Those applicants and licensees who choose not to endorse the CEOG methodology should submit a schedule for submittal of plant specific RCP trip criteria or justification for non-trip of RCPs within 45 days of receipt of this letter.

Potential Impact Number: 8341 **SRP Section Number:** 15.6.5

Document: RG 1.157 **Revision:** 0
Title: BEST-ESTIMATE CALCULATIONS OF EMERGENCY CORE COOLING SYSTEM PERFORMANCE

Summary:

This Reg. Guide describes 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 during a LOCA and for estimating the uncertainty of the calculation.

Potential Impact Number: 8356 **SRP Section Number:** 15.6.5

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Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix K

Summary:

Appendix K to 10CFR50 details the required and acceptable features of ECCS evaluation models for LOCA and the required documentation for those models.

Potential Impact Number: 8508 **SRP Section Number:** 11.3

Document: NRC GENERIC LETTER 89-01
Title: IMPLEMENTATION OF PROGRAMMATIC CONTROLS FOR RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS IN THE ADMINISTRATIVE CONTROLS SECTION OF THE TECHNICAL SPECIFICATIONS AND THE RELOCATION OF PROCEDURAL DETAILS OF RETS TO THE OFFSITE DOSE CALCULATION MANUAL OR

Summary:

Provides model changes to radiological effluent technical specifications for explosive gas monitoring instrumentation of gas effluent systems.

Potential Impact Number: 8607 **SRP Section Number:** 8.2

Document: RG 1.32 **Revision:** 2
Title: CRITERIA FOR SAFETY-RELATED ELECTRIC POWER SYSTEMS FOR NUCLEAR POWER PLANTS

Summary:

The RG provides guidance for offsite power system (and onsite systems, as well) compliance with IEEE Std 308-1974 and Criteria 17 and 18. The RG is referenced as guidance for SRP Section 8.2 in Sections 8.1 and 8.2

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Potential Impact Number: 8655 **SRP Section Number:** 8.2

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

This guide primarily addresses the following three areas: (1) maintaining highly reliable ac electric power systems, (2) developing procedures and training to restore offsite and onsite emergency ac power should either one or both become unavailable, and (3) ensuring that plants can cope with a station blackout for some period of time based on the probability of occurrence of a station blackout at a site as well as the capability for restoring ac power in a timely fashion for that site.
Provides guidance for complying with 10CFR50.63, Loss of All Alternating Current Power. Also establishes target reliability levels for electric power systems, and guidance for power restoration emergency procedures.

Potential Impact Number: 8739 **SRP Section Number:** 11.4

Document: NRC GENERIC LETTER 84-12
Title: COMPLIANCE WITH 10 CFR PART 61 AND IMPLEMENTATION OF THE RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS (RETS) AND ATTENDANT PROCESS CONTROL PROGRAM (PCP)

Summary:

During the development of Part 61, the NRC staff determined that compliance with the radioactive waste form requirements of Part 61 and the certification requirements of 10 CFR 20.311 could be achieved by the development and use of a Process Control Program (PCP) as an attendant part of the licensee's Radiological Effluent Technical Specifications (RETS). This approach was determined to be acceptable by the responsible state regulatory agencies that license the disposal sites. It is now apparent, however, that many licensees do not yet have approved PCPs and that no licensee has a PCP which specifically addresses the new requirements of Part 61.

Potential Impact Number: 8869 **SRP Section Number:** 6.1.1

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Document: RG 1.37 **Revision:** 0
Title: QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER COOLED NUCLEAR POWER PLANTS

Summary:

This guide describes an acceptable method of complying with the Commission's regulations with regard to quality assurance requirements for on-site cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water-cooled nuclear power plant fluid systems. The requirements and recommendations for on-site cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water- cooled nuclear power plant fluid systems that are included in ANSI N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of 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 regulatory positions (which include application of the standard to the operations phase).

Potential Impact Number: 8872 **SRP Section Number:** 6.1.1

Document: RG 1.44 **Revision:** 0
Title: CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL

Summary:

This guide describes acceptable methods of implementing the Commission's requirements/regulations with regard to control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking. This guide applies to light-water-cooled reactors. Unstabilized, austenitic stainless steel of the AISI Type 3XX series used for components that are part of (1) the reactor coolant pressure boundary, (2) systems required for reactor shutdown, (3) systems required for emergency core cooling, and (4) reactor vessel internals that are relied upon to permit adequate core cooling for any mode of normal operation or under credible postulated accident conditions should meet the criteria provided in the regulatory positions.

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Potential Impact Number: 8988 **SRP Section Number:** 6.1.1

Document: NRC BULLETIN 82-02
Title: DEGRADATION OF THREADED FASTENERS IN THE REACTOR COOLANT PRESSURE BOUNDARY OF PWR PLANTS

Summary:

Discusses numerous failure causes for threaded fasteners due to chemistry incompatibilities.

Potential Impact Number: 9049 **SRP Section Number:** 8.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.63

Summary:

The station blackout rule for light water reactors. Requires that all reactors be capable of withstanding and recovering from a complete loss of all AC (except for inverter power) power of a specified duration. The duration shall be based on several factors, including the reliability of offsite and onsite power systems.

Potential Impact Number: 9135 **SRP Section Number:** 3.8.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.34, (f)(3)(v)(B)(1)

Summary:

Containment structure loadings produced by inadvertent full activation of a post-accident inerting hydrogen system, but not including seismic or design basis accident loadings, will not produce stresses in containment in excess of limits set forth

in ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, except that instability evaluation is not required.

Potential Impact Number: 9136 **SRP Section Number:** 3.8.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.34, (f)(3)(v)(A)(1)

Summary:

Containment integrity must be maintained by meeting requirements of ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220.

Potential Impact Number: 9187 **SRP Section Number:** 10.3.6

Document: NRC GENERIC LETTER 89-08
Title: EROSION/CORROSION-INDUCED PIPE WALL THINNING

Summary:

The NRC issued six information notices (86-106 and Supplements 1, 2, and 3; 87-36, and 88-17) and Bulletin 87-01 addressing erosion/corrosion of piping systems. The staff review of licensees' responses to the bulletin indicates that the pipe wall thinning problem is widespread for single-phase and two-phase high-energy carbon steel systems. The staff review also showed that wall thinning in single phase feedwater-condensate systems is more prevalent among pressurized-water reactors (PWRs) but also occurs in boiling water reactors (BWRs). Additional insight into the phenomena related to erosion/corrosion of carbon steel components is provided in the enclosure to this letter (NUREG-1344).

Potential Impact Number: 9245 **SRP Section Number:** 3.10

Document: RG 1.100 **Revision:** 2

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Title: SEISMIC QUALIFICATION OF ELECTRIC AND MECHANICAL EQUIPMENT FOR NUCLEAR POWER PLANTS

Summary:

This regulatory guide describes a method acceptable to the NRC staff for complying with NRC's regulations with respect to seismic qualification of electric and mechanical equipment. The procedures described by IEEE Std 344-1987, Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, are acceptable to the NRC staff for satisfying the Commission's regulations pertaining to seismic qualification of electric and mechanical equipment subject to the modifications described in the regulatory positions. Modifications include application of the standard to mechanical equipment.

Potential Impact Number: 9255 **SRP Section Number:** 6.2.1.5

Document: RG 1.157 **Revision:** 0
Title: BEST-ESTIMATE CALCULATIONS OF EMERGENCY CORE COOLING SYSTEM PERFORMANCE

Summary:

This regulatory guide describes 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 during a loss-of-coolant accident and for estimating the uncertainty in that calculation. Methods for including the uncertainty in the comparisons of the calculational results to the criteria of paragraph 50.46(b), in order to meet the requirement that there be a high probability that the criteria would not be exceeded, are also described in this regulatory guide. Paragraph 50.46(a) also permits licensees to use evaluation models developed in conformance with Appendix K.

Paragraph C.3.12.1 states:

The containment pressure used for evaluating cooling effectiveness during the post-blowdown phase of a loss-of-coolant accident should be calculated in a best-estimate manner and should include the effects of containment heat sinks. The calculation should include the effects of operation of all pressure reducing equipment

assumed to be available. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

Potential Impact Number: 9354 **SRP Section Number:** 6.2.1.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.46

Summary:

Each boiling and pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy cladding must be provided with an emergency core cooling system (ECCS) that must be designed such that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. The evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of Appendix K ECCS Evaluation Models.

Potential Impact Number: 9369 **SRP Section Number:** 12.1

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION
Location: 10CFR20.1003

Summary:

ALARA defined in 20.1003.

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Potential Impact Number: 9375 **SRP Section Number:** 12.1

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION
Location: 20.1101 (b)

Summary:

The licensee shall use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

Potential Impact Number: 9463 **SRP Section Number:** 10.3.6

Document: RG 1.44 **Revision:** 0
Title: CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL

Summary:

This guide describes acceptable methods of implementing the Commission's requirements/regulations with regard to control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking. This guide applies to light-water-cooled reactors. Unstabilized, austenitic stainless steel of the AISI Type 3XX series used for components that are part of (1) the reactor coolant pressure boundary, (2) systems required for reactor shutdown, (3) systems required for emergency core cooling, and (4) reactor vessel internals that are relied upon to permit adequate core cooling for any mode of normal operation or under credible postulated accident conditions should meet the criteria provided in the regulatory positions.

Potential Impact Number: 9469 **SRP Section Number:** 10.3.6

Document: RG 1.37 **Revision:** 0
Title: QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER COOLED NUCLEAR POWER PLANTS

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Summary:

This guide describes an acceptable method of complying with the Commission's regulations with regard to quality assurance requirements for on-site cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water-cooled nuclear power plant fluid systems. The requirements and recommendations for on-site cleaning of materials and components, cleanliness control, and preoperational cleaning and layup of water-cooled nuclear power plant fluid systems that are included in ANSI N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of 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 regulatory positions (which include application of the standard to the operations phase).

Potential Impact Number: 9748 **SRP Section Number:** 4.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix K

Summary:

Assumptions, approximations, and approaches for evaluating LOCA's (and fuel damage) are provided.

Potential Impact Number: 9777 **SRP Section Number:** 4.2

Document: RG 1.157 **Revision:** 0
Title: BEST-ESTIMATE CALCULATIONS OF EMERGENCY CORE COOLING SYSTEM PERFORMANCE

Summary:

On September 16, 1988, the NRC staff amended the requirements of 10 CFR 50.46 and Appendix K to Part 50 so that these regulations reflect the improved understanding of ECCS performance during reactor transients that was obtained through the

extensive research performed since the promulgation of the original requirements in January 1974. Paragraph 50.46(a)(1) now permits licensees or applicants to use either Appendix K assumptions or a realistic evaluation model.

This Regulatory guide describes 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 during a loss-of-coolant accident and for estimating the uncertainty in that calculation. Methods for including the uncertainty in the comparisons of the calculational results to the criteria of paragraph 50.46(b), in order to meet the requirement that there be a high probability that the criteria would not be exceeded, are also described in this regulatory guide.

Potential Impact Number: 9803 **SRP Section Number:** 6.2.2

Document: RG 1.82 **Revision:** 1
Title: WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT

Summary:

'The primary areas of safety concern regarding ECC sumps and pump inlets are (1) post-LOCA hydraulic effects, particularly air ingestion, (2) blockage of debris interceptors resulting from LOCA destruction of insulation and its transport, and (3) the combined effects of items (1) and (2) relative to recirculation pumping operability (i.e., impact on net positive suction head (NPSH) available at the pump inlet).'

Potential Impact Number: 9825 **SRP Section Number:** 8.3.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.63

Summary:

The station blackout rule for light water reactors. Requires that all reactors be capable of withstanding and recovering from a complete loss of all AC (except for inverter

power) power of a specified duration. The duration is to determined based upon several factors including the reliability of offsite and onsite power systems.

Potential Impact Number: 9887 **SRP Section Number:** 3.8.1

Document: RG 1.35.1 **Revision:** 0
Title: DETERMINING PRESTRESSING FORCES FOR INSPECTION OF PRESTRESSED CONCRETE CONTAINMENTS

Summary:

As a supplement to Reg. Guide 1.35, This guide expands and clarifies the NRC staff on determining prestressing forces to be used for in service inspections of prestressed concrete containment structures.

Potential Impact Number: 9888 **SRP Section Number:** 4.4

Document: NRC BULLETIN 88-07
Title: POWER OSCILLATIONS IN BOILING WATER REACTORS (BWRs)

Summary:

Foreign reactor experience has indicated that there are conditions other than the natural circulation condition that existed during the LaSalle event, such as loss of feedwater heaters, which can lead to power oscillations. Furthermore, the power oscillations may be asymmetric and lead to high local neutron flux levels without detection and automatic scram. Therefore, power oscillations are of safety concern because they may result in fuel damage.

Potential Impact Number: 9948 **SRP Section Number:** 8.3.1

Document: RG 1.9
Title: SELECTION, DESIGN, AND QUALIFICATION OF DIESEL-GENERATOR UNITS USED AS STANDBY (ONSITE) ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

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Summary:

Provides guidance for the selection, design, and qualification of diesel-generator units used as standby (onsite) electric power systems at nuclear power plants. The RG is referenced as guidance for SRP Section 8.3.1 in both Sections 8.1 and 8.3.1.

Potential Impact Number: 10030 **SRP Section Number:** 3.8.1

Document: RG 1.136 **Revision:** 2
Title: MATERIALS, CONSTRUCTION, AND TESTING OF CONCRETE CONTAINMENTS (Articles CC-1000, -2000, and 4000 through 6000 of the Code for Concrete Reactor Vessels and Containments)

Summary:

Describes bases acceptable for implementing quality standard requirements with respect to materials, construction, and testing of concrete containments.

REG 1.10, 1.15, 1.18, 1.19, 1.55, and 1.103 were withdrawn on July 8, 1981 and are partially endorsed by REG 1.136.

Potential Impact Number: 10149 **SRP Section Number:** 9.2.2

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

If a system is required for primary coolant charging and makeup, reactor coolant pump seal cooling or injection, decay heat removal, or maintaining appropriate containment integrity specifically to meet the station blackout duration recommended in Regulatory Position 3.1, the following criteria should be met:

1. The system 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

2. 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.

Potential Impact Number: 10214 **SRP Section Number:** 8.3.1

Document: RG 1.108 **Revision:** 1
Title: PERIODIC TESTING OF DIESEL GENERATOR UNITS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

Summary:

Provides guidance for the periodic testing of diesel generators used as onsite standby emergency power sources for nuclear power plants. Provides criteria for valid tests and failed tests and recommended test procedures for surveillance testing. Also provides guidance for the design of testability features, test mode emergency override features, and protective trip emergency bypass features. The RG is cited as guidance for SRP Section 8.3.1 in Sections 8.1 and 8.3.1.

Potential Impact Number: 10219 **SRP Section Number:** 8.3.1

Document: RG 1.153 **Revision:** 0
Title: CRITERIA FOR POWER, INSTRUMENTATION, AND CONTROL

Summary:

Provides guidance for the design, reliability, qualification, and testability of power, instrumentation, and control portions of safety systems for nuclear power plants. Endorses and modifies standard IEEE-603-1980. Also endorses portions of RG 1.97 and IEEE-279. The RG is not cited as guidance for SRP Section 8.3.1 in Section 8.1 or 8.3.1.

Potential Impact Number: 10220 **SRP Section Number:** 8.3.1

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

APPENDIX II
Potential Impacts

Summary:

Provides guidance for compliance with the station blackout rule for light water reactors (10CFR50.63). Covers such topics as standby emergency power source target reliabilities and analyses to determine an appropriate specified duration of blackout for which withstand capability must be demonstrated. The RG is not cited as guidance for SRP Section 8.3.1 in Section 8.1 or 8.3.1.

Potential Impact Number: 10386 **SRP Section Number:** 9.2.2

Document: NRC GENERIC LETTER 91-07
Title: GI-23, 'REACTOR COOLANT PUMP SEAL FAILURES' AND ITS POSSIBLE EFFECT ON STATION BLACKOUT

Summary:

This letter informs licensees of the possible effect of GI-23 on their responses to the station blackout rule, 10 CFR 50.63.

In resolving the SBO issue (USI A-44), the NRC staff considered the relationship of this issue with other NRC generic issues, including GI-23. In determining estimates of core damage frequency for SBO events the staff assumed that the RCP seals would leak at a rate of 20 gallons per minute (gpm) per pump. During the final resolution of the station blackout issue and the development of NUMARC Report 87-00, NUMARC and the staff agreed to assume that RCP seal leakage during an SBO would be no greater than 25 gpm per pump. Further, NUMARC and the staff agreed that if the final resolution of GI-23 results in higher RCP leakage rates, then the SBO analyses would have to be reevaluated.

The staff is soliciting public comments on its current understanding of GI-23. One possible outcome may be that seal cooling be provided by an independent cooling system during off-normal plant conditions involving the loss of all seal cooling, such as could occur during an SBO. This recommendation, if adopted, would apply to all PWRs and might apply to the above-mentioned BWRs, which do not have AC-independent makeup capabilities for their reactor coolant systems. Therefore, utilities should recognize that such a recommendation could affect their analyses and actions addressing conformance to the SBO rule.

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Potential Impact Number: 10405 **SRP Section Number:** 13.3

Document: RG 1.101 **Revision:** 2
Title: EMERGENCY PLANNING AND PREPAREDNESS FOR NUCLEAR POWER REACTORS

Summary:

This Reg. Guide endorses NUREG.0654 guidance on meeting the requirements of 10CFR50.47 and Appendix E.

Potential Impact Number: 10520 **SRP Section Number:** 9.4.5

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

This regulatory guide describes a method acceptable for complying with the requirements with potential habitability concerns for those areas that would require operator access during the station blackout and recovery period.

Evaluations that have already been performed need not be duplicated. For example, if safety-related equipment required during a total loss of ac power has been qualified to operate under environmental conditions exceeding those expected under a station blackout (e.g., loss of heating, ventilation, and air conditioning), additional analyses need not be performed. Equipment will be considered acceptable for station blackout temperature environments if an assessment has been performed that provides reasonable assurance that the required equipment will remain operable.

Potential Impact Number: 10600 **SRP Section Number:** 3.2.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: 10 CFR 50, Appendix R, Section III.L.6, 10 CFR 50, Appendix R, Section III.L.6, 10 CFR 50 Appendix R, Section III.O

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Summary:

Shutdown systems installed to ensure postfire shutdown capability need not be designed to meet seismic Category I criteria, single failure criteria, or other design basis accident criteria, except where required for other reasons, e.g., because of interface with or impact on existing safety systems, or because of adverse valve actions due to fire damage. The reactor coolant pump shall be equipped with an oil collection system if the containment is not inerted during normal operation. The oil collection system shall be designed, engineered, and installed such that failure will not lead to fire during normal or design basis accident conditions and that there is reasonable assurance that the system will withstand the Safe Shutdown Earthquake (see Reg. Guide 1.29).

Potential Impact Number: 10607 **SRP Section Number:** 3.2.1

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Locations: Appendix A, III (c), Appendix A, III (c), Appendix A, III (d)

Summary:

This part describes criteria which guide the Commission in its evaluation of the suitability of proposed sites for stationary power and testing reactors subject to Part 50 of this chapter. In particular, the following definitions of structures, systems and components that must resist earthquakes are provided. The Safe Shutdown Earthquake is that earthquake which produces the maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional. These structures, systems, and components are those necessary to assure: (1) The integrity of the reactor coolant pressure boundary, (2) The capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part. The Operating Basis Earthquake is that earthquake which produces the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional.

Potential Impact Number: 10673 **SRP Section Number:** 3.2.1

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Document: RG 1.26 **Revision:** 3
Title: QUALITY GROUP CLASSIFICATIONS AND STANDARDS FOR WATER-, STEAM-, AND RADIOACTIVE-WASTE-CONTAINING COMPONENTS OF NUCLEAR POWER PLANTS

Summary:

Systems which should be classified as Quality Group C include those other than radioactive waste management systems, not covered by items C.2.a. through C.2.c. that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses that exceed 0.5 rem to the whole body or its equivalent to any part of the body. If located in Seismic Category I structures, only single component failures need be assumed (in these dose calculations). No credit for automatic isolation from other components in the system or for treatment of released material should be taken unless the isolation or treatment capability is designed to the appropriate seismic and quality group standards and can withstand loss of offsite power and a single failure of an active component.

Potential Impact Number: 10707 **SRP Section Number:** 3.2.1

Document: RG 1.29 **Revision:** 3
Title: SEISMIC DESIGN CLASSIFICATION

Summary:

This guide describes a method acceptable to the NRC staff for identifying and classifying those features of light-water-cooled nuclear power plants that should be designed to withstand the effects of the SSE. Those structures, systems, and components that should be designed to remain functional if the SSE occurs have been designated as Seismic Category 1, and are identified in the Guide. The pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 should be applied to all activities affecting the safety-related functions of these structures, systems, and components. The Guide is based on Appendix A, Seismic and Geologic Siting Criteria for Nuclear Power Plants, to 10 CFR Part 100, that requires that all nuclear power plants be designed so that, if the SSE occurs, certain structures, systems, and components remain functional. These plant features are those necessary to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite

exposures comparable to the guideline exposures of 10 CFR Part 100. This Regulatory Guide also includes a position that systems be designated safety-related if they are systems, other than radioactive waste management systems, that 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.

Potential Impact Number: 10747 **SRP Section Number:** 13.3

Document: NRC GENERIC LETTER 82-33
Title: SUPPLEMENT 1 TO NUREG-0737 - REQUIREMENTS FOR EMERGENCY RESPONSE CAPABILITY

Summary:

This letter promulgates additional guidance (clarification) regarding safety parameter display systems, control room design reviews, Reg. Guide 1.97 application to emergency response facilities, upgrade of EOP's emergency response facilities and meteorological data.

During PRB review of Chapter 7 HICB disapproved this potential impact for inclusion in Chapter 7. The impact of this generic letter and NUREG-0737 Supplement 1 on SRP Section 13.3 given the current citation of TMI action plan item III.A.1.2 must be evaluated.

NUREG-0737 Supplement 1 Microfiche No. 17301:324

Potential Impact Number: 10898 **SRP Section Number:** 3.2.1

Document: RG 1.143 **Revision:** 1
Title: DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

Summary:

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This guide furnishes design guidance acceptable to the NRC staff relating to seismic and quality group classification and quality assurance provisions for radioactive waste management systems, structures, and components. Regulatory Position C.5 (and its subsections) provides seismic design analysis methodology based upon using floor response spectra normalized to the OBE. Many other regulatory positions in the guide indicate which structures, systems, and components (SSCs) should (or should not) be seismically designed in accordance with position C.5. Note that position C.5 does not directly define a limiting earthquake with characteristics (e.g. seismic acceleration) which SSCs must withstand and remain functional.

Potential Impact Number: 10906 **SRP Section Number:** 3.2.1

Document: RG 1.151 **Revision:** 0 , 7/1/83
Title: INSTRUMENT SENSING LINES

Summary:

This Guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to the design and installation of safety-related instrument sensing lines in nuclear power plants. Seismic Classification criteria are included in regulatory positions C.2 and C.3. The term safety-related refers to those structures, systems, and components necessary to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100. This definition is also in Reg. Guide 1.29.

Potential Impact Number: 10988 **SRP Section Number:** 3.2.1

Document: NRC GENERIC LETTER 86-10
Title: IMPLEMENTATION OF FIRE PROTECTION REQUIREMENTS

Summary:

Provides guidance for seismic design for fire protection features. Reference is made to Reg. Guide 1.29 for seismic classification of such features.

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Potential Impact Number: 11023 **SRP Section Number:** 3.7.3

Document: RG 1.29 **Revision:** 3
Title: SEISMIC DESIGN CLASSIFICATION

Summary:

Seismic Category I design requirements should extend to the first seismic restraint beyond the defined boundaries. Those portions of structures, systems, or components that form interfaces between Seismic Category I and non-Seismic Category I features should be designed to Seismic Category I requirements.

Potential Impact Number: 11071 **SRP Section Number:** 4.5.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Pipe cracking may occur in heat-affected zones of welds in primary system piping. These cracks occur mainly in austenitic stainless steels which have been sensitized by post-weld heat treatment or by the heating due to the welding processes, and have been found to follow the stress corrosion cracking mechanism. This issue was resolved through the publication of NUREG-0313 and the establishment of NPA-B-05.

Potential Impact Number: 11072 **SRP Section Number:** 4.5.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

GSI A-42 addresses pipe cracking which may occur in heat-affected zones of welds in primary system piping. These cracks occur mainly in austenitic stainless steels which

have been sensitized by post-weld heat treatment or by the heating due to the welding processes, and have been found to follow the stress corrosion cracking mechanism. This issue was resolved through the publication of NUREG-0313 and the establishment of NPA-B-05.

New Generic Issue 119.4 responds to a Piping Review Committee recommendation that regular grade 316 SS and 304 SS materials in BWR recirculation piping be replaced with an alloy which is resistant to intergranular stress corrosion cracking. This item is a regulatory impact issue that recommends updating of Regulatory Guide 1.44 to reflect the staff's findings in NUREG-0313, Rev. 2.

Potential Impact Number: 11074 **SRP Section Number:** 5.2.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Pipe cracking may occur in heat-affected zones of welds in primary system piping. These cracks occur mainly in austenitic stainless steels which have been sensitized by post-weld heat treatment or by the heating due to the welding processes, and have been found to follow the stress corrosion cracking mechanism. This issue was resolved through the publication of NUREG-0313 and the establishment of NPA-B-05.

Potential Impact Number: 11075 **SRP Section Number:** 5.2.4

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Pipe cracking has occurred in the heat-affected zones of welds in primary system piping in BWRs since mid-1960. These cracks have occurred mainly in Type 304 stainless steel which is the type used in most operating BWRs. The major problem is recognized to be IGSCC of austenitic stainless steel components that have been made susceptible to this failure by being "sensitized," either by post-weld heat treatment or by sensitization of a narrow heat affected zone near welds.

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This USI was RESOLVED in February 1981 when NUREG-0313, Revision 1 was issued to all holders of BWR operating licenses or construction permits and to all applicants for BWR operating licenses. MPA B-05 was established by DL for implementation of the resolution at operating plants.

Potential Impact Number: 11076 **SRP Section Number:** 5.3.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Pipe cracking may occur in heat-affected zones of welds in primary system piping. These cracks occur mainly in austenitic stainless steels which have been sensitized by post-weld heat treatment or by the heating due to the welding processes, and have been found to follow the stress corrosion cracking mechanism. This issue was resolved through the publication of NUREG-0313 and the establishment of NPA-B-05.

Potential Impact Number: 11083 **SRP Section Number:** 6.1.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Pipe cracking may occur in heat-affected zones of welds in primary system piping. These cracks occur mainly in austenitic stainless steels which have been sensitized by post-weld heat treatment or by the heating due to the welding processes, and have been found to follow the stress corrosion cracking mechanism. This issue was resolved through the publication of NUREG-0313 and the establishment of NPA-B-05.

Potential Impact Number: 11084 **SRP Section Number:** 6.6

Document: NUREG 0933

APPENDIX II
Potential Impacts

Title: A Prioritization of Generic Safety Issues

Summary:

Pipe cracking may occur in heat-affected zones of welds in primary system piping. These cracks occur mainly in austenitic stainless steels which have been sensitized by post-weld heat treatment or by the heating due to the welding processes, and have been found to follow the stress corrosion cracking mechanism. This issue was resolved through the publication of NUREG-0313 and the establishment of MPA-B-05.

Potential Impact Number: 11085 **SRP Section Number:** 10.3.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Pipe cracking may occur in heat-affected zones of welds in primary system piping. These cracks occur mainly in austenitic stainless steels which have been sensitized by post-weld heat treatment or by the heating due to the welding processes, and have been found to follow the stress corrosion cracking mechanism. This issue was resolved through the publication of NUREG-0313 and the establishment of NPA-B-05.

Potential Impact Number: 11120 **SRP Section Number:** 5.4.7

Document: NRC BULLETIN 80-12
Title: DECAY HEAT REMOVAL SYSTEM OPERABILITY

Summary:

Recent losses of DHR at operating PWR's dictate that positive actions be taken to preclude their occurrence or at least ameliorate their effects (this document contains several action items to be taken).

Potential Impact Number: 11130 **SRP Section Number:** 5.4.7

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Document: NRC GENERIC LETTER 87-12
Title: LOSS OF RESIDUAL HEAT REMOVAL (RHR) WHILE THE REACTOR COOLANT SYSTEM (RCS) IS PARTIALLY FILLED

Summary:

Pursuant to 10 CFR 50.54(f), the NRC is requesting information to assess safe operation of pressurized-water reactors (PWRs) when the reactor coolant system (RCS) water level is below the top of the reactor vessel (RV). The principal concerns are (1) whether the RHR system meets the licensing basis of the plant, such as General Design Criterion 34 (10 CFR Part 50, Appendix A) and Technical Specifications (TS), in this condition; (2) whether there is a resultant unanalyzed event that may have an impact upon safety; and (3) whether any threat to safety that warrants further NRC attention exists in this condition.

Potential Impact Number: 11131 **SRP Section Number:** 5.4.7

Document: NRC GENERIC LETTER 88-17
Title: LOSS OF DECAY HEAT REMOVAL

Summary:

Loss of decay heat removal (DHR) during nonpower operation and the consequences of such a loss have been of increasing concern for years. Numerous industry and NRC publications have addressed the subject. The Diablo Canyon event of April 10, 1987, and ensuing work by both the staff and industry organizations have provided additional insight. Yet the problems continue, as illustrated by (1) the inadequacies demonstrated by many licensees in their response to Generic Letter (GL) 87-12; (2) the event at Waterford on May 12, 1988; (3) the event at Sequoyah on May 23, 1988; (4) the DHR perturbations due to inadequate level at San Onofre on July 7, 1988; and (5) the apparent lack of a complete industry understanding of the potential seriousness of such events.

Potential Impact Number: 11153 **SRP Section Number:** 3.7.1

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA

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Location: 10CFR100 Appendix A V (a) (2)

Summary:

The maximum vibratory ground acceleration of the Operating Basis Earthquake shall be at least one-half the maximum vibratory ground acceleration of the Safe Shutdown Earthquake.

Potential Impact Number: 11158 **SRP Section Number:** 5.4.7

Document: NRC GENERIC LETTER 90-06
Title: 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,'

Summary:

Generic Letter 90-06 forwarded staff positions related to the resolution of Generic Issue 70. Enclosure A to the Generic Letter presents the staff positions resulting from the resolution of Generic Issue 70 related to PWR PORVs. PORVs are now relied upon by many PWRs to perform one, or more, of the following safety-related functions:

1. Mitigation of a design-basis steam generator tube rupture accident,
2. Low-temperature overpressure protection of the reactor vessel during startup and shutdown, or
3. Plant cooldown in compliance with Branch Technical Position RSB 5-1 to SRP 5.4.7, "Residual Heat Removal (RHR) System."

Where PORVs are used or could be used to perform one, or more, of the safety related functions identified above or to perform any other safety-related function that may be identified in the future, it is appropriate to reconsider the safety classification of PORVs and the associated block valves. for certain PWR plants receiving an operating license in recent years, the staff has required these valves to be classified as safety-related components if they perform one, or more, safety-related functions.

Potential Impact Number: 11159 **SRP Section Number:** 15.8

APPENDIX II
Potential Impacts

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.62

Summary:

Defines ATWS and specifies requirements to reduce risks from ATWS events (on an NSSS vendor-specific basis) which are applicable to all light-water-cooled nuclear power plants.

The final rule, including a discussion of how it was developed and acceptable design criteria for required preventive and mitigative features, are published in 49 FR 26036 through 26045.

Potential Impact Number: 11160 **SRP Section Number:** 3.7.1

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: 10CFR100 Appendix A VI (a) (2)

Summary:

Operating Basis Earthquake. The Operating Basis Earthquake shall be defined by response spectra. All structures, systems, and components of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public shall be designed to remain functional and within applicable stress and deformation limits when subjected to the effects of the vibratory motion of the Operating Basis Earthquake in combination with normal operating loads. The analysis or test shall take into account soil-structure interaction effects and the expected duration of vibratory motion.

Potential Impact Number: 11220 **SRP Section Number:** 15.8

Document: NRC BULLETIN 88-07
Title: POWER OSCILLATIONS IN BOILING WATER REACTORS (BWRs)

APPENDIX II
Potential Impacts

Summary:

The recirculation pump trip used as part of the response to an ATWS event at BWR's could possibly provoke power oscillations similar to the La Salle event. The implications, if any, on the bounds for neutron flux oscillations considered in previous safety analyses, including peak power resulting from asymmetric oscillations and the effects of recirculation pump trip actions for ATWS events, require further study.

Potential Impact Number: 11308 **SRP Section Number:** 5.4.7

Document: NRC BULLETIN 86-01
Title: MINIMUM FLOW LOGIC PROBLEMS THAT COULD DISABLE RHR PUMPS

Summary:

The purposes of this bulletin are: (1) to inform BWR licensees and applicants of a recently identified problem with the minimum flow logic for which a single failure could disable all RHR pumps, (2) to request that licensees affected by the problem immediately provide appropriate instructions and training to plant operators on how to recognize the problem if it occurs and take appropriate mitigating actions, (3) to request that licensees notify the NRC of the existence of the problem at their facility within 7 days of receipt of this bulletin, and (4) to request that licensees inform the NRC of measures taken to correct design or installation problems that are identified as a result of this bulletin.

Potential Impact Number: 11310 **SRP Section Number:** 5.4.7

Document: NRC BULLETIN 88-04
Title: POTENTIAL SAFETY-RELATED PUMP LOSS

Summary:

The purpose of this bulletin is to request all licensees to investigate and correct as applicable two miniflow design concerns. The first concern involves the potential for the dead-heading of one or more pumps in safety-related systems that have a miniflow line common to two or more pumps or other piping configurations that do not preclude

pump-to-pump interaction during miniflow operation. A second concern is whether or not the installed miniflow capacity is adequate for even a single pump in operation.

Potential Impact Number: 11322 **SRP Section Number:** 5.4.7

Document: NRC GENERIC LETTER 90-04
Title: REQUEST FOR INFORMATION ON THE STATUS OF LICENSEE IMPLEMENTATION OF GENERIC SAFETY ISSUES RESOLVED WITH IMPOSITION OF REQUIREMENTS OR CORRECTIVE ACTIONS

Summary:

This issue concerns the inadvertent closing of RHR suction valves when the system is in use.

Interlocks are provided on these valves to ensure that a double barrier (i.e., two closed valves) is maintained between the RCS and RHR systems when the plant is at normal operating conditions. However, the loss of one instrument bus or disturbance of one logic channel will result in the automatic closure of one of the RHR suction line isolation valves. Such closure gives rise to the potential for RHR pump damage and loss of decay heat removal capability if the RHR pump is not interlocked with the RHR suction valves.

Potential Impact Number: 11439 **SRP Section Number:** 3.8.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Criterion 4

Summary:

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

APPENDIX II
Potential Impacts

Potential Impact Number: 11487 **SRP Section Number:** 11.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Section 50.34a, (a)-(b)

Summary:

Discusses design objectives, requirements associated with levels of radioactive materials in effluents to unrestricted areas ALARA (references Appendix I).

Potential Impact Number: 11582 **SRP Section Number:** 11.5

Document: RG 4.15 **Revision:** 1
Title: QUALITY ASSURANCE FOR RADIOLOGICAL MONITORING PROGRAMS (NORMAL OPERATIONS)--EFFLUENT STREAMS AND THE ENVIRONMENT

Summary:

Quality control for continuous effluent monitoring systems. References ANSI N13.10-1974 (Reference 18).

Potential Impact Number: 11601 **SRP Section Number:** 11.5

Document: NRC GENERIC LETTER 89-01
Title: IMPLEMENTATION OF PROGRAMMATIC CONTROLS FOR RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS IN THE ADMINISTRATIVE CONTROLS SECTION OF THE TECHNICAL SPECIFICATIONS AND THE RELOCATION OF PROCEDURAL DETAILS OF RETS TO THE OFFSITE DOSE CALCULATION MANUAL OR

Summary:

Discusses how certain programmatic controls will divide some of the items in the radiological effluent technical specifications and relocate them to the Offsite Dose Calculation Manual or the Process Control Program.

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Potential Impact Number: 11625 **SRP Section Number:** 11.5

Document: RG 4.8 **Revision:** 0
Title: Environmental Technical Specifications for Nuclear Power Plants

Summary:

Refers to Appendix I of 10CFR50, for a guide to requirements for radioactive materials in effluents released to unrestricted areas being as low as practicable (ALARA).

When the Commission published Appendix I, the NRC staff began developing models and calculational procedures for estimating exposures to individuals that will provide a basis for demonstrating compliance with design objectives. When this work is completed, model technical specifications for radioactive effluents from light-water-cooled nuclear reactors and for radiological environmental monitoring programs will be published as an addendum to this regulatory guide.

Potential Impact Number: 11719 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR08
Title: INTERPRETATIONS
Location: 8.5(b)

Summary:

Interprets 10CFR73.55(c)(5) concerning illumination of isolation zones and exterior areas within protected areas.

This revised Potential Impact supersedes Potential Impact 11722.

Potential Impact Number: 11724 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR08
Title: INTERPRETATIONS

APPENDIX II
Potential Impacts

Location: 8.5(d)

Summary:

Interprets 10CFR73.55(d) and specifically 73.55(d)(1) concerning requirements for physical searches pending availability of equipment capable of detecting firearms and explosive devices.

Potential Impact Number: 11752 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.34(e) and 10CFR50.54(v)

Summary:

Physical security safeguards contingency and guard qualification and training plans shall be protected against unauthorized disclosure in accordance with 10CFR73.21.

Potential Impact Number: 11753 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.54(p)

Summary:

Provides requirements for safeguards contingency plan and procedures.

Potential Impact Number: 11754 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.54(p)(3)

APPENDIX II
Potential Impacts

Summary:

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.

Potential Impact Number: 11759 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: 10CFR73.2

Summary:

Provides pertinent definitions used in 10CFR73

Potential Impact Number: 11770 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: 10CFR73.55(b)

Summary:

Describes the requirements for a physical security organization.

Potential Impact Number: 11774 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: 73.55(c)

Summary:

Requires: (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 a 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; and, (6) bullet resistant barriers around the control room.

This revised Potential Impact supersedes Potential Impacts 11785, 11789, 11791, 11794, and 11796.

Potential Impact Number: 11807 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: 10CFR73.55(d)

Summary:

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.

This revised Potential Impact supersedes Potential Impacts 11811, 11813, 11814 and 11815.

Potential Impact Number: 11816 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: 10CFR73.55(e)

Summary:

Requirements for: (1) alarms and alarm stations; (2) tamper resistance and self-checking of alarm devices; and, (3) alarming of emergency exits.

Potential Impact Number: 11817 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: 10CFR73.55(f)

Summary:

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.

Potential Impact Number: 11818 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: 10CFR73.55(g)

Summary:

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.

This (revised) Potential Impact supersedes Potential Impacts 11819, 11820, and 11821.

Potential Impact Number: 11822 **SRP Section Number:** 13.6

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Potential Impacts

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: 10CFR73.55(h)(1)

Summary:

Requires a safeguards contingency plan in accordance with 10CFR73, Appendix C.

Potential Impact Number: 11823 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: 10CFR73.55(h)

Summary:

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.

This revised potential impact supersedes Potential Impacts 11824, 11825, 11827 and 11829.

Potential Impact Number: 11837 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: Appendix B

Summary:

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.

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This revised potential impact supersedes Potential Impacts 11832, 11849, 11852, 11854, 11856, 11858, 11861, 11863, 11867, 11871, 11873, 11875, 11881, 11884, 11888, 11889, 11896, 11898, 11904, 11905, 11911, 11914, 11916, 11919, 11924, 11929, 11934, 11937, 11941, 11942, 11952.

Potential Impact Number: 11868 **SRP Section Number:** 8.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.63

Summary:

The station blackout rule for light water reactors. Each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout as defined in Section 50.2. The specified station blackout duration shall be based on the following factors: (i) The redundancy of the onsite emergency ac power sources; (ii) The reliability of the onsite emergency ac power sources; (iii) The expected frequency of loss of offsite power; and (iv) The probable time needed to restore offsite power.
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.

Potential Impact Number: 11870 **SRP Section Number:** 3.9.6

Document: NRC BULLETIN 83-03
Title: CHECK VALVE FAILURES IN RAW WATER COOLING SYSTEMS OF DIESEL GENERATORS

Summary:

This bulletin notified licensees and construction permit holders about numerous incidents of failed check valves in systems important to safety and requested added

surveillance and testing of check valves in raw water cooling systems for diesel generators. A response to this bulletin was required. The bulletin was directed primarily at the failure mode of disassembly or partial disassembly of check valve internals (e.g. the check valve disc separated from the hinge). NRC analysis of operating experience with check valves had shown that disassembly and partial disassembly of check valve internals was not effectively found by Section XI testing as it was implemented at this time. Tests performed for Section XI or Technical Specifications usually required only forward flow through check valves. These tests would not detect internal check valve failures unless the disassembled parts moved to block flow during the test.

Licensees were requested to review the plant Pump and Valve In-Service Test (IST) program required by Section XI of the ASME Boiler and Pressure Vessel Code and modify it if necessary to include check valves in the flow path of cooling water (boundaries were identified) for the diesel generators from the intake to the discharge. For such valves, licensees were requested to examine the IST program and modify it if necessary to include verification procedures that confirm the integrity of the valve internals. This could be accomplished by using both a forward flow and a back flow test or by valve disassembly and inspection. Other equally effective means of assuring integrity of the valves could be used. A reasonable schedule for the test of these valves was to be included in the IST program. Licensees were requested to perform initial valve integrity verification procedures for such valves using the methods described above.

Potential Impact Number: 11874 **SRP Section Number:** 3.9.6

Document: NRC BULLETIN 85-03
Title: MOTOR-OPERATED VALVE COMMON MODE FAILURES DURING PLANT TRANSIENTS DUE TO IMPROPER SWITCH SETTINGS

Summary:

The purpose of this bulletin was to request licensees and permit holders to develop and implement a program to ensure that switch settings on certain safety-related motor-operated valves are selected, set and maintained correctly to accommodate the maximum differential pressures expected on these valves during both normal and abnormal events within the design basis. A response to the bulletin was required.

For motor-operated valves in the high pressure coolant injection/core spray and emergency feedwater systems (RCIC for BWRs) required to be tested for operational

readiness in accordance with 10 CFR 50.55a(g), addressees were requested to develop and implement a program to ensure that valve operator switches are selected, set and maintained properly. This included a program (including revised settings and test procedures) to determine and document valve operation design bases, review and revise, as necessary, methods for selecting and setting all switches (i.e., torque, torque bypass, position limit, overload) for each valve operation (opening and closing), implementation of revised settings, and testing of MOVs at maximum differential pressure with the exception that testing under conditions simulating a break in the line containing the valve was not required. Justification (including the alternative to maximum differential pressure testing used to verify the correct settings) was to be provided for any cases where testing with the maximum differential pressure could not practicably be performed. If the licensee determined that a valve was inoperable, justification for continued operation in accordance with the applicable technical specification was required.

Potential Impact Number: 11877 **SRP Section Number:** 3.9.6

Document: NRC BULLETIN 85-03 Sup 1
Title: MOTOR-OPERATED VALVE COMMON MODE FAILURES DURING PLANT TRANSIENTS DUE TO IMPROPER SWITCH SETTINGS

Summary:

The purpose of this supplement to NRC Bulletin 85-03 was to clarify (1) which valves are to be included and (2) the meaning of the phrase ". . . inadvertent equipment operations (such as inadvertent valve closures or openings) . . ." as used in the bulletin. A response to the bulletin was required.

Actions requested of BWR licensees and permit holders: For safety-related motor-operated valves in the high pressure coolant injection/core spray and reactor core isolation cooling systems not included in the actions planned or completed in response to the original bulletin, develop and implement a program to ensure that valve operator switches are selected, set, and maintained properly. This should include:

a. Review and document the design basis for the operation of each valve. This documentation should include the maximum differential pressure expected during both opening and closing of the valve for both normal and abnormal events to the extent that the events are included in the existing, approved design basis (i.e., the design basis documented in pertinent licensee submittals such as FSAR analyses and fully approved

operating and emergency procedures, etc.). In addition, when determining the maximum differential pressure for valves that can be inadvertently mispositioned, the fact that the valve must be able to recover from such mispositioning should be included. Any motor-operated valve that is not blocked from inadvertent operation from either the control room, the motor control center, or the valve itself should be considered capable of being mispositioned.

b. Perform action item b of the original bulletin for any additional valves identified above. The intent is to provide assurance that a program exists for selecting and setting valve operator switches to ensure a high reliability of safety system valves. If changing the switch settings is not sufficient to ensure the capability for repositioning a particular mispositioned valve, a justification for continued operation should be provided in the bulletin response if the licensee does not elect to implement additional actions, such as administrative or procedural controls or equipment modifications, to minimize the likelihood of valve malfunction.

c. Perform action item c of the original bulletin for any additional valves identified above.

d. Perform action item d of the original bulletin for any additional valves identified above.

Potential Impact Number: 11917 **SRP Section Number:** 3.9.6

Document: NRC GENERIC LETTER 89-10
Title: SAFETY-RELATED/1/ MOTOR-OPERATED VALVE TESTING

Summary:

It is now recognized that the ASME Code Section XI testing alone is not sufficient to provide assurance of MOV operability under design-basis conditions. Assurance of design basis operability is necessary in order to meet the requirements in General Design Criteria 1, 4, 18, and 21 of Appendix A to 10 CFR Part 50 and Criterion XI of Appendix B to 10 CFR Part 50. This letter is part of the resolution of Generic Issue II.E.6.1, "In Situ Testing of Valves."

By this letter NRC extends the scope of the program outlined in Bulletin 85-03 and Supplement 1 of Bulletin 85-03 to include all safety-related MOVs as well as all position-changeable MOVs. The licensee's program should provide for the testing, inspection, and maintenance of MOVs so as to provide the necessary 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.

Any MOV in a safety-related system that is not blocked from inadvertent operation from either the control room, the motor control center, or the valve itself should be considered capable of being mispositioned (referred to as position-changeable MOVs) and should be included in the program. When determining the maximum differential pressure or flow for position-changeable MOVs, the fact that the MOV must be able to recover from mispositioning should be considered.

The program to respond to this letter should address recommended actions a. through h. of the letter. Items a., b., and c. and the first paragraph of d. are repeated, with limited changes, from Bulletin 85-03 or from Supplement 1 of that bulletin. The second paragraph of item d. and items e., f., g., and h. provide additional clarification and guidance.

Potential Impact Number: 11920 **SRP Section Number:** 3.9.6

Document: NRC GENERIC LETTER 89-10 Sup 3
Title: 'CONSIDERATION OF THE RESULTS OF NRC-SPONSORED TESTS OF MOTOR-OPERATED VALVES'

Summary:

The staff performed tests of MOVs as part of an ongoing research effort. The tests were conducted on 6-inch and 10-inch gate valves typically used to provide containment isolation in the steam supply lines of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems, and in the supply line for the Reactor Water Cleanup (RWCU) system at boiling water reactor (BWR) nuclear power plants. As discussed in Information Notice 90-40, "Results of NRC-Sponsored Testing of Motor-Operated Valves," the NRC-sponsored tests revealed that, regardless of fluid conditions, the tested valves required more thrust for opening and closing under various differential pressure and flow conditions than would have been predicted from standard industry calculations using typical friction factors. Thus, although the NRC-sponsored tests focused on the HPCI, RCIC and RWCU containment isolation valves at BWR plants, the information obtained from those tests may be applicable to valves used in other systems at BWR and pressurized water reactor (PWR) plants. For

example, calculations using low valve friction factors may underestimate thrust requirements for opening and closing valves.

BWR licensees are requested to assess (and address with other requested actions detailed in the letter supplement) the applicability of the data from the NRC-sponsored MOV tests, to determine the "as-is" capability of the HPCI, RCIC, and RWCU MOVs and to identify any deficiencies in those MOVs. Where applicable, BWR licensees should also evaluate the MOVs used for containment isolation in lines to the isolation condensers. All licensees and construction permit holders are requested to consider the applicability of the information obtained from the MOV tests and the staff evaluation of the test results to other MOVs within the scope of Generic Letter 89-10. In addition, all licensees and permit holders should consider this information in the development of priorities for implementing the generic letter program.

Potential Impact Number: 11922 **SRP Section Number:** 3.9.6

Document: NRC GENERIC LETTER 89-10 Sup 1
Title: RESULTS OF THE PUBLIC WORKSHOPS

Summary:

This letter supplement provides further clarification on testing requirements of MOV's discussed in Generic Letter 89-10, Safety-Related Motor-Operated Valve Testing and Surveillance. Generic Letter 89-10, issued on June 28, 1989, requests holders of nuclear power plant operating licenses and construction permits to establish a program to provide for the testing, inspection, and maintenance of safety-related motor-operated valves (MOVs) and certain other MOVs in safety-related systems. This supplement to Generic Letter 89-10 includes the opening remarks by NRC representatives and the responses provided by the NRC staff to all significant questions. These responses reflect the best information currently available to the staff from research and operating experience regarding the operability of MOVs under design-basis conditions. For example, the staff believes that, at the present time, the most reliable means of demonstrating operability of an MOV under design-basis conditions is to perform a test of the MOV in situ under those conditions. Such testing in situ, however, will not be practicable for some MOVs within the scope of Generic Letter 89-10.

Potential Impact Number: 11926 **SRP Section Number:** 3.9.6

APPENDIX II
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Document: NRC GENERIC LETTER 90-06
Title: 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,'

Summary:

Based on the analysis and findings for GI-70, the staff concludes that the following actions should be taken to improve the reliability of PORVs and block valves:....

Include PORVs, valves in PORV control air systems, and block valves within the scope of a program covered by Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants," of Section XI of the ASME Boiler and Pressure Vessel Code. Stroke testing of PORVs should only be performed during Mode 3 (HOT STANDBY) or Mode 4 (HOT SHUTDOWN) and in all cases prior to establishing conditions where the PORVs are used for low-temperature overpressure protection. Stroke testing of the PORVs should not be performed during power operation. Additionally, the PORV block valves should be included in the licensees' expanded MOV test program discussed in NRC Generic Letter 89-10, "Safety-Related Motor Operated Valve Testing and Surveillance," dated June 28, 1989.

Potential Impact Number: 11936 **SRP Section Number:** 4.5.1

Document: NRC BULLETIN 82-02
Title: DEGRADATION OF THREADED FASTENERS IN THE REACTOR COOLANT PRESSURE BOUNDARY OF PWR PLANTS

Summary:

The purpose of this bulletin is to: (1) notify licensees and construction permit holders about incidents of severe degradation of threaded fasteners (bolts and studs) in closures in the reactor coolant pressure boundary (RCPB), and (2) to require appropriate actions. Many conditions at operating facilities are described involving stress corrosion cracking degradation/failure of RCPB bolts and studs due to boric acid attack, use of lubricants containing molybdenum disulfide, and/or use of sealing compounds containing halides or sulfides. RCPB materials most vulnerable to such degradation/failure are also identified in this bulletin. A response to this bulletin was required from pressurized water reactors (PWRs) holding an operating license. Actions

required to be taken by PWR licensees included NDE examination of control rod drive flange and pressurizer heater connections that do not have seal welds.

Potential Impact Number: 11957 **SRP Section Number:** 8.3.2

Document: RG 1.47 **Revision:** 0
Title: BYPASSED AND INOPERABLE STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS

Summary:

Provides design guidance for the design of inoperable/bypass status indication for nuclear power plant safety systems. The RG is cited as guidance for SRP Section 8.3.2 in Sections 8.1 and 8.3.2.

Potential Impact Number: 11992 **SRP Section Number:** 3.9.6

Document: NRC GENERIC LETTER 89-13
Title: SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

Summary:

Actions listed to ensure service water systems comply with 10CFR Part 50 and ASME Section XI.

Potential Impact Number: 12101 **SRP Section Number:** 8.3.2

Document: RG 1.153 **Revision:** 0
Title: CRITERIA FOR POWER, INSTRUMENTATION, AND CONTROL

Summary:

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Provides guidance for the design, reliability, qualification, and testability of power, instrumentation, and control portions of safety systems for nuclear power plants. The RG is not cited as guidance for SRP Section 8.3.2 in Section 8.1 or 8.3.2.

Potential Impact Number: 12103 **SRP Section Number:** 8.3.2

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

Provides guidance for compliance with the station blackout rule for light water reactors (10CFR50.63). Covers such topics as standby emergency power source target reliabilities and analyses to determine an appropriate specified duration of blackout for which withstand capability must be demonstrated. ... Consideration should be given to timely operator actions inside or outside the control room that would increase the length of time that the plant can cope with a station blackout provided it can be demonstrated that these actions can be carried out in a timely fashion. For example, if station battery capacity is a limiting factor in coping with a station blackout, shedding nonessential loads on the batteries could extend the time until the battery is depleted. If load shedding or other operator actions are considered, corresponding procedures should be incorporated into the plant or specific technical guidelines and emergency operating procedures.The RG is not cited as guidance for SRP Section 8.3.2 in Section 8.1 or 8.3.2.

Potential Impact Number: 12134 **SRP Section Number:** 8.3.2

Document: RG 1.128 **Revision:** 1
Title: INSTALLATION DESIGN AND INSTALLATION OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS

Summary:

Provides guidance for performing the installation design and installation of large lead storage batteries for all types of nuclear power plants. The RG is cited as guidance for SRP Section 8.3.2 in Sections 8.1 and 8.3.2.

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Potential Impact Number: 12198 **SRP Section Number:** 13.6

Document: RG 5.12 **Revision:** 0
Title: GENERAL USE OF LOCKS IN THE PROTECTION AND CONTROL OF FACILITIES AND SPECIAL NUCLEAR MATERIALS

Summary:

Locks are acceptable devices to be used in adhering to the physical protection requirements to assist in controlling access to areas, facilities, and materials through doors, gates, container lids, and similar material or personnel access points, and are considered essential components of a physical barrier. This guide provides criteria acceptable to the Regulatory Staff for the selection and use of commercially available locks in the protection of facilities and SNM, including electric lock system connections with the alarm system.

This revised potential impact supersedes Potential Impacts 12202, 12203, 12204, 12205 and 12206.

Potential Impact Number: 12208 **SRP Section Number:** 13.6

Document: RG 5.20 **Revision:** 0
Title: TRAINING, EQUIPPING, AND QUALIFYING OF GUARDS AND WATCHMEN

Summary:

Reg. Guide 5.20 provides acceptable criteria for a program for training, equipping and qualifying guards and watchmen.

This (revised) potential impact supersedes Potential Impacts 12209, 12213, 12215, 12218, 12220, 12225, and 12226.

Potential Impact Number: 12244 **SRP Section Number:** 13.6

Document: RG 5.44 **Revision:** 2

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Title: PERIMETER INTRUSION ALARM SYSTEMS

Summary:

This guide 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.

This revised potential impact supersedes Potential Impacts 12232, 12233, 12234, 12235, 12236, 12238, 12239, 12240, 12245, 12246, 12247, 12248, 12249 and 12251.

Potential Impact Number: 12253 **SRP Section Number:** 13.6

Document: RG 5.44 **Revision:** 2
Title: PERIMETER INTRUSION ALARM SYSTEMS

Summary:

Provides perimeter intrusion detection alarm systems should be operability tested at least once every 7 days, as well as performance tested quarterly, after each inoperable period, and after any repairs. Examples of testing methods are provided. Instruction for calculating the confidence limit on detection probability are also provided.

This potential impact supersedes Potential Impacts 12328, 12329 and 12330.

Potential Impact Number: 12254 **SRP Section Number:** 6.3

Document: NRC BULLETIN 79-24
Title: FROZEN LINES

Summary:

This Bulletin describes a problem discovered during a monthly surveillance test of the HPCI pumps, there was no flow through the recirculation line from the pump discharge to the Borated Water Storage Tank (BWST). Investigation revealed that water in a portion of the line exposed to the outside weather was frozen. This portion of the line to the BWST is common to both HPCI pumps.

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Potential Impact Number: 12258 **SRP Section Number:** 8.3.2

Document: NRC BULLETIN 79-27
Title: LOSS OF NON CLASS 1-E INSTRUMENTATION AND CONTROL POWER SYSTEM BUS DURING OPERATION

Summary:

The bulletin describes an event at Oconee Unit 3 involving a unit trip and loss of non-class 1E inverter/ups power to the integrated control system resulting in a plant transient and apparent difficulties achieving cold shutdown. Several licensee actions are directed to provide assurance that cold shutdown may be achieved in the event of loss of class 1E or non-class 1E instrumentation and control power.

Potential Impact Number: 12269 **SRP Section Number:** 6.3

Document: NRC GENERIC LETTER 83-02
Title: NUREG-0737 Technical Specifications

Summary:

The pipe break-detection circuitry should be modified so that pressure spikes resulting from HPCI system initiation will not cause inadvertent system isolation. (II.K.3.15)

Potential Impact Number: 12293 **SRP Section Number:** 8.3.2

Document: NRC GENERIC LETTER 91-06
Title: RESOLUTION OF GENERIC ISSUE A-30, 'ADEQUACY OF SAFETY-RELATED DC POWER SUPPLIES,' PURSUANT TO 10 CFR 50.54(f)

Summary:

Provides staff positions on the resolution of generic issue A-30, concerning the adequacy of safety related DC power supplies. Basic position is that through adequate

maintenance, monitoring, and surveillance, adequate reliability will be assured. An extensive questionnaire is provided to request information regarding the need for further staff action at specific plants.

Potential Impact Number: 12349 **SRP Section Number:** 13.6

Document: RG 5.65 **Revision:** 0
Title: VITAL AREA ACCESS CONTROLS, PROTECTION OF PHYSICAL SECURITY EQUIPMENT, AND KEY AND LOCK CONTROLS

Summary:

This RG provides guidance for openings in vital area barriers, separation of protected area and vital area barriers at water sources, and time-dependent vital areas (spent fuel pools).

This (revised) Potential Impact supersedes Potential Impacts 12348, 12350, 12351, 12370, 12371 and 12372.

Potential Impact Number: 12352 **SRP Section Number:** 13.6

Document: RG 5.65 **Revision:** 0
Title: VITAL AREA ACCESS CONTROLS, PROTECTION OF PHYSICAL SECURITY EQUIPMENT, AND KEY AND LOCK CONTROLS

Summary:

This RG provides guidance for control of access to vital areas under routine and emergency conditions and for suspending security measures.

This revised potential impact supersedes Potential Impacts 12354, 12357, 12353, 12358, 12356, 12359, 12361, 12362, 12363, and 12364.

Potential Impact Number: 12360 **SRP Section Number:** 13.6

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Document: RG 5.65 **Revision:** 0
Title: VITAL AREA ACCESS CONTROLS, PROTECTION OF PHYSICAL SECURITY EQUIPMENT, AND KEY AND LOCK CONTROLS

Summary:

This RG provides guidance for physical protection plan and contingency plan interfaces.

This revised potential impact supersedes Potential Impacts 12361, 12362, 12363, 12364, and 12365.

Potential Impact Number: 12390 **SRP Section Number:** 13.6

Document: NRC GENERIC LETTER 87-08
Title: IMPLEMENTATION OF 10 CFR 73.55 MISCELLANEOUS AMENDMENTS AND SEARCH REQUIREMENTS

Summary:

The letter answers generic questions (in a question and answer format) posed by power reactor licensees concerning implementation of 10CFR73.55.

Potential Impact Number: 12432 **SRP Section Number:** 13.6

Document: NRC GENERIC LETTER 89-07
Title: POWER REACTOR SAFEGUARDS CONTINGENCY PLANNING FOR SURFACE VEHICLE BOMBS

Summary:

The letter requires licensees to modify their safeguards contingency procedures to address the possibility of a land vehicle bomb, and describes the acceptable approach. The letter references 10CFR50.34(d); 10CFR73, Appendix C; and 10CFR50.54 (P)(1).

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Potential Impact Number: 12433 **SRP Section Number:** 13.6

Document: NRC GENERIC LETTER 89-07 Sup 1
Title: 'POWER REACTOR SAFEGUARDS CONTINGENCY PLANNING FOR SURFACE VEHICLE BOMBS'

Summary:

This supplement provides an enclosure in question and answer format to clarify Generic Letter 89-07 and provide additional information.

Potential Impact Number: 12449 **SRP Section Number:** 13.6

Document: NRC GENERIC LETTER 83-21
Title: Clarification of Access Control Procedures for Law Enforcement Visits

Summary:

The intent of 10CFR73.55 (d)(1) is to detect unauthorized materials. Licensees need not require on-duty U.S., state, and local law enforcement officers to leave their firearms and ammunition outside protected areas, provided that they have positive identification of their status.

Potential Impact Number: 12546 **SRP Section Number:** 3.11

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.49

Summary:

Environmental qualification of electrical equipment important to safety of nuclear power plants.

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Potential Impact Number: 12767 **SRP Section Number:** 6.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.34 (f)(1)(vii)

Summary:

Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWR's only). (II.K.3.18)

Potential Impact Number: 12768 **SRP Section Number:** 6.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.34 (f)(1)(x)

Summary:

Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWR's only). (II.K.3.28).

Potential Impact Number: 12859 **SRP Section Number:** 3.2.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR50.55a (c)

Summary:

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This portion of 10 CFR 55a states that components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III of the ASME BPVC with certain exceptions. The acceptable exceptions to this requirement are also described.

Potential Impact Number: 12867 **SRP Section Number:** 9.3.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: GDC 5

Summary:

Structures, systems and components important to safety shall not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety function.

Potential Impact Number: 12868 **SRP Section Number:** 9.3.1

Document: NRC BULLETIN 80-01
Title: OPERABILITY OF ADS VALVE PNEUMATIC SUPPLY

Summary:

The ADS pneumatic supply (either nitrogen or air) may not be operable for all possible events because of a combination of misapplication of check valve, a lack of leak testing of backup accumulator and seismic reliability.

Actions to be taken by Licensees of GE BWR facilities with an operating license which use a pneumatic operator for ADS function: 1. Determine if your facility has installed hard-seat check valves to isolate the ADS accumulator system from the pneumatic supply system. 2. Determine if periodic leak tests have been performed on your ADS accumulator systems to assure emergency pneumatic supply for the FSAR-required number and duration of valve operations. 3. Review seismic qualifications of the ADS pneumatic supply system: (a) from accumulator system isolation check valve to ADS valve operator, (b) from isolation valve outside containment up to ADS accumulator check valve.

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Potential Impact Number: 12869 **SRP Section Number:** 9.3.1

Document: NRC BULLETIN 80-25
Title: OPERATING PROBLEMS WITH TARGET ROCK SAFETY-RELIEF VALVES AT BWRs

Summary:

A review of your S/R valve pneumatic supply systems shall be performed to determine the potential for and magnitude of an overpressure condition. The determined overpressure potential of the pneumatic supply shall be compared with the maximum operating pressure capabilities of the solenoid actuator valves serving the S/R valves, so as to determine whether supply pressure could result in valve malfunction. Protective devices (such as relief valves) shall be installed in the proximity of the S/R valves and set to protect against supply pressure in excess of the operating pressure capabilities-of the solenoid actuator device. In addition, consideration should be given to modification or replacement to reduce the sensitivity of the solenoid actuator to pneumatic supply overpressure.

Potential Impact Number: 12870 **SRP Section Number:** 9.3.1

Document: NRC CIRCULAR 81-14
Title: MAIN STEAM ISOLATION VALVE FAILURES TO CLOSE

Summary:

Based on data available in Licensee Event Report (LER) files, the failure rate of main steam isolation valves (MSIVs) closing has ranged from a ten-year low of three to fourteen in 1980. Recent failures, similar to those occurring over the last ten years, are primarily related to the following two causes: (1) poor quality control air to the pilot valves, and (2) binding of the MSIV valve stems with the valve stem packing.

Recommended Actions for Holders of Operating Licenses: 1. Review MSIV operating experience for problems that are causing failure to close events or are causing equipment degradation that requires other than routine maintenance to prevent a failure to close. 2. Evaluate corrective action identified in maintenance records, LERs, etc , for adequacy in addressing the root cause of problems and develop plans for

additional corrective action as necessary. 3. Where control air quality is suspected of contributing to problems: a. Review the air system(s) to ensure that measures have or will be taken to prevent air quality degradation in the future.

Potential Impact Number: 12873 **SRP Section Number:** 9.3.1

Document: NRC GENERIC LETTER 88-14
Title: INSTRUMENT AIR SUPPLY SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

Summary:

The purpose of this generic letter is to request that each licensee/applicant review NUREG-1275, Volume 2, an, perform a design and operations verification of the instrument air system. This verification should include:

1. Verification by test that actual instrument air quality is consistent with the manufacturer's recommendations for individual components served.

2. Verification that maintenance practices, emergency procedures, and training are adequate to ensure that safety-related equipment will function as intended on loss of instrument air.

3. Verification that the design of the entire instrument air system including air or other pneumatic accumulators is in accordance with its intended function, including verification by test that air-operated safety-related components will perform as expected in accordance with all design-basis events, including a loss of the normal instrument air system. This design verification should include an analysis of current air operated component failure positions to verify that they are correct for assuring required safety functions.

In addition to the above, each licensee/applicant should provide a discussion of their program for maintaining proper instrument air quality.

Potential Impact Number: 12915 **SRP Section Number:** 3.2.2

Document: CODE OF FED. REGS 10CFR50

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Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.2

Summary:

'Reactor coolant pressure boundary' means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are: (1) Part of the reactor coolant system, or (2) Connected to the reactor coolant system.

Potential Impact Number: 12917 **SRP Section Number:** 3.2.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50, Appendix A, Criterion 30

Summary:

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical.

Potential Impact Number: 12918 **SRP Section Number:** 9.3.1

Document: NRC GENERIC LETTER 89-18
Title: RESOLUTION OF UNRESOLVED SAFETY ISSUE A-17, 'SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS'

Summary:

The safety significance of event initiation (of support systems such as compressed air) coupled with limited capabilities for mitigation may not have been recognized. In the case of the air system, the system was assumed to fail safe, i.e. bleed off, and as a result, a partial failure was unanalyzed.

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Potential Impact Number: 12921 **SRP Section Number:** 3.2.2

Document: RG 1.7
Title: CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT

Summary:

The combustible gas control system should -be designated Seismic Category I (see Regulatory Guide 1.29, 'Seismic Design Classification'), and the Group B quality standards of Regulatory Guide 1. 26, 'Quality Group Classifications and Standards for Water-Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants, ' should be applied.

Potential Impact Number: 12927 **SRP Section Number:** 9.3.1

Document: NRC NOTICE 80-30
Title: POTENTIAL FOR UNACCEPTABLE INTERACTION BETWEEN THE CONTROL ROD DRIVE SCRAM FUNCTION AND NON-ESSENTIAL CONTROL AIR AT CERTAIN GE BWR FACILITIES

Summary:

Continued NRC evaluation of possible failure modes of the control rod drive system related to the Browns Ferry event (IE Bulletin 80-17), has identified a potentially significant single failure mechanism which could degrade the scram function. The slow loss of non-essential control air pressure could degrade the scram function at certain facilities which have relatively poor communication between the scram discharge volume (SDV) and the scram discharge instrumented volume (SDIV) such that the SDIV drains more rapidly than the SDV drain rate into the SDIV.

Potential Impact Number: 12929 **SRP Section Number:** 9.3.1

Document: NRC NOTICE 80-40
Title: EXCESSIVE NITROGEN SUPPLY PRESSURE ACTUATES SAFETY-RELIEF VALVE OPERATION TO CAUSE REACTOR DEPRESSURIZATION

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Summary:

This notice describes problems with BWR Target Rock safety-relief valves attributed to actuating air overpressure. The following are excerpts from the notice.

The design normal operating pressure of the compressed air or nitrogen systems supplying the safety-relief valves is 90-110 psi. At a pressure of 145 psi the solenoid valve may begin to leak since excessive pressure acts to unseat the disk.

The particular solenoid actuator valves used with these two-stage safety- relief valves and manufactured by TRC. Their design is such that excessive control pressure tends to unseat the solenoid valve disk. The three-stage TR safety-relief valves use either AVCO (Automatic Valve Company) or ASCO (Automatic Switch Company) solenoid valves according to information from GE. The AVCo solenoid valve has been tested to 300 psi and found not to leak and the ASCO solenoid valve tends to seat with increasing supply pressure.

Potential Impact Number: 12933 **SRP Section Number:** 9.3.1

Document: NRC NOTICE 81-38
Title: POTENTIALLY SIGNIFICANT EQUIPMENT FAILURES RESULTING FROM CONTAMINATION OF AIR-OPERATED SYSTEMS

Summary:

Several failures of the compressed air system have been identified that have the potential for common mode failures. These include oil contamination, water accumulation and the presence of desiccant in the air system.

Potential Impact Number: 12939 **SRP Section Number:** 9.3.1

Document: NRC NOTICE 85-35
Title: FAILURE OF AIR CHECK VALVES TO SEAT

Summary:

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The MSIVs failed to close on a main steam isolation signal as required. The cause was determined to be failure of instrument air check valves to seat in response to gradually decreasing air pressure.

Potential Impact Number: 12950 **SRP Section Number:** 3.2.2

Document: RG 1.70 **Revision:** 3
Title: STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS

Summary:

This section should identify those fluid systems or portions of fluid systems important to safety and the industry codes and standards applicable to each pressure-retaining component in the systems.

Potential Impact Number: 12963 **SRP Section Number:** 3.2.2

Document: RG 1.72 **Revision:** 2
Title: SPRAY POND PIPING MADE FROM FIBERGLASS-REINFORCED THERMOSETTING RESIN

Summary:

Safety-related spray pond piping components made from fiberglass-reinforced thermosetting resin should comply with ASME Code Case N-155-1 (1792-1) supplemented by the positions specified in this regulatory guide which by virtue of supplementing an ASME Code Case also supplement the ASME Code.

A footnote is also provided which indicates that Safety-related spray pond piping components made from fiberglass-reinforced thermosetting resin should comply with Quality Group A, B, and C requirements of the ASME Boiler and Pressure Vessel Code for Class 3 components (NRC Regulatory Guide 1.26, 'Quality Group Classification and Standards for Water-, Steam-, and Radioactive Waste Containing Components at Nuclear Power Plants').

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Potential Impact Number: 12971 **SRP Section Number:** 9.3.1

Document: NRC NOTICE 86-57
Title: OPERATING PROBLEMS VALVES AT NUCLEAR POWER PLANTS

Summary:

The failure of the SOVs can be traced to the following different causes: ... 2) hydrocarbon contaminants, probably because backup air systems are being used periodically that are not designed to 'oil free' specifications as required by Class IE service. The findings indicated that the SOV failures could have been caused by a combination of hydrocarbon contamination of the air system and ... (certain valves and components require an oil-free system).

Potential Impact Number: 12978 **SRP Section Number:** 3.9.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR50.55a(d)

Summary:

For nuclear power plant whose application for a construction permit is docketed after May 14, 1984 components classified Quality Group B must meet the requirements for Class 2 Components of ASME B&PV Code, Section III.

Potential Impact Number: 12981 **SRP Section Number:** 3.9.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.55a(e)

Summary:

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For a nuclear power plant whose application for a construction permit is docketed after May 14, 1984, components classified Quality Group C must meet the requirements for Class 3 components in Section III of ASME B&PV Code.

Potential Impact Number: 12986 **SRP Section Number:** 9.3.1

Document: NRC NOTICE 88-24
Title: FAILURES OF AIR-OPERATED VALVES AFFECTING SAFETY-RELATED SYSTEMS

Summary:

In many plants, the air supply systems for safety-related components are not designed as safety-related systems. Hence, 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.

Potential Impact Number: 12989 **SRP Section Number:** 3.2.2

Document: RG 1.96 **Revision:** 1
Title: DESIGN OF MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEMS FOR BOILING WATER REACTOR NUCLEAR POWER PLANTS

Summary:

The leakage control system and any necessary subsystems, including the source of any sealing fluid if a fluid seal type of system is used, should be designed in accordance with Quality Group B requirements, with the exception of any portion of LCS piping that connects to main steam system piping between inner and outer containment isolation valves of the main steam system for either single- or dual-barrier containment structures. Such piping, up to and including the first isolation valve in the LCS piping, should be designed in accordance with Quality Group A requirements supplemented by Appendix A of this guide.

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Potential Impact Number: 13002 **SRP Section Number:** 9.3.1

Document: RG 1.68.3 **Revision:** 0
Title: PREOPERATIONAL TESTING OF INSTRUMENT AND CONTROL AIR SYSTEMS

Summary:

This guide (which replaces Reg. Guide 1.80) describes an acceptable method to the NRC staff for complying with the Commission's regulations with respect to preoperational testing to verify that instrument and control air systems and the loads they supply will operate properly. The recommendations of this guide should also be applied to: 1) compressed gas systems not designated as instrument or control air systems ... 2) compressed air systems that supply specific loads, 3) systems important to safety that use compressed gas other than air.

Reg. Guide 1.68.3 is an acceptance criteria, however, there are no review procedures in the SRP.

Potential Impact Number: 13006 **SRP Section Number:** 3.2.2

Document: RG 1.143 **Revision:** 1
Title: DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

Summary:

This Reg. Guide establishes design and quality assurance requirements for radioactive waste management systems.

Potential Impact Number: 13009 **SRP Section Number:** 9.3.1

Document: RG 1.68 **Revision:** 2
Title: INITIAL TEST PROGRAMS FOR WATER-COOLED NUCLEAR POWER PLANTS

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Summary:

Testing should be performed for compressed gas systems supplying pneumatic equipment, components or instrumentation that are required ... * Note, Reg. Guide 1.80 is replaced by 1.68.3.

Potential Impact Number: 13012 **SRP Section Number:** 3.2.2

Document: RG 1.151 **Revision:** 0 , 7/1/83
Title: INSTRUMENT SENSING LINES

Summary:

The requirements of ISA-S67.02, 'Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants,' 1980, provide a basis acceptable to the NRC staff for the design and installation of safety-related instrument sensing lines in nuclear power plants. Instrument sensing lines that are connected to ASME Class 1 or 2 process piping or vessels should not be less than ASME Class 2 from their connections to the process piping or vessel to and including the accessible isolation valve. Instrument sensing lines that are connected to ASME Class 1 or 2 process piping or vessels and that are used to actuate or monitor safety related systems should not be less than ASME Class 2 Seismic Category I from their connections to the process piping or vessel to the sensing instrumentation. Instrument sensing lines that are connected to ASME Class 3 process piping and vessels and that are used to actuate or monitor safety-related systems should not be less than ASME Class 3 from their connection to the process piping or vessel to the sensing instrumentation.

Potential Impact Number: 13015 **SRP Section Number:** 9.3.1

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

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The capability of all systems and components necessary to provide core cooling and decay heat removal following a station blackout should be determined, including ... compressed air capacity.

Potential Impact Number: 13016 **SRP Section Number:** 9.3.1

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

If the compressed air capacity is not sufficient to remove decay heat and to maintain appropriate containment integrity for the station blackout duration recommended in Regulatory Position 3.1, a system to provide sufficient capacity from an alternative source that meets Regulatory Position 3.5 is an acceptable means to increase the air capacity provided any power source necessary to provide additional air is independent of both the offsite and the blacked-out unit's onsite emergency ac power systems.

Potential Impact Number: 13025 **SRP Section Number:** 3.2.2

Document: NRC CIRCULAR 80-18
Title: 10 CFR 50.59 SAFETY EVALUATIONS FOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

Summary:

Radioactive waste management system modifications should be evaluated against the seismic, quality group and quality assurance criteria in Regulatory Guide 1.143.

Potential Impact Number: 13085 **SRP Section Number:** 3.2.2

Document: NRC GENERIC LETTER 90-06
Title: 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,'

Summary:

Where PORVs are used or could be used to perform one, or more safety- related functions or to perform any other safety-related function that may be identified in the future, it is appropriate to reconsider the safety classification of PORVs and the associated block valves. For certain PWR plants receiving an operating license in recent years, the staff has required these valves to be classified as safety-related components if they perform one, or more, safety-related functions.

Potential Impact Number: 13140 **SRP Section Number:** 3.2.2

Document: NRC GENERIC LETTER 90-06
Title: 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,'

Summary:

As part of the NRC staff acceptance of LTOP protection system designs for the implementation of the resolution of USI A-26, it was concluded that the costs associated with upgrading existing system to meet the guidance of Regulatory Guide 1.26 were not justifiable. Further evaluations performed for GI-94 have also concluded that it is not cost beneficial to upgrade these systems to fully safety-grade standards.

Potential Impact Number: 13247 **SRP Section Number:** 3.2.2

Document: RG 1.26 **Revision:** 3
Title: QUALITY GROUP CLASSIFICATIONS AND STANDARDS FOR WATER-, STEAM-, AND RADIOACTIVE-WASTE-CONTAINING COMPONENTS OF NUCLEAR POWER PLANTS

Summary:

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The NRC staff has developed a quality classification system for safety-related components containing water, steam, or radioactive material in water-cooled nuclear power plants. The system consists of four quality groups, A through D, methods for assigning components to these quality groups, and the specific quality standards applicable to each quality group. The initial portion of this system is described in Section 50.55a of 10 CFR Part 50, which requires that components of the reactor coolant boundary be designed, fabricated, erected and tested to the highest available national standards; this corresponds to the quality to the quality standard required for quality Group A of the NRC system. This guide describes a method for determining acceptable quality standards for the remaining safety-related components containing radioactive material, water, or steam, i.e., quality Group B, C, and D components. Other systems not covered by this guide, such as instrument and service air, diesel engine and its generators and auxiliary support systems, diesel fuel, emergency and normal ventilation, fuel handling, and radioactive waste management systems should be designed, fabricated, erected, and tested to quality standards commensurate with the safety function to be performed.

Potential Impact Number: 13255 **SRP Section Number:** 3.2.2

Document: RG 1.137 **Revision:** 1
Title: FUEL-OIL SYSTEMS FOR STANDBY DIESEL GENERATORS

Summary:

ANSI N195-1976 states that the standard provides the design requirements for the fuel-oil system for standby diesel generators and that it sets forth other specific design requirements such as safety class, materials, physical arrangement, and applicable codes and regulations.

Potential Impact Number: 13289 **SRP Section Number:** 2.4.11

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.79(b)

Summary:

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An application referencing certified design must describe those portions of the design which are site-specific, such as the service water intake structure and the ultimate heat sink.

Potential Impact Number: 13449 **SRP Section Number:** 5.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Section 50.60, page 1

Summary:

All lightwater reactors must meet fracture toughness requirements as set forth in 10CFR50, Appendices G and H.

Potential Impact Number: 13451 **SRP Section Number:** 5.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Section 50.61

Summary:

Fracture toughness requirements for protection against PTS.

Potential Impact Number: 13476 **SRP Section Number:** 5.3.2

Document: NRC GENERIC LETTER 82-26
Title: NUREG-0744 REV. 1; PRESSURE VESSEL MATERIAL FRACTURE TOUGHNESS

Summary:

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This Generic Letter transmitted NUREG-0744, Revision 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. NUREG-0744 provides a method for evaluating reactor vessel materials when their Charpy upper shelf energy is predicted to fall below 50 ft-lb. Plants will use the prescribed method when analysis of irradiation damage predicts that the Charpy upper shelf energy is below 50 ft-lb.

Potential Impact Number: 13482 **SRP Section Number:** 5.3.2

Document: NRC GENERIC LETTER 88-11
Title: NRC POSITION ON RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS AND ITS IMPACT ON PLANT OPERATIONS

Summary:

Reg. Guide 1.99, Revision 2, becomes effective May 1988. It will be used in reviewing submittals regarding pressure/temperature limits, and for others except PTS, that require an estimate of embrittlement of reactor beltline materials.

Potential Impact Number: 13537 **SRP Section Number:** 5.3.2

Document: RG 1.99 **Revision:** 2
Title: RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS

Summary:

Holders of licenses should use the methods described in this guide to predict the effect of neutron radiation on reactor vessel materials as required by paragraph V.A of Appendix G to 10CFR50.

Potential Impact Number: 13646 **SRP Section Number:** 3.9.3

Document: NRC BULLETIN 88-11

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Title: PRESSURIZER SURGE LINE THERMAL STRATIFICATION

Summary:

This Bulletin was addressed to all holders of operating licenses or construction permits for pressurized water reactors (PWRs). The purpose of this bulletin is to (1) request that addressees establish and implement a program to confirm pressurizer surge line integrity in view of the occurrence of thermal stratification and (2) require addressees to inform the staff of the actions taken to resolve this issue.

Potential Impact Number: 13929 **SRP Section Number:** 4.3

Document: RG 1.77 **Revision:** 0
Title: ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS

Summary:

The ejected rod worth should be calculated based on the maximum worth rod.

The calculation of effective delayed neutron fraction and prompt neutron lifetime should be based on the well-known definitions resulting from perturbation theory, such as described by Henry (Ref. 1), using available experimental delayed neutron data and averaging by the fraction of fission in the various fissionable materials.

The moderator reactivity coefficients due to voids, coolant pressure changes, and coolant temperature changes should be calculated based on the various assumed conditions of the fuel and moderator using standard transport and diffusion theory codes.

The Doppler coefficient should be calculated based on the effective resonance integrals and should include corrections for pin shadowing (Dancoff correction. Calculations of the Doppler coefficient of reactivity should be based on and should compare conservatively with available experimental data such as those of Hellstrand (Ref. 6).

Potential Impact Number: 14106 **SRP Section Number:** 3.2.2

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Document: NUREG 1316
Title: TECHNICAL FINDINGS AND REGULATORY ANALYSIS RELATED TO GENERIC ISSUE 70, 'EVALUATION OF POWER OPERATED RELIEF VALVE AND BLOCK VALVE RELIABILITY IN PWR NUCLEAR POWER PLANTS.'

Summary:

NRC staff positions regarding PWR PORV and PORV block valve safety related functions are stated. Describes NRC staff positions with regard to the applicability of regulation, codes and standards to PWR PORV and PORV block valves. An alternative listed was to require PORVs and block valves to be safety related. This NUREG is referenced in Generic Letter 90-06. Microfiche no. 52611.

Potential Impact Number: 14107 **SRP Section Number:** 3.2.2

Document: NUREG 1326
Title: REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC ISSUE 94, 'ADDITIONAL LOW-TEMPERATURE OVERPRESSURE PROTECTION FOR LIGHT WATER REACTORS.'

Summary:

An alternative would require that the low-temperature overpressure protection system be upgraded to a fully safety-grade system. Referenced in Generic Letter 90-06. Microfiche No. 52231

Potential Impact Number: 14116 **SRP Section Number:** 5.2.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Section 50.55a

Summary:

Each operating license for a boiling or pressurized water-cooled nuclear power facility is subject to the conditions in paragraphs (f), "Inservice testing requirements," and (g),

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"Inservice inspection requirements," of this section and each construction permit for a utilization facility is subject to the following conditions specified in (a) - (h) in addition to those specified in 50.55.

Potential Impact Number: 14125 **SRP Section Number:** 5.2.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 32

Summary:

Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Potential Impact Number: 14134 **SRP Section Number:** 5.2.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix G, V. Inservice Requirements, C(1)

Summary:

A volumetric examination of 100% of the belt line materials that do not satisfy the requirements of Section V.B of Appendix G is made and any flaws characterized according to Section XI and as otherwise specified by the Director, Office of NRR.

Potential Impact Number: 14137 **SRP Section Number:** 5.2.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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Location: Appendix H, II. Surveillance Program Criteria, B.2

Summary:

Inservice inspection of surveillance capsule holder attachments and attachment welds is required if the holders are attached to the reactor vessel or vessel cladding.

Potential Impact Number: 14166 **SRP Section Number:** 9.3.1

Document: NRC NOTICE 85-06
Title: CONTAMINATION OF BREATHING AIR SYSTEMS

Summary:

There is that potential for an unreviewed safety question situation when system operations are changed from what is described in the Safety Analysis Report, such as cross-connecting contaminated and noncontaminated systems.

Potential Impact Number: 14168 **SRP Section Number:** 5.2.4

Document: NRC GENERIC LETTER 81-03
Title: IMPLEMENTATION OF NUREG-0313, REV. 1, 'TECHNICAL REPORT ON MATERIAL SELECTION AND PROCESSING GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING (GENERIC TASK A-42)

Summary:

This Generic Letter forwarded the final NUREG-0313, Rev. 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," dated July 1980. This final report constitutes the staff's resolution of the NRC's Generic Technical Activity A-42, "Pipe Cracks in Boiling Water Reactors.". NUREG-0313, Rev. 1 sets forth the NRC staff's revised acceptable methods to reduce the intergranular stress corrosion cracking susceptibility of BWR ASME Code Class 1, 2, and 3 pressure boundary piping and safe ends. For plants that cannot fully comply with the material selection, testing and processing guidelines of this report, varying degrees of augmented Inservice Inspection and leak detection requirements are presented.

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Potential Impact Number: 14181 **SRP Section Number:** 5.2.4

Document: NRC GENERIC LETTER 88-01
Title: NRC POSITION ON IGSCC IN BWR AUSTENITIC STAINLESS STEEL PIPING

Summary:

Attachment A to this letter contains staff positions, including: 1) inspection methods and personnel: 2) inspection schedules; and, 3) inspection sample expansion. The technical bases for these positions are detailed in NUREG-0313, Rev. 2, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping."

Potential Impact Number: 14184 **SRP Section Number:** 9.3.1

Document: NRC BULLETIN 80-10
Title: CONTAMINATION OF NONRADIOACTIVE SYSTEM AND RESULTING POTENTIAL FOR UNMONITORED,UNCONTROLLED RELEASE OF RADIOACTIVITY TO ENVIRONMENT

Summary:

Review your facility design and operation to identify systems that are considered as nonradioactive (or described as nonradioactive in the FSAR), but could possibly become radioactive through interfaces with radioactive systems, i.e., a nonradioactive system that could become contaminated due to leakage, valving errors or other operating conditions in radioactive systems. In particular, special consideration should be given to the following systems: auxiliary boiler system, demineralized water system, isolation condenser system, PWR secondary water clean-up system, instrument air system, and the sanitary waste system.

Potential Impact Number: 14189 **SRP Section Number:** 9.3.1

Document: NRC NOTICE 79-08

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Title: INTERCONNECTION OF CONTAMINATED SYSTEMS WITH SERVICE AIR SYSTEMS USED AS THE SOURCE OF BREATHING AIR

Summary:

NRC contacts with operating power reactors have indicated that facilities use service air as a source of breathing air. In several cases, the service air system is physically interconnected to other systems which carry radioactive material and relies on valves or other barriers to prevent contamination of the plant source of breathing air.

At those facilities where service air is used as a source of breathing air, certain precautions can be taken to minimize the potential of a problem similar to the one described. By review of the service air system and as-built or design drawings, interconnections between contaminated systems and the source of breathing air can be identified. Where interconnections are identified, the design should assure that adequate separation is provided. In addition, operating procedures should provide for controls to prevent contamination of the breathing air source.

Potential Impact Number: 14204 **SRP Section Number:** 5.2.4

Document: NRC GENERIC LETTER 89-08
Title: EROSION/CORROSION-INDUCED PIPE WALL THINNING

Summary:

The U.S. Nuclear Regulatory Commission (NRC) is requiring information to assess safe operation of reactors when erosion/ corrosion significantly degrades piping and components of high-energy carbon steel piping systems. The principal concern is whether the affected plants continue to meet their licensing basis when erosion/corrosion degrades the pressure boundary to below the applicable code design value.

Licensees were requested to provide responses on whether or not they had implemented or intended to implement a long term erosion/corrosion monitoring program that provides assurances that procedures or administrative controls are in place to assure that the NUMARC program or another equally effective program is implemented and the structural integrity of all high-energy (two phase as well as single phase) carbon steel systems is maintained.

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Potential Impact Number: 14308 **SRP Section Number:** 5.2.4

Document: RG 1.150 **Revision:** 1
Title: ULTRASONIC TESTING OF REACTOR VESSEL WELDS DURING PRESERVICE AND INSERVICE EXAMINATION

Summary:

Reg. Guide describes procedures for conducting preservice and inservice examinations of reactor vessel welds by ultrasonic testing.

Potential Impact Number: 14391 **SRP Section Number:** 5.2.4

Document: NRC BULLETIN 82-02
Title: DEGRADATION OF THREADED FASTENERS IN THE REACTOR COOLANT PRESSURE BOUNDARY OF PWR PLANTS

Summary:

The purpose of this bulletin is to: (1) notify licensees and construction permit holders about incidents of severe degradation of threaded fasteners (bolts and studs) in closures in the reactor coolant pressure boundary (RCPB), and (2) to require appropriate actions.

The scope of action items listed below is limited to the RCPB. Included are the threaded fasteners (studs or bolts) in (1) steam generator and pressurizer manway closures, (2) valve bonnets, and pump flange connections installed on lines having a nominal diameter of 6 inches or greater and (3) control rod drive (CRD) flange and pressurizer heater connections that do not have seal welds to provide leak-tight integrity. That is, CRDs having an omega seal weld design are excluded from this bulletin action. The reactor vessel head closure studs are also excluded for those PWR licensees committed to the provisions of Regulatory Guide 1.65, "Materials and Inspection for Reactor Vessel Closure Studs."

Required actions included:

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1. Where procedures do not exist, develop and implement maintenance procedures for threaded fastener practices.

2. Threaded fasteners of closure connections, identified in the scope of this bulletin, when opened for component inspection or maintenance shall be removed, cleaned, and inspected per IWA-2210 and IWA-2220 of ASME Code Section XI (1974 edition or later) before being reused.

Potential Impact Number: 14407 **SRP Section Number:** 5.2.4

Document: NRC BULLETIN 82-03 **Revision:** 1 , 10/28/82
Title: STRESS CORROSION CRACKING IN THICK-WALL, LARGE-DIAMETER, STAINLESS STEEL, RECIRCULATION SYSTEM PIPING AT BWR PLANTS

Summary:

IGSCC in large-diameter recirculation piping of BWRs described. Licensees asked to demonstrate effectiveness of UT methodology for examining welds in recirculation piping. Results of exams on recirculation piping requested by NRC along with list of corrective actions. Information on weld sampling plan, description of UT methods, and estimate of effectiveness requested by NRC.

Potential Impact Number: 14415 **SRP Section Number:** 5.2.4

Document: NRC BULLETIN 83-02
Title: STRESS CORROSION CRACKING IN LARGE-DIAMETER STAINLESS STEEL RECIRCULATION SYSTEM PIPING AT BWR PLANTS

Summary:

Instances of IGSCC in large-diameter recirculation piping at BWRs reviewed. Licensees requested to (1) demonstrate effectiveness of UT methodology used for pipe inspection, (2) augment ISI Program, (3) report results and corrective actions, (4) retain ISI records, and (5) prepare a written report.

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Potential Impact Number: 14507 **SRP Section Number:** 15.6.5.A

Document: RG 1.4
Title: ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS

Summary:

This guide gives assumptions that may be used in evaluating the radiological consequences of this LOCA for a PWR.

Potential Impact Number: 14516 **SRP Section Number:** 15.6.5.A

Document: RG 1.3
Title: ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT FOR BOILING WATER REACTORS

Summary:

This guide gives acceptable assumptions that may be used in evaluating the radiological consequences of a LOCA for BWRs.

The assumptions related to the release of radioactive material from the fuel and containment include not taking credit for retention of iodine in the suppression pool.

Potential Impact Number: 14678 **SRP Section Number:** 5.4.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue addresses risks and consequences of steam generator and steam line overfills. This issue also incorporated Issues 66 and 67 to address steam generator

and steam system integrity in a single issue. This issue consists of 4 tasks. Task 3 contains, in part, issues related to steam generator tube inspection (previously issue 67). This issue was given a MEDIUM priority ranking and later resolved with no new requirements.

Potential Impact Number: 14683 **SRP Section Number:** 15.6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Resolution of Issue 135 was expected to provide better understanding of steam generator and secondary steam integrity, including the effects of water hammer on secondary system components and piping as well as the resultant radiological consequences. The Issue involved four tasks, including:

Task 2 was intended to review the results and conclusions of studies on SGTR and propose specific modifications to SRP 15.6.3 including tube integrity, operator action time, and offsite dose limits.

Task 3 was to reassess the following concerns in Issue 67 for potential inclusion in an integrated resolution: reassessment of radiological consequences, reevaluation of design basis SGTR, supplemental tube inspections, integrity of steam generator tube sleeves, denting criteria, improved accident monitoring, reactor vessel inventory measurement, RCP trip, control room design review, EOPs, organizational responses, and RCS pressure control.

Potential Impact Number: 14687 **SRP Section Number:** 2.2.1 - 2.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Certain plants employ water chemistry systems and low-level waste volume reduction systems which require large quantities of liquid hydrogen, liquid oxygen or propane.

Little guidance on the storage and use of these materials is available in the SRP or in Reg. Guide 1.91. However, NRR considers this issue resolved due to the publication in September, 1987 of EPRI NP-5283-SR-A, 'Guidelines for Permanent BWR Hydrogen Water Chemistry Installation - 1987 Revision.'

The second system is a portable incinerator called a Mobile Volume Reduction System (MVRS) for use as a waste volume reduction process for low level radioactive waste. Commonwealth Edison has applied to the NRC for installation and operation of an MVRS at their Dresden 2/3 site. The MVRS uses large quantities (up to 1000 gallons) of LPG (propane) for incineration of radioactive waste. The staff has completed its review of the Dresden application for modification to their operating license and has approved the siting and use of the MRVS at the Dresden 2/3 sites in a SER published on August 13, 1986.

Thus, resolution of this issue is assumed to be the modification of SRP Sections 2.2.1, 2.2.2, and 2.2.3, and Regulatory Guide 1.91, as appropriate, to incorporate the staff acceptance criteria developed through the Dresden 2/3 MVRS review and the BWROG guidelines topical report review.

Potential Impact Number: 14705 **SRP Section Number:** 6.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Carbon steel piping in which high energy single-phase fluid flows have been observed to experience erosion of its walls over time. As a result of two ruptures of such pipes, the following measures were called for: (1) more stringent inspection of carbon steel piping; and (2) replacement of all piping that does not meet minimum thickness standards. This issue has been resolved.

Potential Impact Number: 14772 **SRP Section Number:** 9.3.1

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

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Summary:

Safety analysis reports claim that air or nitrogen accumulators for the automatic depressurization system (ADS) valves are provided with sufficient capacity to cycle the valves open five times at design pressures. GE has also stated that the emergency core cooling (ECC) systems are designed to withstand a hostile environment and still perform their function for 100 days following an accident. Licensee should verify that the accumulators on the ADS valves meet these requirements, even considering normal leakage. If this cannot be demonstrated, the licensee must show that the accumulator design is still acceptable.

The ADS valves, accumulators, and associated equipment and instrumentation must be capable of performing their functions during and following exposures to hostile environments and taking no credit for nonsafety-related equipment or instrumentation.

This NUREG-0737 potential impact was found through a direct search of 0737, and not through a cited reference in the MBD data base. NUREG-0737 is located on the MDB data base.

Potential Impact Number: 14829 **SRP Section Number:** 3.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue was concerned with the use of non-safety grade power operated relief valves and block valves for mitigating design basis accidents. This issue was resolved by requiring revisions to the technical specifications for PORVs and block valves, and revisions to SRP sections 3.2.2, 5.2.2, and 5.4.7 were forwarded to NRR for inclusion into the SRP.

Potential Impact Number: 14875 **SRP Section Number:** 4.4

Document: NRC GENERIC LETTER 86-09
Title: TECHNICAL RESOLUTION OF GENERIC ISSUE NO. B-59-(N-1) LOOP OPERATION IN BWRs AND PWRs

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Summary:

The majority of the presently operating BWRs-and PWRs are designed to operate with less than full reactor coolant flow. If a PWR reactor coolant pump or a BWR recirculation pump becomes inoperative, the flow provided by the remaining (N-1) loops is sufficient for steady state operation at a power level less than full power. The NRC staff has disallowed this mode of operation for most plants primarily because of insufficient ECCS analyses as well- as thermal-hydraulic stability concerns associated with BWRs.

The staff recently completed a safety evaluation report (SER) for the request by Beaver Valley Unit 1 for (N-1) loop operation. The review by the staff of the Beaver Valley Unit 1 submittal has not identified any unacceptable consequences associated with (N-1) loop operation. Therefore, other PWR owners may wish to evaluate the merits of (N-1) loop submittals for their plants based on the Beaver Valley 1 experience or on the approval of (N-1) loop operation for Millstone 3. However, the specific design characteristics of each plant must be reviewed in the same detail as Beaver Valley 1 and Millstone 3 to assure that all safety considerations relative to (N-1) loop operation are evaluated.

The review of BWR (N-1) loop operation has been complicated by potential thermal-hydraulic instability and jet pump vibration problems during single loop operation (SLO). In low flow operating regions, it has been necessary to develop special operating procedures to assure that General Design Criteria 10 and 12 are satisfied in regard to thermal-hydraulic instabilities. Technical Specifications consistent with these procedures have been accepted by the staff for reactors which are not demonstrably stable based on analyses using approved analytical methods. Tests performed at Browns Ferry Unit 1 demonstrated that SLO has similar stability characteristics as two loop operation under the same power/flow operating conditions. They also confirmed the staff's finding that Technical Specifications which have been proposed for some BWRs are appropriate for the detection and suppression of thermal hydraulic instabilities. Permanent SLO has been approved for Duane Arnold, and the staff expects to approve permanent SLO for other owners who have submitted SLO ECCS analysis as soon as Technical Specification changes similar to those for Duane Arnold are submitted.

Potential Impact Number: 14877 **SRP Section Number:** 4.3

Document: CODE OF FED. REGS 10CFR50

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Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50, Appendix A - GDC 28

Summary:

Limits on reactivity insertion rate and amount shall maintain core safety.

Potential Impact Number: 14884 **SRP Section Number:** 4.4

Document: NRC BULLETIN 88-07 Sup 1
Title: POWER OSCILLATIONS IN BOILING WATER REACTORS (BWRs)

Summary:

The preliminary results of these evaluations and the NRC's ongoing independent study indicate that when regional power oscillations become detectable on the average power range monitors (APRMs), the MCPR safety margin may be insufficient under some operating conditions to ensure that manual actions taken in response to APRM signals would prevent violation of the MCPR safety limit. In addition, design features of many reactor protection systems (e.g., filtered APRM signals to provide a Simulated Thermal Power Monitor in the power/flow scram circuit) render them ineffective for automatic scram protection during regional thermal hydraulic instabilities.

Potential Impact Number: 14890 **SRP Section Number:** 3.4.1

Document: NRC BULLETIN 80-24
Title: PREVENTION OF DAMAGE DUE TO WATER LEAKAGE INSIDE CONTAINMENT (OCTOBER 17, 1980 INDIAN POINT 2 EVENT)

Summary:

Several inches of water had accumulated on the containment floor at Indian Point Unit 2 without the operator's knowledge. This accumulation was later determined to have amounted to over 100,000 gallons, which flooded the reactor vessel pit and wetted the lower nine feet of the reactor vessel while the reactor was at operating temperature. Several actions including increased surveillance for flooding of the containment are requested from licensees to preclude similar events at their facilities.

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Potential Impact Number: 14902 **SRP Section Number:** 5.4.1.1

Document: RG 1.14 **Revision:** 1
Title: REACTOR COOLANT PUMP FLYWHEEL INTEGRITY

Summary:

ASME SA-533-B Class 1, and SA 508 Classes 2 and 3 generally have suitable toughness for typical flywheel applications.

Potential Impact Number: 14911 **SRP Section Number:** 3.2.2

Document: C&S: API 620
Title: RECOMMENDED RULES FOR DESIGN AND CONSTRUCTION OF LARGE, WELDED, LOW PRESSURE STORAGE TANKS

Summary:

API 620, sixth edition - 1977 is referenced in SRP Section 3.2.2 as the acceptable standard for Quality Group D storage tanks with pressure rating in the range of 0-15 psig.

Potential Impact Number: 14913 **SRP Section Number:** 3.2.2

Document: C&S: API 650
Title: WELDED STEEL TANKS FOR OIL STORAGE

Summary:

API 650, sixth edition, Revision 1 - 1978 is referenced in SRP Section 3.2.2 as one of the acceptable standards for Quality Group D atmospheric storage tanks.

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Potential Impact Number: 14915 **SRP Section Number:** 3.2.2

Document: C&S: AWWA D100
Title: WELDED STEEL TANKS FOR WATER STORAGE

Summary:

AWWA D-100 - 1979 is referenced in SRP Section 3.2.2 as one of the acceptable standards for Quality Group D atmospheric storage tanks.

Potential Impact Number: 14918 **SRP Section Number:** 3.2.2

Document: C&S: ANSI B.96.1
Title: WELDED ALUMINUM - ALLOY STORAGE TANKS

Summary:

ANSI B.96.1 - 1980 is referenced in SRP Section 3.2.2 as one of the acceptable standards for Quality Group D atmospheric storage tanks.

Potential Impact Number: 14923 **SRP Section Number:** 3.2.2

Document: C&S: ANSI B.31.1 Revision Date: 1/1/73
Title: POWER PIPING (1973)

Summary:

Section 3.2.2 of SRP cites paragraph 136.4 of ANSI B.31.1 - 1973 in appendices A and B as an acceptable approach for pressure boundary examination and acceptance for certain portions of BWR main steam systems designated quality group 'D+QA'.

Potential Impact Number: 14924 **SRP Section Number:** 3.2.2

Document: C&S: ANSI B.31.1 Revision Date: 1/1/80

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Title: POWER PIPING (1980)

Summary:

ANSI B.31.1 - 1980 is referenced in SRP Section 3.2.2 as the acceptable standard for Quality Group D piping and valve components.

Potential Impact Number: 14925 **SRP Section Number:** 3.2.2

Document: C&S: ASME III
Title: CLASS 1 AND 2 SYSTEMS

Summary:

ASME BPVC , Section III - 1980 is cited in SRP section 3.2.2 as code applicable to Quality Group A, B, and C systems and components.

Potential Impact Number: 14927 **SRP Section Number:** 3.2.2

Document: C&S: ASME XI
Title: INSERVICE INSPECTION

Summary:

ASME BPVC, Section XI is cited in SRP Section 3.2.2 appendix A as containing inservice inspection requirements applicable to portions of BWR main steam piping.

Potential Impact Number: 14932 **SRP Section Number:** 5.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

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This issue was concerned with the use of non-safety grade power operated relief valves and block valves for mitigating design basis accidents. This issue was resolved by requiring revisions to the technical specifications for PORVs and block valves, and revisions to SRP sections 3.2.2, 5.2.2, and 5.4.7 were forwarded to NRR for inclusion into the SRP.

Potential Impact Number: 14934 **SRP Section Number:** 5.4.7

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue was concerned with the use of non-safety grade power operated relief valves and block valves for mitigating design basis accidents. This issue was resolved by requiring revisions to the technical specifications for PORVs and block valves, and revisions to SRP sections 3.2.2, 5.2.2, and 5.4.7 were forwarded to NRR for inclusion into the SRP.

Potential Impact Number: 14935 **SRP Section Number:** 5.4.2.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.55a, (a)(1)

Summary:

Structures, systems, and components shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with importance of the safety function to be performed.

Potential Impact Number: 14936 **SRP Section Number:** 5.4.2.2

Document: CODE OF FED. REGS 10CFR50

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Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.55a, (b)(2)(iii)

Summary:

If the technical specifications of a nuclear power plant include surveillance requirements for steam generators different than those of Article IWB-2000, the inservice inspection program for steam generator tubing shall be governed by the requirements in the technical specifications.

Potential Impact Number: 14937 **SRP Section Number:** 5.4.2.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.55a, (b)(2)

Summary:

Gives inservice inspection requirements for Class 1 nuclear components.

Potential Impact Number: 14950 **SRP Section Number:** 5.4.2.2

Document: RG 1.83 **Revision:** 1
Title: INSERVICE INSPECTION OF PRESSURIZED WATER REACTOR STEAM GENERATOR TUBES

Summary:

Describes staff positions with regard to steam generator tube inspection programs.

Potential Impact Number: 14951 **SRP Section Number:** 5.4.2.2

Document: RG 1.121 **Revision:** 0
Title: BASES FOR PLUGGING DEGRADED PWR STEAM GENERATOR TUBES

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Summary:

This guide describes a method acceptable to the NRC staff for establishing the limiting safe conditions of tube degradation of steam generator tubing, beyond which defective tubes as established by inservice inspection should be removed from service by welding plugs at each end of the tube. This guide applies only to pressurized water reactors (PWRs).

Potential Impact Number: 14955 **SRP Section Number:** 5.4.2.2

Document: RG 1.147 **Revision:** 8
Title: INSERVICE INSPECTION CODE CASE ACCEPTABILITY ASME SECTION XI DIVISION 1

Summary:

This regulatory guide lists those Section XI ASME Code Cases that are generally acceptable to the NRC staff for implementation in the inservice inspection of light-water-cooled nuclear power plants. NRC acceptance of Code Cases N-401-1 and N-402 concerning use of digital ET equipment and ET calibration standard material are discussed.

Potential Impact Number: 14965 **SRP Section Number:** 5.4.2.2

Document: NRC GENERIC LETTER 85-02
Title: STAFF RECOMMENDED ACTIONS STEMMING FROM NRC INTEGRATED PROGRAM FOR THE RESOLUTION OF UNRESOLVED SAFETY ISSUES REGARDING STEAM GENERATOR TUBE INTEGRITY

Summary:

NRC requested information on licensee's overall program for maintaining steam generator tube integrity and steam generator tube rupture mitigation. This letter contains staff recommendations based on work performed for USIs A-3, A-4, and A-5 with regard steam generator tube inspections in Enclosures 1 and 2 to the letter.

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Potential Impact Number: 14968 **SRP Section Number:** 5.4.2.2

Document: NRC GENERIC LETTER 91-04
Title: CHANGES IN TECHNICAL SPECIFICATION SURVEILLANCE INTERVALS TO ACCOMMODATE A 24-MONTH FUEL CYCLE

Summary:

Provides guidance to licensees on license amendment requests for changes in surveillance intervals to accommodate a 24-month fuel cycle.

Potential Impact Number: 15072 **SRP Section Number:** 15.6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Following an STGR, the affected steam generator could overfill due to primary-to-secondary leakage and continued operation of the safety injection pumps.

Potential Impact Number: 15082 **SRP Section Number:** 6.7

Document: NRC GENERIC LETTER 86-17
Title: AVAILABILITY OF NUREG-1169. 'TECHNICAL FINDINGS RELATED TO GENERIC ISSUE C-8; BOILING WATER REACTOR MAIN STEAM ISOLATION VALVE LEAKAGE AND LEAKAGE TREATMENT METHODS'

Summary:

This letter is to inform the licensees that NUREG 1169 was published in June, 1986. This NUREG presents the results of the NRC staff and contractor efforts to assess Generic Issue C-8 'Main Steam Isolation Valves Leakage and Leakage Control System Failures.' NUREG-1169 represents technical findings associated with Generic Issue C-8 and the staff expects it to be used in the regulatory resolution of the issue. Neither this letter nor NUREG-1169 constitute a change in the existing generic staff positions or

allowable MSIV leakage rates, nor the specifications for MSIV leakage control systems. Therefore, no action is being requested of applicants or licensees.

This potential impact was identified on the basis that NUREG-1169 provides substantial background on MSIVLCS issues including evaluations related to the elimination of MSIVLCS.

Potential Impact Number: 15133 **SRP Section Number:** 15.6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

SRP Section 15.6.3 does not address a steam generator overfill in the SGTR scenario. Also, the assumption in SRP Section 15.6.3 that the leak from an SGTR will be terminated within 30 minutes is not consistent with operating experiences. This issue would upgrade SRP Section 15.6.3 to provide a better assessment of the radiological consequences of an SGTR event.

Potential Impact Number: 15137 **SRP Section Number:** 15.6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue calls for the NRC staff to consider reevaluation, reclassification and redefinition of the design basis for the SGTR event. This recommendation is due to concern over the number of SGTR events and the potential for exceeding the radiological limits of 10CFR100. This has been identified as a Licensing Issue.

Potential Impact Number: 15164 **SRP Section Number:** 5.4.2.2

Document: NUREG 0933

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Title: A Prioritization of Generic Safety Issues

Summary:

Improved Eddy Current Testing could enhance earlier detection of steam generator tube defects and thereby minimize, or mitigate, steam generator tube degradations and ruptures. Resolution of this issue has been integrated into the resolution of Issue 135.

Potential Impact Number: 15190 **SRP Section Number:** 5.4.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

It is the current NRC staff position that some form of supplement steam generator tube inspection can be formulated that would provide added assurance of steam generator tube integrity without significantly increasing occupational radiation exposures. This issue has been classified as a licensing issue.

Potential Impact Number: 15208 **SRP Section Number:** 3.4.1

Document: NRC GENERIC LETTER 89-18
Title: RESOLUTION OF UNRESOLVED SAFETY ISSUE A-17, 'SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS'

Summary:

This generic letter informs licensees and applicants of the final resolution of USI A-17, Systems Interactions in Nuclear Power Plants. The following staff action is involved in the resolution for future plants: Develop a standard review plan for future plants to address protection from internal flooding and water intrusion.

Potential Impact Number: 15325 **SRP Section Number:** 6.4

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Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue was raised in response to a 1982 ACRS letter which outlined deficiencies in the design, installation, maintenance and testing of control room systems for maintaining operator habitability. The solution to this issue 'has been identified.'

Potential Impact Number: 15329 **SRP Section Number:** 9.4.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue was raised in response to a 1982 ACRS letter which outlined deficiencies in the design, installation, maintenance and testing of control room systems for maintaining operator habitability. The solution to this issue 'has been identified.'

Potential Impact Number: 15351 **SRP Section Number:** 5.4.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

An unisolated break of a high pressure coolant injection steam supply line in Mark I or Mark II containment suppression pools could cause the environment in emergency core cooling pumps to exceed design limits. Based on the results of a value/impact analysis, this issue was assigned a high priority towards resolution.

This issue was resolved and requirements were issued to licensees in Generic Letter 89-10.

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Potential Impact Number: 15399 **SRP Section Number:** 10.4.9

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Steam binding of auxiliary feedwater pumps can occur when heated main feedwater leaks into these pumps (through a series of check valves) and then flashes to steam. This issue had an interim resolution with the issuance of IE Bulletin 85-01 that established requirements to develop procedures to detect and correct steam binding.

Potential Impact Number: 15400 **SRP Section Number:** 5.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

A number of over pressure transient events, have suggested potential weaknesses in existing overpressure protection criteria or in their implementation. This issue was resolved with the issuance of Generic Letter 90-06, which required a revision to the technical specifications for overpressure protection at certain plants.

Potential Impact Number: 15414 **SRP Section Number:** 5.4.7

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue was raised as a result of probabilistic risk assessments which indicated that the dominant interfacing system LOCA events were through the RHR suction valves. This issue was incorporated into the resolution of New Generic Issue 105, 'Interfacing Systems LOCA at LWRs.'

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Potential Impact Number: 15435 **SRP Section Number:** 5.4.7

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue applies to all Westinghouse PWRs. The loss of one instrument bus or logic channel will result in the automatic closure of one of the RHR suction line isolation valves. This could result in RHR pump damage and loss of decay heat removal by RHR. This issue was resolved; new requirements were established in Generic Letter 88-17.

Potential Impact Number: 15464 **SRP Section Number:** 3.9.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Issue B-63, which was resolved and implemented as MPA B-45, required leak testing of the check valves that isolate those low pressure systems that are connected to the RCS outside the containment. Operating experience indicated that the isolation valves between the RCS and low pressure interfacing systems (including related test and maintenance requirements) may not adequately protect against overpressurization of low pressure systems, thus Generic Issue 105 was established.

GI 105 addresses rupture of low pressure piping systems which interface with the reactor coolant pressure boundary due to pressure isolation system failure.

The study of ISLOCA at a BWR confirmed past PRA studies which generally indicated little risk contribution from ISLOCA sequences. The staff found that ISLOCAs at PWRs were plant-specific in nature; however, the ongoing IPE program includes licensee analysis of ISLOCA sequences. With respect to future applicants, a draft SRP Section covering design review of systems interfacing with the RCS in ALWRs was provided to NRR for information and use as appropriate. A supplement to Information Notice 92- 36 was also recommended to share insights from the ISLOCA program and to inform

licensees of the availability of material useful for IPE ISLOCA analyses not yet completed, or as a check on analyses already completed. Thus, this issue was RESOLVED and no new requirements were established.

Potential Impact Number: 15465 **SRP Section Number:** 5.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue addresses rupture of low pressure piping systems which interface with the reactor coolant pressure boundary due to pressure isolation system failure. This issue was assigned a high priority for resolution.

Potential Impact Number: 15469 **SRP Section Number:** 5.4.7

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue addresses rupture of low pressure piping systems which interface with the reactor coolant pressure boundary due to pressure isolation system failure. This issue was assigned a high priority for resolution. The resolution of this issue requires evolutionary light-water reactor designs to reduce the possibility of a 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.

Potential Impact Number: 15470 **SRP Section Number:** 6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

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Summary:

This issue addresses rupture of low pressure piping systems which interface with the reactor coolant pressure boundary due to pressure isolation system failure. This issue was assigned a high priority for resolution. In addition to the plant specific IPE related resolution for existing licensees, a proposed SRP section will be developed covering ISLOCA reviews for evolutionary ALWRs.

Potential Impact Number: 15482 **SRP Section Number:** 5.4.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue concerns requirements which were developed by the NRC staff as a result of a steam generator tube rupture event at the Ginna Plant in January 1982. In view of the resolution of Issues A-3, A-4, and A-5, the NRC determined that no new requirements for steam generators were necessary. This issue was resolved, and no new requirements were established.

Potential Impact Number: 15509 **SRP Section Number:** 9.2.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue raised concerns relative to multiplant units that have only two ESW pumps per plant but have crosstie capabilities. Service water pumps may not always be available during all possible operating conditions increasing vulnerability to core-melt sequences at the site.

Potential Impact Number: 15544 **SRP Section Number:** 3.9.3

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Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue was raised to address the NRC staff's concerns that there were no NRC requirements for dynamic qualification or surveillance testing of large bore hydraulic snubbers.

In resolving this issue, the staff concluded that there were few cost-beneficial changes to existing requirements that would result in improved large bore hydraulic snubber reliability and reduced risk. However it was recommended that a Regulatory Guide be developed for future plants. Thus, the issue was resolved and no new requirements were established for existing plants.

Potential Impact Number: 15572 **SRP Section Number:** 3.6.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This task responds to recommendations from the Piping Review Committee to change SRP Sections
3.6.1, 3.6.2 and 3.9.3 to include more realistic criteria regarding pipe rupture requirements and seismic/LOCA loads. This is a regulatory impact issue.

Potential Impact Number: 15575 **SRP Section Number:** 3.7.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This task responds to recommendations from the Piping Review Committee. It proposes to revise Regulatory Guides 1.61 and 1.84, and SRP Section 3.9.2 to incorporate ASME Code Case N-411-1 and/or new positions on pipe damping for

high-frequency loads and for time-history analyses. This is a regulatory potential impact issue.

Potential Impact Number: 15583 **SRP Section Number:** 9.5.5

Document: RG 1.9
Title: SELECTION, DESIGN, AND QUALIFICATION OF DIESEL-GENERATOR UNITS USED AS STANDBY (ONSITE) ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

Summary:

Diesel generator shall meet the requirements of IEEE Standard 387-1984.

Regulatory Position C-1.5 through C-1.8 includes positions on design features to be provided by the diesel generator unit and its support systems. These sections supplement the IEEE requirements for surveillance systems and diesel generator protective trips.

Potential Impact Number: 15591 **SRP Section Number:** 2.3.1

Document: RG 1.76 **Revision:** 0
Title: DESIGN BASIS TORNADO FOR NUCLEAR POWER PLANTS

Summary:

This guide describes a design basis tornado acceptable to the Regulatory staff for each of three regions within the contiguous United States that a nuclear power plant should be designed to withstand without undue risk to the health and safety of the public.

SRP-UDP has previously identified staff positions in evolutionary plant SERs and related SECY papers suggesting that Reg. Guide 1.76 guidance should be revised as a future work item (See IPD-7.0 Need Number 3.5.1.4-1, SRP Section 3.5.1.4 Integrated Impacts, and related Potential Impact Numbers 20729, 22633, 22635, and 22636 for further details).

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Potential Impact Number: 15754 **SRP Section Number:** 3.7.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This task is in response to a Piping Review Committee recommendation that the limit on vibratory ground acceleration for an operating basis earthquake be decoupled from the limit for a safe shutdown earthquake. There is no technical basis for linking OBE to SSE. This is a regulatory impact issue.

Potential Impact Number: 15761 **SRP Section Number:** 5.2.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This item responds to a Piping Review Committee recommendation that regular grade 316 SS and 304 SS materials in BWR recirculation piping be replaced with an alloy which is resistant to intergranular stress corrosion cracking. This item is a regulatory impact issue that requires updating of Regulatory Guide 1.44 to reflect the staff's findings in NUREG-0313, Rev. 2.

Potential Impact Number: 15762 **SRP Section Number:** 5.3.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This item responds to a Piping Review Committee recommendation that regular grade 316 SS and 304 SS materials in BWR recirculation piping be replaced with an alloy which is resistant to intergranular stress corrosion cracking. This item is a regulatory

impact issue that requires updating of Regulatory Guide 1.44 to reflect the staff's findings in NUREG-0313, Rev. 2.

Potential Impact Number: 15763 **SRP Section Number:** 6.1.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This item responds to a Piping Review Committee recommendation that regular grade 316 SS and 304 SS materials in BWR recirculation piping be replaced with an alloy which is resistant to intergranular stress corrosion cracking. This item is a regulatory impact issue that requires updating of Regulatory Guide 1.44 to reflect the staff's findings in NUREG-0313, Rev. 2.

Potential Impact Number: 15764 **SRP Section Number:** 10.3.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This item responds to a Piping Review Committee recommendation that regular grade 316 SS and 304 SS materials in BWR recirculation piping be replaced with an alloy which is resistant to intergranular stress corrosion cracking. This item is a regulatory impact issue that requires updating of Regulatory Guide 1.44 to reflect the staff's findings in NUREG-0313, Rev. 2.

Potential Impact Number: 15871 **SRP Section Number:** 12.3 - 12.4

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION

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Summary:

On May 21, 1991, the Nuclear Regulatory Commission issued a revision to its standards for protection against ionizing radiation, 10 CFR Part 20. Although the Revised Part (20.1001 through 20.2401, together with new appendices) became effective on June 20, 1991, licensees have been permitted to defer implementation of the revised rule until January 1, 1994, with earlier compliance encouraged.

Potential Impact Number: 15880 **SRP Section Number:** 13.4

Document: RG 1.33 **Revision:** 2
Title: QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION)

Summary:

The overall QA program requirements for the operation phase that are included in ANSI N18.7-1976/ANS-3.2 are acceptable to the NRC and provide an adequate basis for complying with most QA requirements in Appendix B of 10CFR50.

Potential Impact Number: 15883 **SRP Section Number:** 13.4

Document: RG 1.33 **Revision:** 2
Title: QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION)

Summary:

Audits of selected aspects with a frequency commensurate with their safety significance and such that an audit of all safety-related functions is completed within a period of two years.

Potential Impact Number: 15885 **SRP Section Number:** 13.4

Document: RG 1.33 **Revision:** 2
Title: QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION)

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Summary:

Most of the Section 4.4 guidelines in ANSI N18.7-1976/ANS-3.2, including those concerning review activities of the onsite operating organization, have sufficient safety importance to be treated as requirements.

Potential Impact Number: 15905 **SRP Section Number:** 12.5

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION

Summary:

On May 21, 1991, the Nuclear Regulatory Commission issued a revision to its standards for protection against ionizing radiation, 10 CFR Part 20. Although the Revised Part (20.1001 through 20.2401, together with new appendices) became effective on June 20, 1991, licensees have been permitted to defer implementation of the revised rule until January 1, 1994, with earlier compliance encouraged.

Potential Impact Number: 16077 **SRP Section Number:** 13.4

Document: NRC GENERIC LETTER 90-04
Title: REQUEST FOR INFORMATION ON THE STATUS OF LICENSEE IMPLEMENTATION OF GENERIC SAFETY ISSUES RESOLVED WITH IMPOSITION OF REQUIREMENTS OR CORRECTIVE ACTIONS

Summary:

Licensees are requested to describe their programs for ensuring that unscheduled shutdowns are properly analyzed; the analyses should include criteria for determining the need for an independent assessment of an event.

Potential Impact Number: 16127 **SRP Section Number:** 13.5.1.1

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Document: NRC GENERIC LETTER 83-28
Title: REQUIRED ACTIONS BASED ON GENERIC IMPLICATIONS OF SALEM ATWS EVENTS

Summary:

For components required to trip the reactor, licensees and applicants shall establish a program to ensure that vendor information is complete, current, controlled, and appropriately referenced in procedures.

Potential Impact Number: 16333 **SRP Section Number:** 13.2.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(i)

Summary:

For an LWR CP, provide simulator capability that correctly models the CR and includes the capability to simulate small-break LOCAs.

Potential Impact Number: 16337 **SRP Section Number:** 13.2.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(i)

Summary:

CP applicants shall describe a simulator capability, which correctly models the CR and includes the capability to simulate SBLOCAs, that will be operable by the OL stage.

Potential Impact Number: 16374 **SRP Section Number:** 13.2.1

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Document: CODE OF FED. REGS 10CFR55
Title: OPERATORS' LICENSES
Location: Sections 55.43(a)

Summary:

The written examination for SROs will query the knowledge, skills, and abilities needed to perform SRO duties. It will be based, in part, on task analyses, the training program, and information in the FSAR.

Potential Impact Number: 16375 **SRP Section Number:** 13.2.1

Document: CODE OF FED. REGS 10CFR55
Title: OPERATORS' LICENSES
Location: Sections 55.43(b)

Summary:

The written examination for SROs will include a representative sample of items, including facility operating limitations in the Tech Specs and their bases, radiation hazards, and fuel handling facilities and procedures.

Potential Impact Number: 16393 **SRP Section Number:** 13.2.1

Document: CODE OF FED. REGS 10CFR55
Title: OPERATORS' LICENSES
Location: Sections 55.45(b)(1)(i) and (ii)

Summary:

The operating test will be administered in a plant walk through and in an acceptable simulation facility.

Potential Impact Number: 16864 **SRP Section Number:** 3.4.1

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Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue is concerned with the identification and correction of detrimental interactions between safety systems. However, this issue is not considered as a separate generic issue since it is covered in Issue A-17.

Potential Impact Number: 16923 **SRP Section Number:** 12.5

Document: NRC BULLETIN 78-07
Title: PROTECTION AFFORDED BY AIR-LINE RESPIRATORS AND SUPPLIED AIR HOODS

Summary:

This bulletin cites LA-7098-PR dealing with protection factors for respirators and an informal report LA-NUREG-6612-MS on supplied air hoods.

The NRC will further evaluate the information in these reports and other pertinent information and will revise its guidance as appropriate, e.g., by reducing the protection factors permitted, or by eliminating the approvals for or specifying particular conditions for the use of such equipment.

Potential Impact Number: 17029 **SRP Section Number:** 5.2.2

Document: NRC GENERIC LETTER 90-06
Title: 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,'

Summary:

Where PORVs are used or could be used to perform one, or more, of the safety-related functions identified above or to perform any other safety-related function that may be

identified in the future, it is appropriate to reconsider the safety classification of PORVs and the associated block valves. For certain PWR plants receiving an operating license in recent years, the staff has required these valves to be classified as safety-related components if they perform one, or more, safety-related functions.

Section three of Enclosure A identifies staff positions/requirements resulting from the resolution of Generic Issue 70, PORV and Block Valve Reliability.

Enclosure B provides staff positions resulting from resolution of Generic Issue 94 Additional Low-Temperature Overpressure Protection for Light-Water Reactors.

Potential Impact Number: 17326 **SRP Section Number:** 3.5.3

Document: RG 1.115 **Revision:** 1
Title: PROTECTION AGAINST LOW-TRAJECTORY TURBINE MISSILES

Summary:

Some degree of protection against low-trajectory turbine missiles may be provided by barriers. There is no body of experimental evidence on the impact effects of large irregularly shaped missiles similar to those observed in turbine failures on either steel or reinforced concrete structures. Conservative damage predictions can be made by using results of tests and conservative assumptions. Some recent data were motivated by protection against tornado missiles. For cases in which the impact was normal and the impact area known, there was good correlation between actual penetration distances into reinforced concrete and those predicted by the Modified National Defense Research Council (NDRC) formula. Predictions of backface scabbing due to missile impact were not as good. For metal structures, application of the Ballistic Research Laboratory (BRL) formula should give conservative results for large missiles.

Potential Impact Number: 17332 **SRP Section Number:** 2.3.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix E, II H, Appendix E, II H, Appendix E, IV B, Appendix E, IV E.2

Summary:

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The Preliminary Safety Analysis Report shall contain a preliminary analysis reflecting the need to include facilities, systems, and methods for identifying the degree of seriousness and potential scope of radiological consequences of emergency situations within and outside the site boundary, including capabilities for dose projection using real-time meteorological information.

The Emergency Plan shall describe the means to be used for determining the magnitude of and for continually assessing the impact of the release of radioactive material.

Emergency facilities shall have adequate equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment.

Potential Impact Number: 17355 **SRP Section Number:** 2.3.3

Document: RG 1.97 **Revision:** 3
Title: INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT

Summary:

Establishes meteorological instrumentation as Type E, Category 3 accident monitoring instrumentation. Meteorological instrument indication parameters and design requirements are discussed.

Potential Impact Number: 17471 **SRP Section Number:** 13.2.2

Document: RG 1.8 **Revision:** 2 , 4/1/87
Title: QUALIFICATION AND TRAINING OF PERSONNEL FOR NUCLEAR POWER PLANTS

Summary:

Reg. Guide 1.8 is referred to in the SRP as a training standard that must be met, yet it has been superseded by Reg. Guide 1.8, Revision 2.

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Potential Impact Number: 17504 **SRP Section Number:** 13.2.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix E, II.F

Summary:

The PSAR shall describe provisions for a training program for licensee employees and for licensee non-employees who might assist in a radiological emergency.

Potential Impact Number: 17505 **SRP Section Number:** 13.2.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix E, IV.F

Summary:

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.

Potential Impact Number: 17535 **SRP Section Number:** 13.2.2

Document: NRC POLICY STATEMENT 50 FR 43621
Title: COMMISSION POLICY STATEMENT ON ENGINEERING EXPERTISE ON SHIFT

Summary:

While the NRC stresses the importance of engineering accident assessment expertise on shift, this statement requires no changes in the training of operators not filling the SRO/STA position.

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Potential Impact Number: 17537 **SRP Section Number:** 13.2.2

Document: NRC POLICY STATEMENT 50 FR 43621
Title: COMMISSION POLICY STATEMENT ON ENGINEERING EXPERTISE ON SHIFT

Summary:

The NRC's objective is to enhance engineering expertise on shift through more stringent educational requirements for the individual filling the SRO/STA position.

Potential Impact Number: 17679 **SRP Section Number:** 2.3.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.47 (b) (9)

Summary:

Onsite and offsite emergency response plans shall ensure adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

Potential Impact Number: 17996 **SRP Section Number:** 4.4

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Although the majority of presently operating BWRs and PWRs are designed to operate at reduced power if one of the coolant loops becomes inoperative, the NRC staff had disallowed this mode due to insufficient analyses.

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Potential Impacts

Potential Impact Number: 18045 **SRP Section Number:** 3.9.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue involved the review of the capabilities of isolating low pressure systems from the reactor coolant pressure boundary at a representative plant from each NSSS vendor.

The staff addressed this issue during review of CP/OL applications in 1980-81 using a proposed Revision 2 of SRP 3.9.6.

In April 1981, an order was issued to all operating reactors to comply with the requirements of the technical resolution for Event V configurations.

Potential Impact Number: 18204 **SRP Section Number:** 6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

The purpose of this issue was to aid the NRC staff in the review of changes to vendor ECCS models and in the performance of staff audit calculations of ECCS performance. This Regulatory Impact Issue has been determined to be resolved with the amendments to 10 CFR 50.46 and Appendix K.

Potential Impact Number: 18205 **SRP Section Number:** 15.6.5

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

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Summary:

The purpose of this issue was to aid the NRC staff in the review of changes to vendor ECCS models and in the performance of staff audit calculations of ECCS performance. This Regulatory Impact Issue has been determined to be resolved.

Potential Impact Number: 18209 **SRP Section Number:** 6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue involved following the work of research groups in determining best estimate decay heat data and associated uncertainties for use in LOCA calculations. This Regulatory Impact Issue has been resolved through the amendments to 10 CFR 50.46.

Potential Impact Number: 18211 **SRP Section Number:** 15.6.5

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue involved following the work of research groups in determining best estimate decay heat data and associated uncertainties for use in LOCA calculations. This Regulatory Impact Issue has been resolved.

Potential Impact Number: 18221 **SRP Section Number:** 6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

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Potential Impacts

This issue involved NRC staff evaluations of vendor's data and approaches for determining the effects of power density, decay heat, stored energy, and fission power decay on LOCA heat sources. This Regulatory Impact Issue has been resolved through the amendments to 10 CFR 50.46.

Potential Impact Number: 18225 **SRP Section Number:** 15.6.5

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue involved NRC staff evaluations of vendor's data and approaches for determining the effects of power density, decay heat, stored energy, and fission power decay on LOCA heat sources. This Regulatory Impact Issue has been resolved.

Potential Impact Number: 18311 **SRP Section Number:** 11.4

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue addresses the development of acceptance criteria for rad-waste solidification agents for the packaging of diverse plant wastes for shallow land burial. This issue was resolved with the publication of 10CFR61.56, which addresses waste characteristics.

Potential Impact Number: 18326 **SRP Section Number:** 13.5.1.1

Document: CODE OF FED. REGS 10CFR26
Title: FITNESS-FOR-DUTY PROGRAM
Location: 26.1

Summary:

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This Part 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.

Potential Impact Number: 18328 **SRP Section Number:** 13.5.1.1

Document: CODE OF FED. REGS 10CFR26
Title: FITNESS-FOR-DUTY PROGRAM
Location: 26.28

Summary:

Each licensee subject to this Part, and each contractor or vendor implementing a fitness-for-duty program under the provisions of section 26.23, shall establish a procedure for licensee and contractor or vendor employees to appeal a positive alcohol or drug determination. The procedure must provide notice and an opportunity to respond and may be an impartial internal management review. A licensee review procedure need not be provided to employees of contractors or vendors when the contractor or vendor is administering his own alcohol and drug testing.

Potential Impact Number: 18439 **SRP Section Number:** 13.5.1.1

Document: NRC POLICY STATEMENT 46 FR 23836
Title: NUCLEAR PLANT STAFF WORKING HOURS

Summary:

Licensees or applicants of operating plants shall establish controls to prevent situations where fatigue could reduce the ability of operating personnel to keep the plant in a safe condition. The controls should focus on shift staffing and the use of overtime.

Potential Impact Number: 18440 **SRP Section Number:** 13.5.1.1

Document: NRC POLICY STATEMENT 46 FR 23836

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Title: NUCLEAR PLANT STAFF WORKING HOURS

Summary:

Enough operating personnel should be employed to maintain coverage without routine heavy use of overtime. The objective is to have operating personnel work a normal 8 hour day; 40 hour week.

Potential Impact Number: 18441 **SRP Section Number:** 13.5.1.1

Document: NRC POLICY STATEMENT 46 FR 23836
Title: NUCLEAR PLANT STAFF WORKING HOURS

Summary:

Guidelines for overtime on temporary basis. An individual will work less than 16 hours straight, including shift turnover time, an individual will work no more than 16 hours in a 24 hour period, or 24 in a 48 hour period, or 72 hours in a 7 day period, a break of 8 hours between work periods including shift turnover, overtime should be decided on an individual basis.

Potential Impact Number: 18505 **SRP Section Number:** 17.1

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

Appendices A and B (of Reg. Guide 1.155) provide guidance on quality assurance (QA) activities and specifications respectively for non-safety-related equipment used to meet the requirements of 50.63 and not already covered by existing QA requirements in Appendix B or R of Part 50. Appropriate activities should be implemented from among those listed in these appendices depending on whether the non-safety equipment is being added (new) or is existing. This QA guidance is applicable to non-safety systems and equipment for meeting the requirements of 50.63 of 10 CFR Part 50. The guidance

on QA and specifications incorporates a lesser degree of stringency by eliminating requirements for involvement of parties outside the normal line organization. NRC inspections will focus on the implementation and effectiveness of the quality controls described in Appendices A and B. Additionally, the equipment installed to meet the station blackout rule must be implemented such that it does not degrade the existing safety-related systems. This is to be accomplished by making the nonsafety- related equipment as independent as practicable from existing safety-related systems. The non-safety systems identified in Appendix B are acceptable to the NRC staff for responding to a station blackout. A 10 point QA program is specified in Appendix A. The Recall ALLNOSRP index version of the Reg. Guide is missing Appendix B thus precluding analysis of further potential impacts.

Potential Impact Number: 18507 **SRP Section Number:** 17.1

Document: RG 1.143 **Revision:** 1
Title: DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

Summary:

Provides a QA program acceptable to the NRC staff for the design, construction, and testing of radwaste management systems. The QA program is reprinted from ANSI N199-1976/ANS-55.2, Liquid Radioactive Waste Processing Systems for Pressurized Water Reactor Plants.

Potential Impact Number: 18509 **SRP Section Number:** 17.1

Document: RG 1.88 **Revision:** 2
Title: COLLECTION, STORAGE, AND MAINTENANCE OF NUCLEAR POWER PLANT QUALITY ASSURANCE RECORDS

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to collection, storage, and maintenance of quality assurance records for all types of nuclear power plants. The requirements and

guidelines for collection, storage, and maintenance of nuclear power plant quality assurance records that are included in ANSI N45.2.9-1974 are acceptable to the NRC staff 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 regulatory positions. NFPA No. 232-0 1975, 'Standard for the Protection of Records,' is within its scope an acceptable alternative to subdivision of 5.6 of N4.5.2-1974.

Potential Impact Number: 18510 **SRP Section Number:** 17.1

Document: RG 1.97 **Revision:** 3
Title: INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-watercooled nuclear power plant. Table 1 provides a list of quality assurance-related Reg. Guides that should be followed (1.28, 1.30, 1.38, 1.58, 1.64, 1.88, 1.123, 1.144, 1.146) for Category 1 and some/most Category 2 instrumentation. A note at end of paragraph 5 implies that an in-progress Reg. Guide 1.28 revision will reduce this list to 1.30 and 1.38.

Potential Impact Number: 18518 **SRP Section Number:** 17.1

Document: RG 1.64 **Revision:** 2
Title: QUALITY ASSURANCE REQUIREMENTS FOR THE DESIGN OF NUCLEAR POWER PLANTS

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to quality assurance requirements for the design of all types of nuclear power plants. Requirements and recommendations for establishing and executing a QA program during the design phase that are in ANSI

N45.2.11-1974 are acceptable with modifications as described in the regulatory positions.

Potential Impact Number: 18536 **SRP Section Number:** 13.5.1.1

Document: RG 1.114 **Revision:** 2
Title: GUIDANCE TO OPERATORS AT THE CONTROLS AND TO SENIOR OPERATORS IN THE CONTROL ROOM OF A NUCLEAR POWER UNIT

Summary:

Defines an acceptable method to the NRS 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).

Potential Impact Number: 18569 **SRP Section Number:** 17.1

Document: RG 1.74 **Revision:** 0
Title: QUALITY ASSURANCE TERMS AND DEFINITIONS

Summary:

This guide identifies quality assurance terms and acceptable definitions that are important to the understanding of these quality assurance requirements for the design, construction, and operation of nuclear power plant structures, systems, and components. This guide applies to all types of nuclear power plants. The quality assurance terms and definitions contained in ANSI N45.2.10-1973, Quality Assurance Terms and Definitions, are generally acceptable for use in describing and implementing quality assurance programs for the design, construction, and operation of nuclear power plant structures, systems, and components subject to the following: The definition of procurement documents should be considered to include such documents as contracts, letters of intent, work orders, purchase orders, or proposals and their acceptances which authorize the seller to perform services or supply equipment, material, or facilities on behalf of the purchaser.

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Potential Impact Number: 18570 **SRP Section Number:** 17.1

Document: RG 1.146 **Revision:** 0
Title: QUALIFICATION OF QUALITY ASSURANCE PROGRAM AUDIT PERSONNEL FOR NUCLEAR POWER PLANTS

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to qualification of quality assurance program audit personnel for nuclear power plants. The requirements that are included in ANSI/ASME N45.2.23-1978 for qualification of quality assurance program audit personnel for nuclear power plants are acceptable to the NRC staff 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 regulatory positions.

Potential Impact Number: 18571 **SRP Section Number:** 17.1

Document: RG 1.144 **Revision:** 1
Title: AUDITING OF QUALITY ASSURANCE PROGRAMS FOR NUCLEAR POWER PLANTS

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to auditing of quality assurance programs for nuclear power plants. The requirements that are included in ANSI/ASME N45.2.12-1977 for auditing quality assurance programs for nuclear power plants are acceptable to the NRC staff and provide an adequate basis for complying with the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50, subject to several modifications described in the regulatory positions.

Potential Impact Number: 18577 **SRP Section Number:** 17.1

Document: RG 1.123 **Revision:** 1

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Title: QUALITY ASSURANCE REQUIREMENTS FOR CONTROL OF PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR POWER PLANTS

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to quality assurance requirements for control of procurement of items and services during the design, construction, and operations phases of nuclear power plants. The requirements that are included in ANSI N45.2.13-1976 for control of procurement of items and services for nuclear power plants are acceptable to the NRC staff and provide an adequate basis for complying with the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50, subject to modifications of the regulatory positions. The standard (ANSI N45.2.13-1976) does not provide requirements specific to spare and replacement parts. Section 5.2.13 of ANSI N18.7-1976/ANS 3.2, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, which is endorsed by Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation), addresses control of spare and replacement parts during the operations phase of nuclear power plants. As a result, the provisions of Section 5.2.13 of ANSI N18.7- 1976 related to control of spare and replacement parts are considered applicable and should be used in conjunction with the provisions of ANSI N45.2.13-1976. The Reg. Guide also recommends treatment of several of the guidelines (as indicated by the verb should) of the standard as requirements.

Potential Impact Number: 18578 **SRP Section Number:** 17.1

Document: RG 1.28 **Revision:** 3
Title: QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION)

Summary:

Basic and supplementary requirements of ANSI/ASME NQA-1 1983 and the 1a-1983 addenda thereto are acceptable for the establishment and execution of QA programs during the design and construction phases of nuclear power plants subject to identified additions and modifications (in the regulatory positions).

APPENDIX II
Potential Impacts

Potential Impact Number: 18579 **SRP Section Number:** 17.1

Document: RG 1.28 **Revision:** 3
Title: QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION)

Summary:

Appendix 2A-1 (providing non-mandatory guidance per the standard) of NQA-1, or acceptable alternatives, should be met (i.e. treated as requirements) as part of Supplement 2S-1 of NQA-1, 'Supplementary Requirements for the Qualification of Inspection and Test Personnel.'

Potential Impact Number: 18580 **SRP Section Number:** 17.1

Document: RG 1.28 **Revision:** 3
Title: QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION)

Summary:

Specifies the retention periods for programmatic and product nonpermanent records and when the retention period begin. References Table 1, of the guide, that provides a list of nonpermanent and lifetime records and their respective retention times. Section 2.8, Retention of Records, of Supplement 17S-1, Supplementary Requirements for Quality Assurance Records, states that the retention period for nonpermanent records is required to be established in writing. Programmatic nonpermanent records should be retained for at least 3 years and product nonpermanent records should be retained for at least 10 years or the life of the item if less than 10 years. For programmatic nonpermanent records, the retention period should be considered to begin upon completion of the activity. For product nonpermanent records generated before commercial operation begins, the retention period should be considered to begin upon completion of delivery. In addition, product and programmatic nonpermanent records should be retained at least until the date of issuance of the full power operating license of the unit. Table 1 provides a list of nonpermanent and lifetime records and their respective retention times. Although Table 1 is intended to be a comprehensive list, it is the responsibility of the owner to assure itself, in accordance with Criterion 17 of Appendix B to 10 CFR Part 50, that sufficient records are maintained to furnish evidence of activities affecting quality. It should be recognized that the nomenclature of

these records may vary. For records not listed in Table 1, the type most nearly describing the record in question should be followed with respect to its retention period.

Potential Impact Number: 18581 **SRP Section Number:** 17.1

Document: RG 1.28 **Revision:** 3
Title: QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION)

Summary:

Section 2, Scheduling, of Supplement 18S-1, Supplementary Requirements for Audits, requires audits to be scheduled in a manner that provides coverage and coordination with ongoing quality assurance program activities. The following guidelines are considered acceptable for scheduling audits: (Internal Audits) Applicable elements of an organization's quality assurance program should be audited at least once each year or at least once during the life of the activity, whichever is shorter. In determining the scope of the audit, an evaluation of the activity being audited may be useful. The evaluation may include results of previous quality assurance program audits and the results of audits from other sources, including the nature and frequency of identified deficiencies and any significant changes in personnel, organization, or quality assurance program.... (provides requirements for external audits which is the source of another impact)

Potential Impact Number: 18582 **SRP Section Number:** 17.1

Document: RG 1.28 **Revision:** 3
Title: QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION)

Summary:

Section 2, Scheduling, of Supplement 18S-1, Supplementary Requirements for Audits, requires audits to be scheduled in a manner that provides coverage and coordination with ongoing quality assurance program activities. The following guidelines are considered acceptable for scheduling audits: (External Audits) After the award of a contract, the applicant or licensee may determine, based on the evaluation conducted

in accordance with Section 5.1 of Appendix 4A-1, that external audits are not necessary for procuring items that are (1) relatively simple and standard in design, manufacturing, and testing and (2) adaptable to standard or automated inspections or tests of the end product to verify quality characteristics after delivery. For other procurement actions not covered by the above exceptions, audits should be conducted as described below. 1. The applicant or licensee should either audit its supplier's quality assurance program on a triennial basis; or arrange for such audit. In either case, the audit should be implemented in accordance with Supplement 18S-1 of ANSI/ASME NQA-1-1983. The triennial period begins when an audit is performed. An audit may be performed when the supplier has completed sufficient work to demonstrate that its organization is implementing a quality assurance program that has the required scope for purchases placed during the triennial period. If a subsequent contract or a contract modification significantly enlarges the scope of or changes the methods or controls for activities performed by the same supplier, an audit of the modified requirements should be conducted, thus starting a new triennial period. If the supplier is implementing the same quality assurance program for other customers that is proposed for use on the auditing party's contract, the pre-award survey may serve as the first triennial audit if conducted in accordance with the requirements of ANSI/ASME NQA 1- 1983. Therefore, when such pre-award surveys are employed as the first triennial audits, they should satisfy the same audit elements and criteria as those used on other triennial audits. 2. The applicant or licensee should perform or arrange for annual evaluations of suppliers. This evaluation should be documented and should take into account, where applicable, (1) review of supplier furnished documents and records such as certificates of conformance, nonconformance notices, and corrective actions, (2) results of previous source verifications, audits, and receiving inspections; (3) operating experience of identical or similar products furnished by the same supplier; and (4) results of audits from other sources, e.g., customer, ASME, or NRC audits. 3. If more than one purchaser buys from a single supplier, a purchaser may either perform or arrange for an audit of the supplier on behalf of itself and other purchasers to reduce the number of external audits of the supplier. The scope of this audit should satisfy the needs of all of the purchasers, and the audit report should be distributed to all the purchasers for whom the audit was conducted. Nevertheless, each of the purchasers relying on the results of an audit performed on behalf of several purchasers remains individually responsible for the adequacy of the audit.

Potential Impact Number: 18583 **SRP Section Number:** 17.1

Document: RG 1.28 **Revision:** 3

APPENDIX II
Potential Impacts

Title: QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION)

Summary:

Methods in Revision 3 of Reg. Guide 1.28 are considered to be generally equivalent to the methods in Revision 2 of Reg. Guide 1.28 and Reg. Guides 1.58, 1.64, 1.74, 1.88, 1.123, 1.144, and 1.146. Licensees and applicants may commit to either Revision 3 of Reg. Guide 1.28 or to Revision 2 of Reg. Guide 1.28 and Reg. Guides 1.58, 1.64, 1.74, 1.88, 1.123, 1.144, and 1.146.

Potential Impact Number: 18584 **SRP Section Number:** 17.1

Document: RG 1.28 **Revision:** 3
Title: QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION)

Summary:

Applicants and licensees committed to ANSI/ASME N45.2 and appropriate series standards as addressed in applicable Reg. Guides may continue to follow these standards instead of NQA-1. Licensees and applicants may commit to either Revision 3 of Reg. Guide 1.28 or to Revision 2 of Reg. Guide 1.28 and Reg. Guides 1.58, 1.64, 1.74, 1.88, 1.123, 1.144, and 1.146. Applicants and licensees may commit to follow either the ANSI/ASME N45.2-series standards or the ANSI/ASME NQA-1-1983 standard but not a combination of the two.

Potential Impact Number: 18587 **SRP Section Number:** 17.1

Document: RG 1.58 **Revision:** 1
Title: QUALIFICATION OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION, AND TESTING PERSONNEL

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to qualification of inspection, examination, and

testing personnel for all types of nuclear power plants. Requirements for qualification of nuclear plant inspection, examination, and testing personnel that are included in ANSI N45.2.6-1978 are acceptable to meet Appendix B subject to additions/modifications. Reg. Guide 1.8 guidance should be applied for personnel who approve or supervise preoperational startup and operational tests.

Potential Impact Number: 18608 **SRP Section Number:** 17.1

Document: NRC GENERIC LETTER 91-05
Title: LICENSEE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS

Summary:

Describes the characteristics of an effective commercial grade procurement and dedication program. This generic letter expresses staff positions regarding certain aspects of licensee commercial-grade procurement and dedication programs which would provide acceptable methods to meet regulatory requirements. The Nuclear Utility Management and Resources Council (NUMARC) Board of Directors recently approved a comprehensive procurement initiative as described in NUMARC 90-13, Nuclear Procurement Program Improvements, which commits licensees to assess their procurement programs and take specific action to enhance or upgrade the program if they are determined to be inadequate. The initiative on the dedication of commercial-grade items, which is part of NUMARC 90-13, was to be implemented by January 1, 1990. The staff position is that the staff will not initiate enforcement action in cases of past programmatic violations that have been adequately corrected. In addition, the staff does not expect licensees to review all past procurements. However, if during current procurement activities, licensees identify shortcomings in the form, fit, or function of specific vendor products, or if failure experience or current information on supplier adequacy indicates that a component may not be suitable for service, corrective actions are required for all such installed and stored items in accordance with Criterion XVI of 10 CFR Part 50, Appendix B. Also in accordance with Criterion XVI, licensees must determine programmatic causes when actual deficiencies in several products from different vendors are identified during current procurement activities and these deficiencies lead to the replacement of installed items as part of the corrective action. In such cases, a further sampling of previously procured commercial-grade items may be warranted. In NRC Generic Letter (GL) 89-02, Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products, the staff described its perspective on good practices in procurement and dedication and provided the NRC's

conditional endorsement of an industry standard (EPRI NP-5652) on methods of commercial-grade procurement and dedication. Several items discussed in letter 89-02 are clarified in letter 91-05.

Potential Impact Number: 18609 **SRP Section Number:** 17.1

Document: NRC GENERIC LETTER 89-02
Title: ACTIONS TO IMPROVE THE DETECTION OF COUNTERFEIT AND FRAUDULENTLY MARKETED PRODUCTS

Summary:

Provides NRC staff recommendations to reduce the likelihood of introducing counterfeit or fraudulent products into the plant. NRC staff conditionally endorses EPRI NP-5652 (June 1988) for evaluating commercial grade products for use in safety-related applications. Enclosure 1 lists previous generic communications on the subjects of non-conforming materials and equipment and instances of inadequate dedication of equipment for safety related applications.

Potential Impact Number: 18610 **SRP Section Number:** 17.1

Document: NRC GENERIC LETTER 88-18
Title: PLANT RECORD STORAGE ON OPTICAL DISKS

Summary:

NRC staff approves the use of optical disk document imaging systems for the storage of record copies of quality assurance records. Identifies appropriate quality controls for an optical disk imaging system.

Potential Impact Number: 18611 **SRP Section Number:** 17.1

Document: NRC GENERIC LETTER 85-06
Title: QUALITY ASSURANCE GUIDANCE FOR ATWS EQUIPMENT THAT IS NOT SAFETY-RELATED

APPENDIX II
Potential Impacts

Summary:

Provides staff Quality Assurance guidance for non safety related equipment encompassed by the ATWS rule (10 CFR 50.62). Guidance provided does not require licensee development of a new program for ATWS related equipment nor full compliance with 10 CFR 50 Appendix B. Quality program guidance is provided in a format comparable to the format of 10 CFR 50 Appendix B (although less stringent quality program requirements than Appendix B are provided in the guidance) to enable licensee incorporation of such ATWS equipment into existing quality assurance programs.

Potential Impact Number: 18612 **SRP Section Number:** 17.1

Document: NRC GENERIC LETTER 84-01
Title: NRC Use of the Terms, 'Important to Safety' and 'Safety Related'

Summary:

QA programs for plant design, construction on operation must meet the more general requirements of General Design Criteria 1 for plant equipment 'important to safety' and the more prescriptive requirements of Appendix B for 'safety-related' plant equipment. The Recall version of the letter in the ALLNOSRP index is missing the Enclosures thus precluding further identification of potential impacts.

Potential Impact Number: 18613 **SRP Section Number:** 17.1

Document: NRC GENERIC LETTER 82-21
Title: TECHNICAL SPECIFICATIONS FOR FIRE PROTECTION AUDITS

Summary:

Guidance on audits of fire protection program including qualification of auditors and scope of the audits. Includes excerpts from 'Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance' sent to licensees and applicants in 1977. The Recall version of the letter in the ALLNOSRP index is missing the Enclosures thus precluding further identification of potential impacts.

APPENDIX II
Potential Impacts

Potential Impact Number: 18652 **SRP Section Number:** 17.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.54 (a)(1)

Summary:

Each licensee shall implement, pursuant to Section 50.34 (b)(6)(ii) of this part, the QA program described or referenced in the SAR including changes to that report.

Potential Impact Number: 18719 **SRP Section Number:** 13.5.1.1

Document: NRC GENERIC LETTER 82-02
Title: Nuclear Power Plant Staff Working Hours

Summary:

Reg. Guide 1.33 and NUREG 0737 will be revised to reflect that administrative procedures will require deviation to the working hours guidelines be documented and available for NRC review.

Potential Impact Number: 18779 **SRP Section Number:** 4.4

Document: NRC GENERIC LETTER 86-02
Title: TECHNICAL RESOLUTION OF GENERIC ISSUE B-19-THERMAL HYDRAULIC STABILITY

Summary:

BWR 4, 5, 6s may not be able to show compliance with General Design Criteria 12 solely using analysis procedures to prove that thermal hydraulic instabilities are prevented by design. For cores which do not meet the analytical criteria, we have concluded that operating limitations 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, are acceptable to demonstrate compliance with GDC 12 for cores loaded with approved fuel designs.

Potential Impact Number: 18789 **SRP Section Number:** 4.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 12

Summary:

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Potential Impact Number: 18798 **SRP Section Number:** 8.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.63

Summary:

The station blackout rule for light water reactors. Each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout as defined in Section 50.2. The specified station blackout duration shall be based on the following factors: (i) The redundancy of the onsite emergency ac power sources; (ii) The reliability of the onsite emergency ac power sources; (iii) The expected frequency of loss of offsite power; and (iv) The probable time needed to restore offsite power. 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.

APPENDIX II
Potential Impacts

Potential Impact Number: 18806 **SRP Section Number:** 8.1

Document: RG 1.118 **Revision:** 2
Title: PERIODIC TESTING OF ELECTRIC POWER AND PROTECTION SYSTEMS

Summary:

Provides guidance for the periodic testing of electric power and protection systems. Also provides design guidance for the provision of testability features for such systems. The Reg. Guide is specified as guidance for SRP Sections 8.3.1 and 8.3.2 in Table 8-1. Guidance modifying the provisions of IEEE Standard 338-1977 is provided. The Reg. Guide is specified as guidance for SRP Sections 8.3.1 and 8.3.2 in Table 8-1.

Potential Impact Number: 18807 **SRP Section Number:** 8.1

Document: RG 1.128 **Revision:** 1
Title: INSTALLATION DESIGN AND INSTALLATION OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS

Summary:

The Reg. Guide supplements the guidance of IEEE Standard 484-1975 covering the installation design and installation of large lead acid storage batteries for generating stations and substations. Some provisions of the Reg. Guide should be considered as guidance in the evaluation of the offsite power systems under SRP Section 8.2 (which may involve switchyards, substations, and in plant switchgear utilizing control power supplied by large lead storage batteries) . The Reg. Guide is specified as guidance only for SRP Section 8.3.2 in Table 8-1.

Potential Impact Number: 18809 **SRP Section Number:** 8.1

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

APPENDIX II
Potential Impacts

Summary:

Provides guidance for complying with 10CFR50.63, Loss of All Alternating Current Power and modifies provisions of NUMARC-8700. Also establishes target reliability levels for electric power systems, and guidance for power restoration emergency procedures. Covers such topics 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 must be demonstrated.The Reg. Guide is not incorporated in Table 8-1.

Potential Impact Number: 18811 **SRP Section Number:** 8.1

Document: RG 1.47 **Revision:** 0
Title: BYPASSED AND INOPERABLE STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS

Summary:

The Reg. Guide provides design guidance for the design of inoperable and bypassed status indications for safety systems and equipment (automatic, system level indication is preferred). The guide describes an acceptable method of compliance with 10 CFR 50 Appendix B, Criterion XIV and IEEE Standard 279-1971 with regard to indicating the inoperable status of a portion of the protection system (as defined in IEEE Std 279-1971), 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. The Reg. Guide is specified as guidance for SRP Sections 8.2, 8.3.1, and 8.3.2 in Table 8-1.

Potential Impact Number: 18814 **SRP Section Number:** 8.1

Document: RG 1.75 **Revision:** 2
Title: PHYSICAL INDEPENDENCE OF ELECTRIC SYSTEMS

Summary:

This guide describes a method acceptable to the NRC staff of complying with IEEE Std 279- 1971 and Criteria 3, 17, and 21 of Appendix A to 10 CFR Part 50 with respect to

the physical independence of the circuits and electric equipment comprising or associated with the Class 1E power system, 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. Also provided is guidance for inclusion of separation criteria information in Safety Analysis Reports and identification practices for redundant divisional circuits, associated circuits, equipment, etc.This guide applies to all types of nuclear power plants. The Reg. Guide modifies the provisions of IEEE Standard 384-1974. The Reg. Guide is specified as guidance for SRP Sections 8.3.1 and 8.3.2 in Table 8-1.

Potential Impact Number: 18815 **SRP Section Number:** 8.1

Document: RG 1.81 **Revision:** 1
Title: SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTI-UNIT NUCLEAR POWER PLANTS

Summary:

The Reg. Guide provides guidance for designs which share components, systems, etc. of electric power systems between units of a multi unit station pursuant to conformance with GDC 5. The guide focuses upon design aspects of onsite power systems but also contains implicit requirements for offsite power systems. With respect to the sharing of d.c. systems, this guide applies to all multi-unit plants. With respect to the sharing of a.c. systems, this guide applies to the multi-unit plants whose a.c. power is supplied by diesel generators. The basic guidance provided is that shared power systems are not considered acceptable in applications docketed after June 1, 1973. Plants built prior to this regulatory position must show adequate capacity for accident mitigation in one unit coincident with orderly shutdown of remaining units. The Reg. Guide is specified as guidance for SRP Sections 8.2, 8.3.1 and 8.3.2 in Table 8-1.

Potential Impact Number: 18829 **SRP Section Number:** 8.1

Document: RG 1.108 **Revision:** 1
Title: PERIODIC TESTING OF DIESEL GENERATOR UNITS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

APPENDIX II
Potential Impacts

Summary:

Provides guidance for the periodic testing of diesel generators used as onsite standby emergency power sources for nuclear power plants. Provides criteria for valid tests and failed tests and recommended test procedures for surveillance testing. Also provides guidance for the design of testability features, test mode emergency override features, and protective trip emergency bypass features. The Reg. Guide is specified as guidance for SRP Section 8.3.1 in Table 8-1.

Potential Impact Number: 18832 **SRP Section Number:** 8.1

Document: RG 1.153 **Revision:** 0
Title: CRITERIA FOR POWER, INSTRUMENTATION, AND CONTROL

Summary:

Provides guidance for the design, reliability, qualification, and testability of power, instrumentation, and control portions of safety systems for nuclear power plants. Endorses and modifies IEEE Standard 603-1980. Also endorses portions of RG 1.97 and IEEE Standard 279-1971 (as taking precedence over certain provisions of IEEE Standard 603-1980). The Reg. Guide indicates that some provisions of IEEE Standard 497-1977 are at variance with current regulatory practice and that in such cases, guidance contained in Reg. Guide 1.97 should be used. The Reg. Guide also indicates that several standards which are referenced in Section 3 of IEEE Standard 603-1980 have not been evaluated by the staff, contain useful information, but must be used consistent with current regulations. The Reg. Guide is not specified as guidance for SRP Sections 8.2, 8.3.1, and/or 8.3.2 in Table 8-1.

Potential Impact Number: 18841 **SRP Section Number:** 8.1

Document: RG 1.9
Title: SELECTION, DESIGN, AND QUALIFICATION OF DIESEL-GENERATOR UNITS USED AS STANDBY (ONSITE) ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

Summary:

APPENDIX II
Potential Impacts

Provides guidance for the selection, design, and qualification of diesel generator units used as standby (onsite) electric power systems at nuclear power plants and extensively modifies the provisions of IEEE Standard 387-1977 (sometimes by applying other standards e.g. IEEE Standard 279-1971, IEEE Standard 308-1974, IEEE Standard 344-1975, and/or IEEE Standard 323-1974). The Reg. Guide is specified as guidance for SRP Section 8.3.1 in Table 8-1.

Potential Impact Number: 18854 **SRP Section Number:** 3.2.2

Document: RG 1.26 **Revision:** 3
Title: QUALITY GROUP CLASSIFICATIONS AND STANDARDS FOR WATER-, STEAM-, AND RADIOACTIVE-WASTE-CONTAINING COMPONENTS OF NUCLEAR POWER PLANTS

Summary:

Regulatory Guide 1.26 is in the acceptance criteria of this SRP section without qualification. The Review Procedures section lists systems that are only mentioned in Regulatory Guide 1.26 as systems not covered (such as Emergency Diesel Fuel Oil and Compressed Air). Appendices A and B of SRP section 3.2.2 augment Regulatory Guide 1.26. Appendices C and D of SRP section 3.2.2 are referenced in the Review Procedures Section to augment Regulatory Guide 1.26, but the information is not complete. Appendices C and D may be the best location to identify all staff exceptions and additions to Regulatory Guide 1.26.

Potential Impact Number: 18855 **SRP Section Number:** 2.3.2

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 10CFR52.17 (a)(1)

Summary:

Applications for Early Site Permits shall contain the information required by the first three sentences of 10 CFR 50.34 (a) (1). The application should describe the meteorological characteristics of the site.

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Potential Impact Number: 18856 **SRP Section Number:** 2.3.2

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 10CFR52..47 (a) (iii)

Summary:

Applications for Standard Design Certification must contain the site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters.

Potential Impact Number: 18857 **SRP Section Number:** 2.3.2

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 10CFR52.79 (a) (1)

Summary:

If an application for a Combined License references an Early Site Permit, the application need not contain information or analysis submitted in connection with the permit. The application must contain information sufficient to demonstrate that the design falls within the parameters specified in the Early Site Permit and to resolve any other environmental issue not considered in any previous proceeding on the site or the design.

Potential Impact Number: 18858 **SRP Section Number:** 2.3.2

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS

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Location: 10CFR52 Appendix M, 2

Summary:

Applications filed for a manufacturing license (authorization to manufacture but not construct or install at a specific location) shall contain the technical information required by Section 50.34(a) (1) except that the preliminary safety analysis report shall be designated as a 'design report' and any required information or analyses relating to site matters shall be predicated on postulated site parameters which must be specified in the application.

Potential Impact Number: 18859 **SRP Section Number:** 2.3.2

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 10CFR52 Appendix O, 3

Summary:

Submittals for a review of a standard design shall include the site parameters postulated for the design, and an analysis and evaluation of the design in terms of such postulated site parameters.

Potential Impact Number: 18861 **SRP Section Number:** 9.3.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: GDC 4

Summary:

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

APPENDIX II
Potential Impacts

Potential Impact Number: 18862 **SRP Section Number:** 9.3.1

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10 CFR Part 50 for the event of loss-of-offsite power, the following positions shall be implemented:

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

(1) Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

(3) Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

Clarification

(1) Although the primary concern resulting from lessons learned from the accident at TMI is that the PORV block valves must be closable, the design should retain, to the extent practical, the capability to also open these valves.

(3) Any changeover of the PORV and block-valve motive and control power from the normal offsite power to the emergency onsite power is to be accomplished manually in the control room.

(4) For those designs in which instrument air is needed for operation, the electrical power supply should be required to have the capability to be manually connected to the emergency power sources.

This NUREG-0737 potential impact was found through a direct search of 0737, and not through a cited reference in the MBD data base. NUREG-0737 is located on the MDB data base.

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Potential Impacts

Potential Impact Number: 18863 **SRP Section Number:** 3.2.2

Document: NUREG CR-5186
Title: VALUE/IMPACT ANALYSIS OF GENERIC ISSUE 94, 'ADDITIONAL LOW TEMPERATURE OVERPRESSURE PROTECTION FOR LIGHT WATER REACTORS'

Summary:

An alternative would require the overpressure mitigation system to be safety grade. Referenced in Generic Letter 90-06. Microfiche No. 48410.

Potential Impact Number: 18864 **SRP Section Number:** 3.2.2

Document: NUREG CR-4692
Title: OPERATING EXPERIENCE REVIEW OF FAILURES OF POWER OPERATED RELIEF VALVES AND BLOCK VALVES IN NUCLEAR POWER PLANTS

Summary:

Upgrading PORV's, operators or control components to safety-grade status could each effect a reduction in PORV failures. Block valve reliability could best be enhanced by upgrading to safety-grade status, where more rigorous testing, diagnostics and maintenance are required. Referenced in Generic Letter 90-06. Microfiche No. 43588.

Potential Impact Number: 18877 **SRP Section Number:** 2.3.1

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 10 CFR 52.47 (a)(iii)

Summary:

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Potential Impacts

Applications for Standard Design Certification must contain the site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters.

Potential Impact Number: 18878 **SRP Section Number:** 2.3.1

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 10 CFR 52.79 (a)(1)

Summary:

If an application for a Combined License references an Early Site Permit, the application need not contain information or analysis submitted in connection with the permit. The application must contain information sufficient to demonstrate that the design falls within the parameters specified in the Early Site Permit and to resolve any other environmental issue not considered in any previous proceeding on the site or the design.

Potential Impact Number: 18879 **SRP Section Number:** 2.3.1

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 10 CFR 52 Appendix M, paragraph 2

Summary:

Applications filed for a manufacturing license (authorization to manufacture but not construct or install at a specific location) shall contain the technical information required by Section 50.34(a) (1) except that the preliminary safety analysis report shall be designated as a 'design report' and any required information or analyses relating to site matters shall be predicated on postulated site parameters which must be specified in the application.

APPENDIX II
Potential Impacts

Potential Impact Number: 18880 **SRP Section Number:** 2.3.1

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 10 CFR 52 Appendix O paragraph 3

Summary:

Submittals for a review of a standard design shall include the site parameters postulated for the design, and an analysis and evaluation of the design in terms of such postulated site parameters.

Potential Impact Number: 18945 **SRP Section Number:** 3.2.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Regulatory Guide 1.26 is the principal document used by the staff in its review of this subject. However the EPRI Evolutionary Requirements documents proposes the use of ANSI/ANS 51.1 and 52.1 as an alternative way of complying with Regulatory Guide 1.26. The staff has not completely endorsed these two standards. The SSARs for CE's System 80+ and GE's ABWR also reference these standards. The staff concludes that it will complete its review of these two standard plant SSARs without these endorsements. Therefore the staff's endorsement of these two standards is not necessary for the Evolutionary Requirements Documents and any future evolutionary plant SSAR will be reviewed in a manner similar to the reviews being conducted for the System 80+ and the ABWR. Therefore, the staff will evaluate this matter during its review of an individual application for FDA/DC.

Potential Impact Number: 19059 **SRP Section Number:** 9.3.1

Document: FINAL SER EPRI CH 9
Title: EPRI URD-EPD: SITE SUPPORT SYSTEMS

APPENDIX II
Potential Impacts

Summary:

ALWRs referencing the Requirements Document are required to have a periodic instrument air quality sampling program, and are required to have design provisions (sampling lines and valves) for obtaining air samples from air operated safety-related plant equipment. The current SRP 9.3.1 contains no such requirement.

Potential Impact Number: 19064 **SRP Section Number:** 9.3.1

Document: FINAL SER EPRI CH 9
Title: EPRI URD-EPD: SITE SUPPORT SYSTEMS

Summary:

In addition, EPRI requires plant designers to evaluate the instrument air system to ensure that check valves isolate in the event of a slow pressure loss in the instrument air system and to provide for sampling capability to ensure continued operation of the check valves and accumulators. This is currently not required by the SRP.

Potential Impact Number: 19066 **SRP Section Number:** 9.3.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 cfr50.34, f.1.x

Summary:

Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWR's only). (II.K.3.28)

Potential Impact Number: 19067 **SRP Section Number:** 9.3.1

APPENDIX II
Potential Impacts

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 cfr50.34, f.2.xx

Summary:

Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources. (Applicable to PWR's only). (II.G.1)

Potential Impact Number: 19081 **SRP Section Number:** 9.3.1

Document: C&S: ISA S7.3
Title: QUALITY STANDARD FOR INSTRUMENT AIR

Summary:

ISA-S7.3-1981 Quality Standard for Instrument Air. Referenced in SRP Section 9.3.1. The code comparison performed in accordance with the IPD determined difference between the cited and current versions of the code and recommends the following: 1). Update the code citation to the current version, (ISA S7.3-1981). 2) Delete reference to ANSI MC 11.1-1976. 3) Delete wording in SRP 9.3.1 Subsection III.2.1, 2.2, 2.3, and 2.4 and replace it with a reference to ISA S7.3-1981.

Potential Impact Number: 19082 **SRP Section Number:** 3.2.2

Document: C&S: ASME III
Title: CLASS 1 AND 2 SYSTEMS

Summary:

ASME BPVC, Section III is identified in 10 CRF 55a as the code applicable to Quality Group A, B, and C components.

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Potential Impacts

Potential Impact Number: 19083 **SRP Section Number:** 3.2.2

Document: C&S: ASME III
Title: CLASS 1 AND 2 SYSTEMS

Summary:

ASME section III is referenced in 10 CFR 50, Appendix G for fracture toughness requirements for the reactor coolant pressure boundary.

Potential Impact Number: 19096 **SRP Section Number:** 3.2.2

Document: C&S: ASME VIII
Title: CLASS 3

Summary:

ASME BPVC, Section VIII -1980 is referenced in SRP Section 3.2.2 as the code applicable to Quality Group D pressure vessels.

Potential Impact Number: 19328 **SRP Section Number:** 15.6.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.46

Summary:

Specifies limits on peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, etc.

Potential Impact Number: 19494 **SRP Section Number:** 8.2

APPENDIX II
Potential Impacts

Document: BTP ICSB 11 (PSB) **Revision:** 2 , 7/1/81
Title: STABILITY OF OFFSITE POWER SYSTEMS

Summary:

BTP ICSB 11 (PSB) -- Stability of Offsite Power Systems. The Branch Technical Position is specified as guidance for SRP Chapter 8 Section 8.2 in Section 8.2 (as BTP ICSB II (PSB)) and in Section 8.1 Table 8-1. Per IPD Section 2.1, this impact is written consistent with the trend to eliminate Branch Technical Positions from the SRP and explicitly incorporate their relevant staff positions in affected SRP Sections. When incorporated as stated, the Branch Technical Position will be deleted from SRP Chapter 8 Appendix 8-A and all citations of the position by number will be deleted.

Potential Impact Number: 19708 **SRP Section Number:** 2.5.2

Document: RG 1.132 **Revision:** 1
Title: SITE INVESTIGATIONS FOR FOUNDATIONS OF NUCLEAR POWER PLANTS

Summary:

The Reg. Guide describes programs of site investigations that meets the needs for evaluating the safety of the site from the standpoint of the performance of foundations and earthworks under most anticipated loading conditions, including earthquakes. It also describes site investigations required to evaluate geotechnical parameters needed for engineering analysis and design. The site investigations discussed in this guide are applicable to both land and offshore sites. Detailed geologic fault investigations required under Appendix A to 10 CFR Part 100, are not discussed nor are hydrologic investigations, except for groundwater measurements. This guide provides general guidance and recommendations for developing site-specific investigation programs as well as specific guidance for conducting subsurface investigations, the spacing and depth of borings, and sampling. Because the details of the actual site investigations program will be highly site dependent, the procedures described herein should be used only as guidance and should be tempered with professional judgment. Alternative and special investigative procedures that have been derived in a professional manner will be considered equally applicable for conducting foundation investigations. Appendix A provides definitions for some of special terms used. Appendix B tabulates methods of conducting subsurface investigations, and Appendix C gives guidelines for the spacing and depth of borings for safety-related structures in regions of favorable or uniform conditions. The Reg. Guide is cited as acceptance criteria for SRP Section 2.5.2.

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Potential Impact Number: 19709 **SRP Section Number:** 2.5.4

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 10 CFR 52.17.(a).(1).(vi)

Summary:

The application should describe the seismic, meteorological, hydrologic, and geologic characteristics of the proposed site (see Appendix A to 10 CFR Part 100).

Potential Impact Number: 19718 **SRP Section Number:** 2.5.4

Document: NRC POLICY STATEMENT 52 FR 34884
Title: NUCLEAR POWER PLANT STANDARDIZATION

Summary:

Features of the design which can only be determined when a specific site is chosen generally are not included in the design approval or certification. Rather the designer defines a set of site enveloping parameters (seismic events, rainfall, flood, etc.) which are used in the design of the plant. These parameters usually are selected to envelop a large portion of the potential sites in the U.S. Once the design is certified by the Commission, conformance of actual sites with the established site envelope most be demonstrated by the applicant and verified by the staff at the time an actual plant application is reviewed.

Potential Impact Number: 19735 **SRP Section Number:** 13.5.1.1

Document: CODE OF FED. REGS 10CFR26
Title: FITNESS-FOR-DUTY PROGRAM
Location: Appendix A, Subpart B, 2.2

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Potential Impacts

Summary:

The licensee testing facilities and HHS-certified laboratories described in this part shall develop and maintain clear and well documented procedures for collection, shipment, and accession of urine and blood specimens under this part.

Potential Impact Number: 19736 **SRP Section Number:** 13.5.1.1

Document: CODE OF FED. REGS 10CFR26
Title: FITNESS-FOR-DUTY PROGRAM
Location: Appendix A, Subpart B, 2.4

Summary:

Specimen collection procedure requirements are give in this section.

Potential Impact Number: 19737 **SRP Section Number:** 13.5.1.1

Document: CODE OF FED. REGS 10CFR26
Title: FITNESS-FOR-DUTY PROGRAM
Location: Appendix A, Subpart B, 2.5.(a).(5)

Summary:

This individual shall be responsible for the laboratory's having a procedure manual which is complete, up-to-date, available for personnel performing tests, and followed by those personnel. The procedure manual shall be reviewed, signed, and dated by this responsible individual whenever procedures are first placed into use or changed or when a new individual assumes responsibility for management of the laboratory. Copies of all procedures and dates on which they are in effect shall be maintained.

Potential Impact Number: 19738 **SRP Section Number:** 13.5.1.1

Document: CODE OF FED. REGS 10CFR26

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Potential Impacts

Title: FITNESS-FOR-DUTY PROGRAM
Location: Appendix A, Subpart B, 2.7.(o).(3)

Summary:

There shall be written procedures for instrument set-up and normal operation, a schedule for checking critical operating characteristics for all instruments, tolerance limits for acceptable function checks, and instructions for major troubleshooting and repair. Records shall be available on preventive maintenance. There shall be written procedures for the actions to be taken when systems are out of acceptable limits or errors are detected. There shall be documentation that these procedures are followed and that all necessary corrective actions are taken.

Potential Impact Number: 19773 **SRP Section Number:** 5.4.2.1

Document: NRC GENERIC LETTER 85-02
Title: STAFF RECOMMENDED ACTIONS STEMMING FROM NRC INTEGRATED PROGRAM FOR THE RESOLUTION OF UNRESOLVED SAFETY ISSUES REGARDING STEAM GENERATOR TUBE INTEGRITY

Summary:

The letter describes an integrated program for assuring steam generator tube integrity and mitigating the consequences of a steam generator tube rupture. The purpose of the letter was to obtain information on licensee programs and to encourage licensees of PWRs to upgrade their programs. An item of this program is secondary chemistry control. The letter cites EPRI-NP-2704 as a source of information with respect to secondary chemistry control programs.

Potential Impact Number: 19791 **SRP Section Number:** 5.4.2.1

Document: NRC BULLETIN 89-01 Sup 2
Title: FAILURE OF WESTINGHOUSE STEAM GENERATOR TUBE MECHANICAL PLUGS

Summary:

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Identifies steam generator tube plugs made from Inconel 600 that may fail due to PWSCC.

Potential Impact Number: 19792 **SRP Section Number:** 5.4.2.1

Document: NRC BULLETIN 89-01 Sup 1
Title: FAILURE OF WESTINGHOUSE STEAM GENERATOR TUBE MECHANICAL PLUGS

Summary:

Identifies steam generator tube plugs made from Inconel 600 that may fail due to PWSCC.

Potential Impact Number: 19798 **SRP Section Number:** 5.2.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix G, Introduction

Summary:

This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

Potential Impact Number: 19809 **SRP Section Number:** 17.1

Document: RG 1.8 **Revision:** 2 , 4/1/87
Title: QUALIFICATION AND TRAINING OF PERSONNEL FOR NUCLEAR POWER PLANTS

APPENDIX II
Potential Impacts

Summary:

This regulatory guide describes a method acceptable to the NRC staff for complying with those portions of the Commission's regulations with regard to the training and qualifications of nuclear power plant personnel. Personnel of test, training, research, and mobile reactors are not covered by this regulatory guide. For the positions listed in ANSI/ANS-3.1-1981 Selection, Qualification and Training of Personnel for Nuclear Power Plants, as shift supervisor, senior operator, licensed operator, and shift technical advisor, the requirements contained in the standard provide a approach acceptable to the NRC staff for complying with the qualifications and training requirements of 10 CFR Parts 50 and 55 subject to the guidance regarding the Statement on Engineering Expertise on Shift and the clarifications, additions, and exceptions in the regulatory positions C.1.a through C.1.j. For radiation protection supervisory personnel, Section 4.4.4 of the standard contains an approach acceptable for the position of radiation protection manager (RPM) subject to the clarification contained in regulatory position C.1.k. For positions listed in the standard other than those under Regulatory Position 1 (modifying the provisions of ANSI/ANS-3.1-1981), the requirements contained in ANSI N18.1-1971, Selection and Training of Nuclear Power Plant Personnel, provide an approach acceptable to the NRC staff for complying with the qualifications and training requirements of 10 CFR Parts 50 and 55.

Potential Impact Number: 19812 **SRP Section Number:** 17.1

Document: RG 1.152 **Revision:** 0
Title: CRITERIA FOR PROGRAMMABLE DIGITAL COMPUTER SYSTEM SOFTWARE IN SAFETY-RELATED SYSTEMS OF NUCLEAR POWER PLANTS

Summary:

This guide 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. This method is applicable to designing software, verifying software, implementing software, and validating computer systems. Structures, systems, and components are safety related if they are relied upon to remain functional during and following design basis events to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 CFR Part 100 guidelines. 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. Although ANSI/IEEE-ANS-7-4.3.2-1982 references IEEE Std 603-1980, ANSI/ASME NQA-1-1979, and IEEE Std 467-1980, these referenced standards are not endorsed by this regulatory guide. They do, however, contain valuable information. If the referenced standards are used, they should be used in a manner consistent with current regulations, which include but are not limited to IEEE Std 279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations, and Appendix B of 10 CFR Part 50.

Potential Impact Number: 19813 **SRP Section Number:** 17.1

Document: RG 1.36 **Revision:** 0
Title: NONMETALLIC THERMAL INSULATION FOR AUSTENITIC STAINLESS STEEL

Summary:

General Design Criterion 1, Quality Standards and Records, of Appendix A to 10 CFR Part 50, General Design Criteria for Nuclear Power Plants, 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 function to be performed. In addition, General Design Criteria 14 and 31 require assurance that the reactor coolant pressure boundary will have an, extremely low probability of gross rupture or rapidly propagating fracture. Stress-corrosion cracking, which is promoted by certain contaminants, is one mechanism whereby such failures may be postulated. This guide describes an acceptable method for implementing these criteria 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.

Potential Impact Number: 19814 **SRP Section Number:** 17.1

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Document: RG 1.54 **Revision:** 0
Title: QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATERCOOLED NUCLEAR POWER PLANTS

Summary:

This guide 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 regulatory positions.

Potential Impact Number: 19821 **SRP Section Number:** 13.5.1.1

Document: NRC GENERIC LETTER 89-23
Title: NRC STAFF RESPONSES TO QUESTIONS PERTAINING TO IMPLEMENTATION OF 10 CFR PART 26

Summary:

Enclosed is a copy of NUREG-1385, 'Fitness-for-Duty in the Nuclear Power Industry: Responses to Implementation Questions.' The NUREG responds to questions concerning implementation of the Fitness-for-Duty Rule (10 CFR Part 26), and may be useful in developing programs which implement the rule. The attached NUREG does not constitute any new requirements beyond 10 CFR Part 26. Furthermore, transmittal of the NUREG does not impose any new regulatory requirements.

Potential Impact Number: 19831 **SRP Section Number:** 17.1

Document: NRC BULLETIN 82-02
Title: DEGRADATION OF THREADED FASTENERS IN THE REACTOR COOLANT PRESSURE BOUNDARY OF PWR PLANTS

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Summary:

The bulletin alerts PWR licensees to the potential for primary coolant pressure boundary threaded fastener degradation/failure due to stress corrosion cracking initiated by primary coolant leakage. All PWR licensees and applicants are required to take the following action: 1. Where procedures do not exist, develop and implement maintenance procedures for threaded fastener practices. These procedures should include, but not limited to the following: (1) maintenance crew training of proper bolting/stud practices, tools application, specifications and requirements, (2) detensioning and retensioning practices (torque iteration), specified tolerances, and other controls for disassembly and reassembly of component closure/seal connections, (3) gasket installation and controls, and (4) retensioning methods and other measures to eliminate reactor coolant leakage during operations. Quality assurance measures should also be established for proper selection, procurement, and application of fastener lubricants and injection sealant compounds to minimize fastener susceptibility to SCC environments.

Potential Impact Number: 19833 **SRP Section Number:** 17.1

Document: NRC GENERIC LETTER 83-11
Title: LICENSEE QUALIFICATION FOR PERFORMING SAFETY ANALYSES IN SUPPORT OF LICENSING ACTIONS

Summary:

The letter informs licensees of the current NRR practice regarding licensee qualification for performing safety analyses in support of licensing actions. NRR has been working with the Vendor Inspection Branch in Region IV to inspect the acceptability of the QA procedures used for code development, verification, use, and maintenance for all licensees and vendors involved with safety analysis codes presently under staff review. While this aspect of review focuses primarily on the competence of the licensee and vendors regarding quality assurance practices, the technical competence of the licensees and vendors with respect to their ability to set up an input deck, execute a code, and properly interpret the results must also be assured. NRR obtains this assurance by reviewing the code verification information submitted by the licensee or vendor. The information we look for includes comparisons performed by the user of the code results to experimental data, plant operational data, or other benchmarked analyses. NRR is concerned however, that some licensees planning to perform their own safety analyses may not intend to demonstrate their ability to use the code by

performing their own code verification. Rather, they plan to rely on the code verification work previously performed by the code developer or others. NRR does not consider this acceptable and each licensee or vendor who intends to use a safety analysis computer code to support licensing actions should demonstrate their proficiency in using the code by submitting code verification performed by them, not others.

Potential Impact Number: 19837 **SRP Section Number:** 5.4.7

Document: NRC BULLETIN 88-08 Sup 3
Title: THERMAL STRESSES IN PIPING CONNECTED TO REACTOR COOLANT SYSTEMS

Summary:

The purpose of this supplement is to (1) provide information to addressees about an event at a foreign reactor relating to thermal stratifications of unisolable piping connected to the reactor coolant system (RCS) similar to the December 9, 1987, Farley 2 event, (2) emphasize to the addressees the need for sufficient review of their RCSs to identify any connected, unisolable piping that could be subjected to unacceptable thermal stratification, and (3) again emphasize the importance of addressees taking action, where such piping is identified, to ensure that the piping will not be subjected to unacceptable thermal stresses. (The unisolable piping was part of the RHR).

Potential Impact Number: 19841 **SRP Section Number:** 17.2

Document: NRC BULLETIN 82-02
Title: DEGRADATION OF THREADED FASTENERS IN THE REACTOR COOLANT PRESSURE BOUNDARY OF PWR PLANTS

Summary:

The bulletin alerts PWR licensees to the potential for primary coolant pressure boundary threaded fastener degradation/failure due to stress corrosion cracking initiated by primary coolant leakage. All PWR licensees and applicants are required to take the following action: 1. Where procedures do not exist, develop and implement maintenance procedures for threaded fastener practices. These procedures should include, but not limited to the following: (1) maintenance crew training of proper

bolting/stud practices, tools application, specifications and requirements, (2) detensioning and retensioning practices (torque iteration), specified tolerances, and other controls for disassembly and reassembly of component closure/seal connections, (3) gasket installation and controls, and (4) retensioning methods and other measures to eliminate reactor coolant leakage during operations. Quality assurance measures should also be established for proper selection, procurement, and application of fastener lubricants and injection sealant compounds to minimize fastener susceptibility to SCC environments.

Potential Impact Number: 19842 **SRP Section Number:** 17.2

Document: CODE OF FED. REGS 10CFR21
Title: REPORTING OF DEFECTS AND NONCOMPLIANCE
Locations: 21.1, 21.1, 21.21, 21.31, 21.41, 21.51, 21.61

Summary:

The regulations in this part establish procedures and requirements for implementation of section 206 of the Energy Reorganization Act of 1974. That section requires any individual director or responsible officer of a firm constructing, owning, operating or supplying the components of any facility or activity which is licensed or otherwise regulated pursuant to the Atomic Energy Act of 1954, as amended, or the Energy Reorganization Act of 1974, who obtains information reasonably indicating: (a) That the facility, activity or basic component supplied to such facility or activity fails to comply with the Atomic Energy Act of 1954, as amended, or any applicable rule, regulation, order, or license of the Commission relating to substantial safety hazards or (b) that the facility, activity, or basic component supplied to such facility or activity contains defects, which could create a substantial safety hazard, to immediately notify the Commission of such failure to comply or such defect, unless he has actual knowledge that the Commission has been adequately informed of such defect or failure to comply. The regulations in this part apply, except as specifically provided otherwise in Parts 31, 34, 35, 39, 40, 60, 61, 70, or 72 of this chapter, to each individual, partnership, corporation, or other entity licensed pursuant to the regulations in this chapter to possess, use, and/or transfer within the United States source material, byproduct material, special nuclear material, and/or spent fuel and high-level radioactive waste, or to construct, manufacture, possess, own, operate and/or transfer within the United States, any production or utilization facility, or independent spent fuel storage installation (ISFSI) or monitored retrievable storage installation (MRS), and to each director (see Section 21.3(f)) and responsible officer (see Section 21.3(j)) of such a licensee. The regulations

in this part apply also to each individual, corporation, partnership or other entity doing business within the United States, and each director and responsible officer of such organization that constructs (see Section 21.3(c)) a production or utilization facility licensed for manufacture, construction or operation (see Section 21.3(h)) pursuant to Part 50 of this chapter, an independent spent fuel storage installation (ISFSI) for the storage of spent fuel licensed pursuant to Part 72 of this chapter or a monitored retrievable storage installation (MRS) for the storage of spent fuel or high-level radioactive waste licensed pursuant to Part 72 of this chapter, or supplies (see Section 21.3(1)) basic components (see Section 21.3(a)) for a facility or activity licensed, other than for export, under Parts 30, 39, 40, 50, 60, 61, 70, 71, or 72 of this chapter. Nothing in these regulations should be deemed to preclude either an individual or a manufacturer/supplier of a commercial grade item (see Section 21.3(a-1)) not subject to the regulations in this part from reporting to the Commission a known or suspected defect or failure to comply and, as authorized by law, the identity of anyone so reporting will be withheld from disclosure. The regulation requires (among other things) prompt NRC notification of conditions described above and defined therein and specification in applicable in procurement documents that the provisions of 10 CFR 21 apply. The regulation is normally addressed in licensee/ applicant quality assurance programs and is thus cited as acceptance criteria for the review of quality assurance program descriptions under SRP Section 17.3.

Potential Impact Number: 19844 **SRP Section Number:** 6.2.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.34(f).(2).(xv)

Summary:

Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions.

Potential Impact Number: 19868 **SRP Section Number:** 17.2

Document: CODE OF FED. REGS 10CFR50

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Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.54 (a)(1)

Summary:

Whether stated therein or not, the following shall be deemed conditions in every license issued: (a)(1) Each nuclear power plant or fuel reprocessing plant licensee subject to the quality assurance criteria in Appendix B of this part shall implement, pursuant to Section 50.34(b)(6)(ii) of this part, the quality assurance program described or referenced in the Safety Analysis Report, including changes to that report. (2) Each licensee described in paragraph (a)(1) of this section shall, by June 10, 1983, submit to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this chapter the current description of the quality assurance program it is implementing for inclusion in the Safety Analysis Report, unless there are no changes to the description previously accepted by NRC. This submittal must identify changes made to the quality assurance program description since the description was submitted to NRC. (Should a licensee need additional time beyond June 10, 1983 to submit its current quality assurance program description to NRC, it shall notify the appropriate NRC Regional Office in writing, explain why additional time is needed, and provide a schedule for NRC approval showing when its current quality assurance program description will be submitted.) (3) After March 11, 1983, each licensee described in paragraph (a)(1) of this section may make a change to a previously accepted quality assurance program description included or referenced in the Safety Analysis Report, provided the change does not reduce the commitments in the program description previously accepted by the NRC. Changes to the quality assurance program description that do not reduce the commitments must be submitted to the NRC at least annually in accordance with the requirements of Section 50.71. Changes to the quality assurance program description that do reduce the commitments must be submitted to NRC and receive NRC approval prior to implementation, as follows: (i) Changes made to the Safety Analysis Report must be submitted, as specified in Section 50.4. Changes made to NRC-accepted quality assurance topical report descriptions must be submitted, as specified in Section 50.4. (ii) The submittal of a change to the Safety Analysis Report quality assurance program description must include all pages affected by that change and must be accompanied by a forwarding letter identifying the change, the reason for the change, and the basis for concluding that the revised program incorporating the change continues to satisfy the criteria of Appendix B of this part and the Safety Analysis Report quality assurance program description commitments previously accepted by the NRC (the letter need not provide the basis for changes that correct spelling, punctuation, or editorial items). (iii) A copy of the forwarding letter identifying the changes must be maintained as a facility record for three years. (iv) Changes to the quality assurance program description included or referenced in the Safety Analysis

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Report shall be regarded as accepted by the Commission upon receipt of a letter to this effect from the appropriate reviewing office of the Commission or 60 days after submittal to the Commission, whichever occurs first.

Potential Impact Number: 19931 **SRP Section Number:** 17.2

Document: RG 7.10 **Revision:** 1
Title: ESTABLISHING QUALITY ASSURANCE PROGRAMS FOR PACKAGING

Summary:

This regulatory guide 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. Terms used in this guide are consistent with terms used in 10 CFR Part 71 and ANSI/ASME NQA-1-1979, Quality Assurance Requirements for Nuclear Power Plants. The essential elements of a quality assurance program acceptable to the NRC staff for complying with the quality assurance requirements of Subpart H of 10CFR Part 71 are contained in Annex 1 of this guide for activities related to design, fabrication, assembly, and testing of packaging and in Annex 2 for activities related to procurement, use, maintenance, and repair of completed packaging. Annex 3 contains simplified guidance specifically applicable to users of radiographic exposure devices for activities related to procurement, use, maintenance, and repair of packages designed to transport such devices. Appendix A provides guidance for development of programs using a graded approach.

Potential Impact Number: 19940 **SRP Section Number:** 13.5.2.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50, Appendix B, section V

Summary:

Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures,

or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Potential Impact Number: 19955 **SRP Section Number:** 5.4.7

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

The capability of all systems and components necessary to provide core cooling and decay heat removal following a station blackout should be determined, including station battery capacity, condensate storage tank capacity, compressed air capacity, and instrumentation and control requirements.

Potential Impact Number: 19960 **SRP Section Number:** 17.3

Document: NRC BULLETIN 82-02
Title: DEGRADATION OF THREADED FASTENERS IN THE REACTOR COOLANT PRESSURE BOUNDARY OF PWR PLANTS

Summary:

The bulletin alerts PWR licensees to the potential for primary coolant pressure boundary threaded fastener degradation/failure due to stress corrosion cracking initiated by primary coolant leakage. All PWR licensees and applicants are required to take the following action: 1. Where procedures do not exist, develop and implement maintenance procedures for threaded fastener practices. These procedures should include, but not limited to the following: (1) maintenance crew training of proper bolting/stud practices, tools application, specifications and requirements, (2) detensioning and retensioning practices (torque iteration), specified tolerances, and other controls for disassembly and reassembly of component closure/seal connections, (3) gasket installation and controls, and (4) retensioning methods and other measures to eliminate reactor coolant leakage during operations. Quality assurance measures should also be established for proper selection, procurement, and application of fastener lubricants and injection sealant compounds to minimize fastener susceptibility to SCC environments.

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Potential Impact Number: 20001 **SRP Section Number:** 17.3

Document: NRC GENERIC LETTER 82-21
Title: TECHNICAL SPECIFICATIONS FOR FIRE PROTECTION AUDITS

Summary:

Guidance on audits of fire protection program including qualification of auditors and scope of the audits. Includes excerpts from 'Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance' sent to licensees and applicants in 1977. The Recall version of the letter in the ALLNOSRP index is missing the Enclosures thus precluding further identification of potential impacts.

Potential Impact Number: 20016 **SRP Section Number:** 5.4.12

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.44 (c) (3) (iii)

Summary:

(iii) To provide improved operational capability to maintain adequate core cooling following an accident, by the end of the first scheduled outage beginning after July 1, 1982 and of sufficient duration to permit required modifications, each light-water nuclear power reactor shall be provided with high point vents for the reactor coolant system, for the reactor vessel head, and for other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause the loss of function of these systems. (High point vents are not required, however, for the tubes in U-tube steam generators.) The high point vents must be remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents and associated controls, instruments and power sources must conform to the requirements of Appendix A and Appendix B of this part. In particular, the vent system shall be designed to ensure a low probability that (A) the vents will not perform their safety functions and (B) there would be inadvertent or irreversible actuation of a vent. Furthermore, the use of these vents during and following an accident must not aggravate the challenge to the containment or the course of the accident.

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Potential Impact Number: 20020 **SRP Section Number:** 5.4.12

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.49

Summary:

(a) Each holder of or each applicant for a license to operate a nuclear power plant shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section. (b) Electric equipment important to safety covered by this section is: (1) Safety-related electric equipment: This equipment is that relied upon to remain functional during and following design basis events to ensure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the consequences of accidents... (c) Requirements for (1) dynamic and seismic qualification of electric equipment important to safety, (2) protection of electric equipment important to safety against other natural phenomena and external events, and (3) environmental qualification of electric equipment important to safety located in a mild environment are not included within the scope of this section....(d) The applicant or licensee shall prepare a list of electric equipment important to safety covered by this section...(e) The electric equipment qualification program must include and be based on the following: (lists numerous environmental conditions and factors). Regulatory positions contained in NUREG-0737 Item II.B.1 place reactor coolant system high point vent control circuits, power supplies, instrumentation, and associated components in the category of equipment covered by this regulation.

Potential Impact Number: 20031 **SRP Section Number:** 3.8.4

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: V. (b).(4)

Summary:

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 structures and the different location with respect to that of the site.

Potential Impact Number: 20033 **SRP Section Number:** 3.8.4

Document: RG 1.136 **Revision:** 2
Title: MATERIALS, CONSTRUCTION, AND TESTING OF CONCRETE CONTAINMENTS (Articles CC-1000, -2000, and 4000 through 6000 of the Code for Concrete Reactor Vessels and Containments)

Summary:

With the issuance of this revision to Regulatory Guide 1.136, the regulatory guides listed will be withdrawn (including Reg. Guides 1.10, 1.15, and 1.55).

Potential Impact Number: 20040 **SRP Section Number:** 3.8.4

Document: RG 1.10
Title: MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CATEGORY I CONCRETE STRUCTURE (WITHDRAWN - SEE 46 FR 37579)

Summary:

Regulatory Guide 1.10 is referenced in SRP section 3.8.4, but Regulatory Guide 1.136 states Regulatory Guide 1.10 has been withdrawn.

Potential Impact Number: 20041 **SRP Section Number:** 3.8.4

Document: RG 1.15
Title: TESTING OF REINFORCING BARS FOR CATEGORY I CONCRETE STRUCTURES (WITHDRAWN 46 FR 37579, 7/21/81)

Summary:

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Regulatory Guide 1.15 is referenced in SRP section 3.8.4, but Regulatory Guide 1.136 states Regulatory Guide 1.15 has been withdrawn.

Potential Impact Number: 20042 **SRP Section Number:** 3.8.4

Document: RG 1.55
Title: CONCRETE PLACEMENT IN CATEGORY I STRUCTURES (WITHDRAWN 46 FR 37579, 7/21/81)

Summary:

Regulatory Guide 1.55 is referenced in SRP section 3.8.4, but Regulatory Guide 1.136 states Regulatory Guide 1.55 has been withdrawn.

Potential Impact Number: 20049 **SRP Section Number:** 17.3

Document: NRC GENERIC LETTER 86-10
Title: IMPLEMENTATION OF FIRE PROTECTION REQUIREMENTS

Summary:

For fire protection systems the licensee should have and maintain a quality assurance program that provides assurance that the fire protection system will be designed, fabricated, erected, tested, maintained and operated so that they will function as intended. Fire protection systems are not safety-related and are therefore not within the scope of Appendix B to 10 CFR Part 50, unless the licensee has committed to include these systems under the Appendix B program for the plant. NRC guidance for an acceptable quality assurance program for fire protection systems, given in Section C.4 of Branch Technical Position CMEB 9.5-1 Rev. 2 dated July 1981, has generally been used in the review and acceptance of similar guidance was referenced in footnotes 3 and 4 to 10 CFR 50.48. They are contained in BTP APCSB 9.5-1 and Appendix A thereto and in Nuclear Plant Fire Protection Functional Responsibilities, Administrative Control and Quality Assurance dated June 14, 1977.

Potential Impact Number: 20089 **SRP Section Number:** 17.3

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Potential Impacts

Document: NRC GENERIC LETTER 91-05
Title: LICENSEE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS

Summary:

Describes the characteristics of an effective commercial grade procurement and dedication program. This generic letter expresses staff positions regarding certain aspects of licensee commercial- grade procurement and dedication programs which would provide acceptable methods to meet regulatory requirements. The Nuclear Utility Management and Resources Council (NUMARC) Board of Directors recently approved a comprehensive procurement initiative as described in NUMARC 90-13, Nuclear Procurement Program Improvements, which commits licensees to assess their procurement programs and take specific action to enhance or upgrade the program if they are determined to be inadequate. The initiative on the dedication of commercial-grade items, which is part of NUMARC 90-13, was to be implemented by January 1, 1990. The staff position is that the staff will not initiate enforcement action in cases of past programmatic violations that have been adequately corrected. In addition, the staff does not expect licensees to review all past procurements. However, if during current procurement activities, licensees identify shortcomings in the form, fit, or function of specific vendor products, or if failure experience or current information on supplier adequacy indicates that a component may not be suitable for service, corrective actions are required for all such installed and stored items in accordance with Criterion XVI of 10 CFR Part 50, Appendix B. Also in accordance with Criterion XVI, licensees must determine programmatic causes when actual deficiencies in several products from different vendors are identified during current procurement activities and these deficiencies lead to the replacement of installed items as part of the corrective action. In such cases, a further sampling of previously procured commercial-grade items may be warranted. In NRC Generic Letter (GL) 89-02, Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products, the staff described its perspective on good practices in procurement and dedication and provided the NRC's conditional endorsement of an industry standard (EPRI NP-5652) on methods of commercial-grade procurement and dedication. Several items discussed in letter 89-02 are clarified in letter 91-05.

Potential Impact Number: 20203 **SRP Section Number:** 17.3

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

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Potential Impacts

Summary:

Appendices A and B (of Reg. Guide 1.155) provide guidance on quality assurance (QA) activities and specifications respectively for non-safety-related equipment used to meet the requirements of 50.63 and not already covered by existing QA requirements in Appendix B or R of Part 50. Appropriate activities should be implemented from among those listed in these appendices depending on whether the non-safety equipment is being added (new) or is existing. This QA guidance is applicable to non-safety systems and equipment for meeting the requirements of 50.63 of 10 CFR Part 50. The guidance on QA and specifications incorporates a lesser degree of stringency by eliminating requirements for involvement of parties outside the normal line organization. NRC inspections will focus on the implementation and effectiveness of the quality controls described in Appendices A and B. Additionally, the equipment installed to meet the station blackout rule must be implemented such that it does not degrade the existing safety-related systems. This is to be accomplished by making the nonsafety- related equipment as independent as practicable from existing safety-related systems. The non-safety systems identified in Appendix B are acceptable to the NRC staff for responding to a station blackout. A 10 point QA program is specified in Appendix A. The Recall ALLNOSRP index version of the Reg. Guide is missing Appendix B thus precluding analysis of further potential impacts.

Potential Impact Number: 20237 **SRP Section Number:** 17.3

Document: RG 1.97 **Revision:** 3
Title: INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-watercooled nuclear power plant. Table 1 provides a list of quality assurance-related Reg. Guides that should be followed (1.28, 1.30, 1. 38, 1.58, 1.64, 1.88, 1.123, 1.144, 1.146) for Category 1 and some/most Category 2 instrumentation. A note at end of paragraph 5 implies that an in- progress Reg. Guide 1.28 revision will reduce this list to 1.30 and 1.38.

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Potential Impact Number: 20358 **SRP Section Number:** 2.4.3

Document: NRC POLICY STATEMENT 52 FR 34884
Title: NUCLEAR POWER PLANT STANDARDIZATION

Summary:

Features of the design which can only be determined when a specific site is chosen generally are not included in the design approval or certification. Rather the designer defines a set of site enveloping parameters (seismic events, rainfall, flood, etc.) which are used in the design of the plant. These parameters usually are selected to envelope a large portion of the potential sites in the U.S. Once the design is certified by the Commission, conformance of actual sites with the established site envelope most be demonstrated by the applicant and verified by the staff at the time an actual plant application is reviewed.

Potential Impact Number: 20367 **SRP Section Number:** 14.2

Document: RG 1.136 **Revision:** 2
Title: MATERIALS, CONSTRUCTION, AND TESTING OF CONCRETE CONTAINMENTS (Articles CC-1000, -2000, and 4000 through 6000 of the Code for Concrete Reactor Vessels and Containments)

Summary:

This guide describes bases acceptable to the NRC staff for implementing the requirements of the Commission's regulations with regard to the materials, construction, and testing of concrete containments.

The requirements specified in Articles CC-1000,-2000, and -4000 through -6000 of the 'Code for Concrete Reactor Vessels and Containments,' ASME Boiler and Pressure Vessel Code, Section III, Division 2, 1980 Edition, (also known as ACI Standard 359-80) are acceptable to the NRC staff for the materials, construction, and testing of concrete containments of nuclear power plants subject to the exceptions described in the regulatory positions.

The referenced Code incorporates the recommendations of several regulatory guides in an acceptable manner. Hence, with the issuance of this revision to Regulatory Guide 1.136, the following listed regulatory guides will be 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.

Potential Impact Number: 20394 **SRP Section Number:** 2.4.10

Document: RG 1.59 **Revision:** 2
Title: DESIGN BASIS FLOODS NUCLEAR POWER PLANTS

Summary:

Endorses ANSI Standard N170-1976, Standards for Determining Design Basis Flooding at Power Reactor Sites, as an acceptable method for determining probable maximum and seismically induced floods (on streams and coastal areas) with one exception/modification.

Potential Impact Number: 20437 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

GDC 1 requires that nuclear power plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. This requirement is applicable to both pressure-retaining and non-pressure-retaining SSCs that are part of the RCPB and other systems important to safety, when reliance is placed on these systems to (1) prevent or mitigate the consequences of accidents and malfunctions originating within the RCPB, (2) permit shutdown of the reactor and maintain it in a safe-shutdown condition, and (3) retain radioactive material.

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Potential Impact Number: 20438 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff concludes that the QG classifications of all pressure-retaining and non-pressure-retaining SSCs important to safety that are identified in SSAR Table 3.2-1 are in conformance with RG 1.26, and with staff positions on previously licensed boiling-water reactor (BWR) plants and are acceptable. Table 3.2-1, in part, identifies major components in fluid systems (such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves) and in mechanical systems (such as cranes, refueling platforms, and other miscellaneous handling equipment). In addition, SSAR P&IDs identify the classification boundaries or interconnecting piping and valves. All of the above SSCs are constructed in conformance with applicable ASME Code and industry standards. Conformance to RG 1.26, previous staff positions, and applicable ASME Codes and industry standards provides assurance that component quality will be commensurate with the importance of the safety function of these systems. This constitutes the basis for satisfying GDC 1 and is acceptable.

SSAR Table 3.2-1 identifies all ABWR safety-related mechanical and electrical equipment as (1) safety Class 1, 2, or 3, (2) seismic Category I, and (3) designed to the quality assurance requirements of 10 CFR Part 50, Appendix B. As discussed in Sections 3.2.1 and 3.2.2 of this report, the staff concludes that Table 3.2-1 is acceptable. On the basis of these evaluations, the staff concludes that GE meets GDC 1, 30, and 10 CFR Part 50, Appendix B, as they relate to qualifying safety-related mechanical and electrical equipment to appropriate quality standards commensurate with the importance of the safety function to be performed.

Potential Impact Number: 20446 **SRP Section Number:** 3.6.2

Document: NRC GENERIC LETTER 87-11
Title: RELAXATION IN ARBITRARY INTERMEDIATE PIPE RUPTURE REQUIREMENTS

Summary:

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SUBJECT: RELAXATION IN ARBITRARY INTERMEDIATE PIPE RUPTURE REQUIREMENTS (GENERIC LETTER 87-11) The NRC has finalized a revision to Branch Technical Position MEB 3-1 of Standard Review Plan Section 3.6.2 in NUREG-0800 (Enclosure 1).
The revision eliminates all dynamic effects (missile generation, pipe whipping, pipe break reaction forces, jet impingement forces, compartment, subcompartment and cavity pressurizations and decompression waves within the ruptured pipe) and all environmental effects (pressure, temperature, humidity and flooding) resulting from arbitrary intermediate pipe ruptures. This action allows the elimination of pipe whip restraints and jet impingement shields placed to mitigate the effects of arbitrary intermediate pipe ruptures, and other related changes. The licensees' updated FSARs should reflect eliminated hardware associated with arbitrary intermediate pipe ruptures. Arbitrary intermediate pipe ruptures, which previously were specified in B.1.c. (1)(d) of MEB 3-1 for ASME Code Class 1 piping and in B.1.c.(2)(b)(ii) of MEB 3-1 for ASME Code Class 2 and 3 piping, are now no longer mentioned or defined in MEB 3-1. Besides the relaxation in requirements relating to arbitrary intermediate pipe ruptures, the revised Standard Review Plan Section 3.6.2 updates the citations to the ASME stress limits to achieve consistency with current practices, and introduces other minor changes. However, requirements for postulated terminal end pipe ruptures, postulated intermediate pipe ruptures at locations of high stress and high usage factor and for leakage cracks are retained in the revision to MEB 3-1. Frank J. Miraglia, Jr. Associate Director for Projects Office of Nuclear Reactor Regulation Enclosure: Revised MEB 3-1 of SRP 3.6.2 RECALL DataBase BRANCH TECHNICAL POSITION MEB 3-1 POSTULATED RUPTURE LOCATIONS IN FLUID SYSTEM PIPING INSIDE AND OUTSIDE CONTAINMENT A. BACKGROUND This position on pipe rupture postulation is intended to comply with the requirements of General Design Criterion 4, of Appendix A to 10 CFR Part 50 for the design of nuclear power plant structures and components. It is recognized that pipe rupture is a rare event which may only occur under unanticipated conditions, such as those which might be caused by possible design, construction, or operation errors; unanticipated loads or unanticipated corrosive environments. Our observation of actual piping failures has indicated that they generally occur at high stress and fatigue locations, such as at the terminal ends of a piping system at its connection to the nozzles of a component. The rules of this position are intended to utilize the available piping design information by postulating pipe ruptures at locations having relatively higher potential for failure, such that an adequate and practical level of protection may be achieved.

Potential Impact Number: 20478 **SRP Section Number:** 5.2.1.1

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Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.55a

Summary:

Each operating license for a boiling or pressurized water-cooled nuclear power facility is subject to the conditions in paragraphs (f) and (g) of this section.

Each construction permit for a utilization facility is subject to the following conditions in addition to those specified in 50.55. (the conditions are laid out in subsection (a) through (h) as follows.)

(a) Establishes general requirements and applicability. Specifically, it requires that Structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. To accomplish this it requires that systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of ASME Boiler and Pressure Vessel Code specified in paragraphs (b), (c), (d), (e), (f), and (g) of this section. Protection systems of nuclear power reactors of all types must meet the requirements specified in paragraph (h) of this section. 50.55a(a) does allow alternatives to the requirements of paragraphs (c) through (h) when authorized by the Director of the Office of Nuclear Reactor Regulation.

b) The ASME Boiler and Pressure Vessel Code
c) Reactor coolant pressure boundary.
d) Quality Group B components.
e) Quality Group C components.
f) Inservice testing requirements.
g) Inservice inspection requirements.
h) Protection systems.

Potential Impact Number: 20564 **SRP Section Number:** 9.5.4

Document: RG 1.9
Title: SELECTION, DESIGN, AND QUALIFICATION OF DIESEL-GENERATOR UNITS USED AS STANDBY (ONSITE) ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

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Summary:

This revision of Regulatory Guide 1.9 integrates into a single regulatory guide pertinent guidance previously addressed in Revision 2 of Regulatory Guide 1.9, Revision 1 of Regulatory Guide 1.108, and Generic Letter 84-15, and it endorses, as appropriate, guidelines set forth in IEEE Std 387-1984.

Potential Impact Number: 20628 **SRP Section Number:** 9.5.7

Document: RG 1.9
Title: SELECTION, DESIGN, AND QUALIFICATION OF DIESEL-GENERATOR UNITS USED AS STANDBY (ONSITE) ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

Summary:

Diesel generator shall meet the requirements of IEEE Standard 387-1984.

Regulatory Position C-1.5 through C-1.8 includes positions on design features to be provided by the diesel generator unit and its support systems. These sections supplement the IEEE requirements for surveillance systems and diesel generator protective trips.

Potential Impact Number: 20640 **SRP Section Number:** 9.5.8

Document: RG 1.9
Title: SELECTION, DESIGN, AND QUALIFICATION OF DIESEL-GENERATOR UNITS USED AS STANDBY (ONSITE) ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

Summary:

Diesel generator shall meet the requirements of IEEE Standard 387-1984.

Regulatory Position C-1.5 through C-1.8 includes positions on design features to be provided by the diesel generator unit and its support systems. These sections

supplement the IEEE requirements for surveillance systems and diesel generator protective trips.

Potential Impact Number: 20674 **SRP Section Number:** 15.6.4

Document: RG 1.5
Title: ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A STEAM LINE BREAK ACCIDENT FOR BOILING WATER REACTORS

Summary:

This safety guide gives acceptable assumptions that may be used in evaluating the radiological consequences of this accident for a boiling water reactor.

Potential Impact Number: 20702 **SRP Section Number:** 4.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.46(a)(1)

Summary:

Each boiling and pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy cladding must be provided with an emergency core cooling system (ECCS) that must be designed such that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must

be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II, Required Documentation, sets forth the documentation requirements for each evaluation model.

Potential Impact Number: 20729 **SRP Section Number:** 3.5.1.4

Document: RG 1.76 **Revision:** 0
Title: DESIGN BASIS TORNADO FOR NUCLEAR POWER PLANTS

Summary:

This guide describes a design basis tornado acceptable to the staff for each of three regions within the contiguous United States that a nuclear power plant should be designed to withstand without undue risk to the health and safety of the public.

C.1 - Nuclear power plants should be designed to withstand the Design Basis Tornado (DBT). The values of the parameters specified in Table 1 of this guide for the appropriate regions of Figure 1 of this guide are generally acceptable to the staff for defining the DBT for a nuclear power plant. Sites located near the general boundaries of adjoining regions may involve additional consideration.

C.2 - If a DBT proposed for a given site is characterized by less conservative values for the parameters than the regional values in Table 1, a comprehensive analysis should be provided to justify the selection of the less conservative design basis tornado.

Potential Impact Number: 20749 **SRP Section Number:** 3.5.2

Document: RG 1.117 **Revision:** 1
Title: REGULATORY GUIDE 1.117 TORNADO DESIGN CLASSIFICATION

Summary:

This guide describes a method acceptable to the NRC staff for identifying those structures, systems, and components of light-water-cooled reactors that should be protected from the effects of the Design Basis Tornado including tornado missiles, and remain functional. In order to ensure protection for more probable events having less

severe consequences (than the design basis tornado), the selection of structures, systems, and components to be protected against the effects of a DBT is based on not allowing offsite exposures to exceed an appropriate fraction (25%) of 10 CFR Part 100 guidelines. This guide is referenced throughout SRP Section 3.5.2. Meeting Positions C.1, C.2, C.3, and the appendix of this guide is cited as a basis for findings of conformance with the relevant provisions of GDC-2 and GDC-4 in the evaluation findings part of SRP Section 3.5.2.

Potential Impact Number: 20798 **SRP Section Number:** 5.2.3

Document: RG 1.37 **Revision:** 0
Title: QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER COOLED NUCLEAR POWER PLANTS

Summary:

This guide describes an acceptable method of complying with the Commission's regulations with regard to quality assurance requirements for on-site cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water-cooled nuclear power plant fluid systems. The requirements and recommendations for on-site cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water- cooled nuclear power plant fluid systems that are included in ANSI N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of 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 regulatory positions (which include application of the standard to the operations phase).

Potential Impact Number: 20820 **SRP Section Number:** 4.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, General Design Criterion 4

Summary:

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Criterion 4 -- Environmental and dynamic effects design bases. Reference to this GDC (as acceptance criteria) should be added to (SRP Section 4.6) to support reviews conducted in accordance with SRP Section 4.6, Part III (Review Procedures), Item 2.

Potential Impact Number: 20823 **SRP Section Number:** 4.6

Document: NRC BULLETIN 80-17
Title: FAILURE OF 76 OF 185 CONTROL RODS TO FULLY INSERT DURING A SCRAM AT A BWR

Summary:

The bulletin describes an event (and its preliminary causes) at Browns Ferry Unit 3 (a BWR) involving the failure of 76 control rods to fully insert following a manual scram (and following several subsequent manual attempts to scram). The cause of the failure was preliminarily attributed to high level in the scram discharge volume. Extensive operating BWR licensee actions (short term actions e.g. increased surveillance, operator training, procedure upgrades, data gathering, reporting, etc.) are mandated.

Potential Impact Number: 20824 **SRP Section Number:** 4.6

Document: NRC BULLETIN 80-17 Sup 1
Title: FAILURE OF 76 OF 185 CONTROL RODS TO FULLY INSERT DURING A SCRAM AT A BWR

Summary:

The bulletin describes an event (and its preliminary causes) at Browns Ferry Unit 3 (a BWR) involving the failure of 76 control rods to fully insert following a manual scram (and following several subsequent manual attempts to scram). The cause of the failure was attributed to high level (possibly due to inadequate draining or venting of the system) in the scram discharge volume. Supplement 1 provides further short term compensatory/corrective actions for licensees of operating BWRs and intermediate term corrective actions to install a Continuous Monitoring System (CMS) to improve monitoring of the level in the discharge volume system and to perform a design improvement study for the discharge volume venting system design.

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Potential Impact Number: 20825 **SRP Section Number:** 4.6

Document: NRC BULLETIN 80-17 Sup 2
Title: FAILURES REVEALED BY TESTING SUBSEQUENT TO FAILURE OF CONTROL RODS TO INSERT DURING A SCRAM AT A BWR

Summary:

The bulletin describes an event (and its preliminary causes) at Browns Ferry Unit 3 (a BWR) involving the failure of 76 control rods to fully insert following a manual scram (and following several subsequent manual attempts to scram). The cause of the failure was attributed to high level in the scram discharge volume. Supplement 2 describes further scram/CRD system deficiencies identified by licensees while performing short term testing/data gathering required by bulletin 80-17. Short term actions for operating BWR licensees are mandated for providing alternate vent paths for discharge volume systems whose venting relies upon components other than the vent valve alone.

Potential Impact Number: 20826 **SRP Section Number:** 10.4.9

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

The Office of Nuclear Reactor Regulation is requiring reevaluation of the auxiliary feedwater (AFW) systems for all PWR operating plant licensees and operating license applications. This action includes: (1) Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater-transient conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities, and test and maintenance outages; (2) Perform 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 as principal guidance; and (3) Reevaluate the AFW system flowrate design bases and criteria.

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Potential Impact Number: 20827 **SRP Section Number:** 4.6

Document: NRC BULLETIN 80-17 Sup 3
Title: FAILURE OF CONTROL RODS TO INSERT DURING A SCRAM AT A BWR

Summary:

The bulletin describes an event (and its preliminary causes) at Browns Ferry Unit 3 (a BWR) involving the failure of 76 control rods to fully insert following several manual attempts to scram. Supplement 3 describes a potential single failure mechanism identified during staff review of events/conditions surrounding the issuance of bulletin 80-17. The potential single failure was also discussed in IE Information Notice 80-30. It involves gradual or partial loss of control air system pressure, which could cause partial opening of scram outlet valves without rod motion. The resultant accumulated leakage could fill the scram discharge volume in a few minutes. The possibility exists for the scram discharge volume to fill to a level which could prevent reactor scram before automatic protective action or before the operators could be warned. Further short term licensee actions are mandated including upgraded manual scram procedures/requirements and functional testing of discharge volume level instrumentation following scrams.

Potential Impact Number: 20828 **SRP Section Number:** 4.6

Document: NRC BULLETIN 80-17 Sup 4
Title: FAILURE OF CONTROL RODS TO INSERT DURING A SCRAM AT A BWR

Summary:

Provides additional requirements (above those required by Bulletin 80-17 and Supplements 1-3) for BWR Scram Discharge Volume and Instrument Discharge Volume (SDV and SIDV)Continuous Monitoring Systems (CMS). A description of observed (at Dresden) design/installation deficiencies is provided along with discussion of outstanding staff concerns. Operating BWR licensees are requested to take extensive actions to initially test, confirm the correctness of installation, develop various required procedures, and assure the operability (via frequent periodic surveillance testing) of CMS systems, including high (SDV or SIDV) level automatic scram functions.

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Potential Impact Number: 20830 **SRP Section Number:** 10.4.9

Document: NRC GENERIC LETTER 88-03
Title: RESOLUTION OF GENERIC SAFETY ISSUE 93, 'STEAM BINDING OF AUXILIARY FEEDWATER PUMPS'

Summary:

The issue concerns the potential disabling of auxiliary feedwater (AFW) pumps by steam binding that is caused by backleakage of main feedwater (MFW) past the isolation check valves between the AFW and MFW systems. The issue is significant because most AFW systems are potentially vulnerable to common mode failure of the redundant AFW pumps as a result of steam binding.

Potential Impact Number: 20834 **SRP Section Number:** 4.6

Document: NRC GENERIC LETTER 81-09
Title: BWR SCRAM DISCHARGE SYSTEM

Summary:

The letter transmits Supplement 1 to the staff SER of December 1, 1980 regarding BWR scram discharge volume systems. The original SER included a requirement for installation of a system (which is not required to meet quality and design standards required for the reactor protection system) that will actuate to dump the air in the control air system prior to reaching an air pressure which would open the scram outlet valves. SER Supplement 1 provides a basis for continued operation and revised implementation schedule for plants which have not installed such a system in accordance with the original schedule. SER Supplement 1 also modifies requirements of IE Bulletin 80-17 Supplement 3 by requiring (for human factors reasons) a manual scram upon receipt of a distinct sounding (licensee to modify if necessary) loss of air annunciator.

Potential Impact Number: 20837 **SRP Section Number:** 4.6

Document: NRC GENERIC LETTER 81-18

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Title: BWR SCRAM DISCHARGE SYSTEM; CLARIFICATION OF DIVERSE INSTRUMENTATION REQUIREMENT

Summary:

Clarifies staff position on the diverse instrument requirement for BWR scram discharge system level monitoring instrumentation (Position originally taken in Dec. 1, 1980 staff SER). Two alternatives for acceptable compliance to meet the diversity requirement were originally given. The letter modifies part d of original alternative 2 by requiring additional (or substitute) float assemblies (including level switches) for automatic scram which are made by a different manufacturer (for plants that continue to use original float type level sensing instrumentation).

Potential Impact Number: 20893 **SRP Section Number:** 15.1.1 - 15.1.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34 (f) (xvi)

Summary:

(xvi) Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only). (II.E.5.1).
Assure that any design modifications resulting from the resolution of TMI Item, stated above, are properly accounted for in the analysis of the transient addressed in the subject SRP section.

Potential Impact Number: 20930 **SRP Section Number:** 3.8.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR 50.34 , (f)(3)(iv), (f)(3)(v) & (f)(3)(vi)

Summary:

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Contents of application; technical information

Potential Impact Number: 21027 **SRP Section Number:** 2.4.12

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.17 (a)(1)

Summary:

The application (for an early site permit) must contain the information required by 10 CFR 50.33 (a)-(d), the first three sentences of 50.34(a)(1), and, to the extent approval of emergency plans is sought under paragraph (b)(2)(ii) of this section, the information required by 50.33 (g) and (j), and 50.34(b)(6)(v). In particular, the application should describe the following: ... (vi) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site (see Appendix A to 10 CFR Part 100)...

Potential Impact Number: 21032 **SRP Section Number:** 5.3.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.61

Summary:

For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall submit projected values of RT(PTS) for reactor vessel beltline materials. The assessment must use the calculative procedures given in paragraph (b)(2) of this section. The assessment must specify the bases for the projection, including the assumptions regarding core loading patterns. The submittal must list the copper and nickel contents, and the fluence values used in the calculation for each beltline material.

For each pressurized water nuclear power reactor for which the value of RT(PTS) for any material in the beltline is projected to exceed the PTS screening criterion before the expiration date of the operating license, or the projected expiration data if a change in

the license has been requested. or the end of a renewal term if a request for license renewal has been submitted, the licensee shall submit an analysis and schedule for implementation of such flux reduction programs as are reasonably practicable to avoid exceeding the PTS screening criterion set forth in paragraph (b)(2) of this section.

Potential Impact Number: 21035 **SRP Section Number:** 3.9.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 4

Summary:

Criterion 4 -- Environmental and dynamic effects design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Potential Impact Number: 21044 **SRP Section Number:** 3.8.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 4

Summary:

Environmental and dynamic effects design bases

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Potential Impact Number: 21072 **SRP Section Number:** 12.5

Document: RG 8.25 **Revision:** 1 , 6/1/92
Title: AIR SAMPLING IN THE WORKPLACE

Summary:

This guide identifies methods acceptable to the NRC staff for calibrating air sampling instruments for total volume of air sampled and provides guidance for ensuring that volumes of air sampled are known within certain limits.

Potential Impact Number: 21095 **SRP Section Number:** 3.8.3

Document: RG 1.136 **Revision:** 2
Title: MATERIALS, CONSTRUCTION, AND TESTING OF CONCRETE CONTAINMENTS (Articles CC-1000, -2000, and 4000 through 6000 of the Code for Concrete Reactor Vessels and Containments)

Summary:

The requirements of Articles CC-1000,-2000, and -4000 through -6000 of the Code for Concrete Reactor Vessels and Containments, ASME Boiler and Pressure Vessel Code, Section III, Division 2, 1980 Edition, are acceptable to the NRC staff for the materials, construction, and testing of concrete containments of nuclear power plants subject to certain conditions.

Additionally, Regulatory Guides 1.10, 1.15, 1.18, 1.19, 1.55, and 1.103 were withdrawn on July 8, 1981.

Potential Impact Number: 21110 **SRP Section Number:** 3.5.1.3

Document: RG 1.115 **Revision:** 1
Title: PROTECTION AGAINST LOW-TRAJECTORY TURBINE MISSILES

Summary:

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Potential Impacts

This guide 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.

Potential Impact Number: 21206 **SRP Section Number:** 11.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Part VI, Criterion 61

Summary:

Criterion 61 -- Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems.

Potential Impact Number: 21215 **SRP Section Number:** 4.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50, Appendix G, V.A

Summary:

The effects of neutron radiation on the reference temperature and upper shelf energy of reactor vessel beltline materials, including welds, are to be predicted from the results of pertinent radiation effects studies in addition to the results of the surveillance program of Appendix H to this part.

Potential Impact Number: 21216 **SRP Section Number:** 4.3

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Potential Impacts

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50, Appendix H, II.A

Summary:

No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E>1MEV) at the end of the design life of the vessel will not exceed 10**17 n/square cm.

Potential Impact Number: 21229 **SRP Section Number:** 11.3

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION
Location: 10 CFR 20.106

Summary:

Provides limits on concentrations of radioactive materials in effluents (from licensed activities) to unrestricted areas, provides conditions under which licensees may propose higher limits, specifies that limits apply at the boundary between restricted and unrestricted areas, and provides flexibility for NRC to impose temporary limits when warranted.

Potential Impact Number: 21246 **SRP Section Number:** 11.3

Document: NRC GENERIC LETTER 89-01
Title: IMPLEMENTATION OF PROGRAMMATIC CONTROLS FOR RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS IN THE ADMINISTRATIVE CONTROLS SECTION OF THE TECHNICAL SPECIFICATIONS AND THE RELOCATION OF PROCEDURAL DETAILS OF RETS TO THE OFFSITE DOSE CALCULATION MANUAL OR

Summary:

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The NRC staff has examined the contents of the Radiological Effluent Technical Specifications (RETS) in relation to the Commission's Interim Policy Statement on Technical Specification Improvements. The staff has determined 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 (ODCM). These actions simplify the RETS, meet the regulatory requirements for radioactive effluents and radiological environmental monitoring, and are provided as a line-item improvement of the TS, consistent with the goals of the Policy Statement. New programmatic controls for radioactive effluents and radiological environmental monitoring are incorporated in the TS to conform to the regulatory requirements of 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50.

Potential Impact Number: 21297 **SRP Section Number:** 2.3.4

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.17 (a)(1)(vi)

Summary:

The application (for an early site permit) must contain the information required by 10 CFR 50.33 (a)-(d), the first three sentences of 50.34(a)(1), and, to the extent approval of emergency plans is sought under paragraph (b)(2)(ii) of this section, the information required by 50.33 (g) and (j), and 50.34(b)(6)(v). In particular, the application should describe the following:... (vi) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site (see Appendix A to 10 CFR Part 100)...

Potential Impact Number: 21298 **SRP Section Number:** 2.3.4

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.47 (a)(1)

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Summary:

(a) The requirements of this paragraph apply to all applications for design certification. (1) An application for design certification must contain: (i) The technical information which is required of applicants for construction permits and operating licenses by 10 CFR Part 20, Part 50 and its appendices, and Parts 73 and 100, and which is technically relevant to the design and not site-specific; (ii) Demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f); (iii) The site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters....

Potential Impact Number: 21316 **SRP Section Number:** 2.3.5

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.47 (a)(1)

Summary:

(a) The requirements of this paragraph apply to all applications for design certification. (1) ... An application for design certification must contain: ... (iii) The site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters....

Potential Impact Number: 21317 **SRP Section Number:** 2.3.5

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.17 (a)(1)

Summary:

The application (for an early site permit) must contain the information required by 10 CFR 50.33 (a)-(d), the first three sentences of 50.34(a)(1), and, to the extent approval

of emergency plans is sought under paragraph (b)(2)(ii) of this section, the information required by 50.33 (g) and (j), and 50.34(b)(6)(v). In particular, the application should describe the following: ... (iv) The anticipated maximum levels of radiological and thermal effluents each facility will produce.

Potential Impact Number: 21318 **SRP Section Number:** 2.3.5

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.79 (b)

Summary:

The application (for a combined license) must contain the technically relevant information required of applicants for an operating license by 10 CFR 50.34. The final safety analysis report and other required information may incorporate by reference the final safety analysis report for a certified standard design. In particular, an application referencing certified design must describe those portions of the design which are site-specific...

Potential Impact Number: 21373 **SRP Section Number:** 15.6.5.B

Document: RG 1.7
Title: CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT

Summary:

50% of the halogens and 1% of the solids present in the core are intimately mixed with the coolant water in accordance with the values listed in Table 1 of Reg 1.7.

Potential Impact Number: 21374 **SRP Section Number:** 3.11

Document: NRC POLICY STATEMENT 49 FR 8422

APPENDIX II
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Title: ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT

Summary:

The United States Court of Appeals for the District of Columbia Circuit vacated and remanded a Commission rule which removed from nuclear power plant operating licenses a June 30, 1982 deadline for the completion of the environmental qualification of certain safety-related electrical equipment. The Court remanded to the Commission with direction to obtain public comments on the current documentation justifying the continued operation of nuclear power plants pending the completion of the environmental qualification program. This Statement of Policy is intended to explain the Commission's response to the D.C. Circuit's remand and to describe other related actions the NRC will take until the conclusion of the rulemaking proceeding which the Commission intends to initiate by an accompanying Notice of Proposed Rulemaking.

Potential Impact Number: 21382 **SRP Section Number:** 3.2.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix B

Summary:

This appendix establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components. The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those structures, systems, and components; these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying. As used in this appendix, quality assurance comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements. The Appendix requires an 18 point quality assurance program (corresponding to the 18 chapters of the Appendix). Per SRP Section 3.2.1 and regulatory position C.4 of Reg.

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Guide 1.29, the pertinent quality assurance requirements of Appendix B to 10 CFR Part 5O should be applied to all activities affecting the safety-related functions of those portions of structures, systems, and components designated as seismic category I.

Potential Impact Number: 21391 **SRP Section Number:** 4.5.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix G

Summary:

This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Code forms the basis for the requirements of this appendix. ASME Code means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. If no section is specified, the reference is to Section III, Division 1, Rules for Construction of Nuclear Power Plant Components. Section XI means Section XI, Division 1, Rules for Inservice Inspection of Nuclear Power Plant Components. If no edition or addenda is specified, the applicable ASME Code edition and addenda and any limitations and modifications thereof are specified in Section 50.55a of this part.

Potential Impact Number: 21441 **SRP Section Number:** 3.6.2

Document: NRC GENERIC LETTER 84-04
Title: SAFETY EVALUATION OF WESTINGHOUSE TOPICAL REPORTS DEALING WITH ELIMINATION OF POSTULATED PIPE BREAKS IN PWR PRIMARY MAIN LOOPS

Summary:

Information concerning break and crack location criteria and methods of analysis for evaluating the dynamic effects associated with postulated breaks and cracks in high-

and moderate-energy fluid system piping, including 'field run' piping, inside and outside of containment should be provided in the applicant's safety analysis report (SAR).

Potential Impact Number: 21442 **SRP Section Number:** 9.1.5

Document: NRC GENERIC LETTER 85-11
Title: COMPLETION OF PHASE II OF 'CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS' NUREG-0612.

Summary:

The design layout, which shows the functional geometric layout of the handling equipment, including the areas of movement over and around the fixed locations of safety-related facilities during the handling of heavy loads, is reviewed to determine that the various handling operations can be performed safely. The main emphasis in the OHLHS review is on critical load handling in which inadvertent operations or equipment malfunctions, either separately or in combination, could cause a release of radioactivity, a criticality accident, the inability to cool fuel within the reactor vessel or spent fuel pool or prevent safe shutdown of the reactor.

Potential Impact Number: 21443 **SRP Section Number:** 9.1.5

Document: NRC GENERIC LETTER 83-42
Title: CLARIFICATION TO GENERIC LETTER 81-07 REGARDING RESPONSE TO NUREG-0612, 'CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS'

Summary:

The design layout, which shows the functional geometric layout of the handling equipment, including the areas of movement over and around the fixed locations of safety-related facilities during the handling of heavy loads, is reviewed to determine that the various handling operations can be performed safely. The main emphasis in the OHLHS review is on critical load handling in which inadvertent operations or equipment malfunctions, either separately or in combination, could cause a release of radioactivity, a criticality accident, the inability to cool fuel within the reactor vessel or spent fuel pool or prevent safe shutdown of the reactor.

APPENDIX II
Potential Impacts

Potential Impact Number: 21527 **SRP Section Number:** 9.5.1

Document: NRC BULLETIN 92-01 Revision Date: 6/24/92
Title: FAILURE OF THERMO-LAG 330 FIRE BARRIER SYSTEM TO MAINTAIN CABLING IN WIDE CABLE TRAYS AND SMALL CONDUITS FREE FROM FIRE DAMAGE

Summary:

NRC BULLETIN NO. 92-01 dealt with the failure of Thermo-Lag 330 fire barrier system to maintain cabling in wide cable trays and small conduits free from fire damage. The Code of Federal Regulations (10 CFR 50.48(a)) requires that each operating nuclear power plant have a fire protection plan that satisfies Appendix A to 10 CFR Part 50, General Design Criteria (GDC) 3, "Fire Protection." The NRC considered the failures of the Thermo-Lag fire barrier fire endurance testing and has determined that the 1- and 3-hour pre-formed assemblies installed on small conduit and wide cable trays (wider than 14 inches) do not provide the level of safety as required by NRC requirements. Licensees whose plants contained Thermo-Lag 330 were required to take specific actions to assure proper fire protection.

Potential Impact Number: 21528 **SRP Section Number:** 9.5.1

Document: NRC BULLETIN 92-01 SUP 1 Revision Date: 8/28/92
Title: FAILURE OF THERMO-LAG 330 FIRE BARRIER SYSTEM TO PERFORM ITS SPECIFIED FIRE ENDURANCE FUNCTION

Summary:

NRC BULLETIN NO. 92-01 dealt with the failure of Thermo-Lag 330 fire barrier system to maintain cabling in wide cable trays and small conduits free from fire damage. The Code of Federal Regulations (10 CFR 50.48(a)) requires that each operating nuclear power plant have a fire protection plan that satisfies Appendix A to 10 CFR Part 50, General Design Criteria (GDC) 3, "Fire Protection." The NRC considered the failures of the Thermo-Lag fire barrier fire endurance testing and has determined that the 1- and 3-hour pre-formed assemblies installed on small conduit and wide cable trays (wider than 14 inches) do not provide the level of safety as required by NRC requirements. Licensees whose plants contained Thermo-Lag 330 were required to take specific

actions to assure proper fire protection. These actions are essentially the same as those listed in Bulletin 92-01, but the scope has been expanded to include all sizes of conduits and trays and to include walls, ceilings, and equipment enclosures.

Potential Impact Number: 21593 **SRP Section Number:** 3.9.6

Document: NRC GENERIC LETTER 80-14 Revision Date: 2/23/80
Title: LWR PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

Summary:

Generic Letter 80-14 addresses ECCS check valve Event V concerns. The generic letter outlines staff positions and acceptable methods to assure component integrity. Acceptable methods to assure component integrity include: (1) continuous pressure monitoring on the low pressure side of each check valve, (2) periodic IST leakage testing on each valve every time the plant is shutdown and each time either check valve is moved from the fully closed position, (3) periodic ultrasonic examination on each valve every time the plant is shutdown and each time either check valve is moved from the fully closed position, or (4) periodic radiographic examination on each valve every time the plant is shutdown and each time either check valve is moved from the fully closed position. Various combinations of the above methods may be acceptable. Likewise, other equivalent provisions may also be found acceptable.

Potential Impact Number: 21609 **SRP Section Number:** 3.9.6

Document: NRC GENERIC LETTER 89-10 SUP 4 Revision Date: 2/12/92
Title: CONSIDERATION OF VALVE MISPOSITIONING IN BOILING WATER REACTORS

Summary:

In Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve (MOV) Testing and Surveillance," the staff recommended, among other things, that any MOV in a safety-related system that is not blocked from inadvertent operation from either the control room, the motor control center, or the valve itself be considered capable of being mispositioned (referred to as position-changeable MOVs) and be included in licensees' MOV programs. When determining the maximum differential pressure or flow

for position-changeable MOVs, the licensee should consider the fact that the MOV must be able to recover from mispositioning. Supplement 1 to GL 89-10 limited the prevention of inadvertent MOV operation within the context of the generic letter to the potential for MOV mispositioning from the control room.

The Boiling Water Reactors Owners' Group (BWROG) submitted a backfit appeal on the recommendations for position-changeable valves. The staff has reviewed and evaluated the issues concerning mispositioning of valves from the control room and has determined that the recommendations in GL 89-10 should be changed for boiling water reactors (BWRs).

The staff no longer considers the recommendations for inadvertent operation of MOVs from the control room to be within the scope of GL 89-10 for BWRs. However, the staff believes that consideration of valve mispositioning benefits safety.

Modifying the provisions in GL 89-10 for valve mispositioning does not affect the GL 89-10 provisions for licensees to review safety analyses, emergency procedures, and other plant documentation to determine the design basis fluid conditions under which all MOVs in safety-related piping systems may intentionally be called upon to function. This position also does not supersede the NRC's generic recommendations or regulations on valve mispositioning that pertain to other issues such as intersystem loss-of-coolant accidents (ISLOCA) or fire protection (10 CFR 50, Appendix R).

Potential Impact Number: 21614 **SRP Section Number:** 5.3.1

Document: NRC GENERIC LETTER 91-17 Revision Date: 10/17/91
Title: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

Generic letter 91-17 provides information on the U.S. Nuclear Regulatory Commission's resolution of Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants." The NRC has resolved this issue without developing any new requirements. The bases for resolution are further documented in the enclosed NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants".

NUREG-1339 recommends that a new section of the SRP be prepared to provide guidance to the staff for the review of future plants. It is recognized that NUREG-1339

makes an exception for closure studs which are already addressed in SRP sections 5.3.1 and 5.2.3, however, the staff noted the absence of an SRP section on general reviews of bolting and recommended that one be prepared and issued.

Potential Impact Number: 21615 **SRP Section Number:** 5.3.1

Document: NRC GENERIC LETTER 92-01 **Revision:** 1 , 3/6/92
Title: REACTOR VESSEL STRUCTURAL INTEGRITY, 10 CFR 50.54(f)

Summary:

Generic Letter 92-01, Revision 1 is part of a program to evaluate reactor vessel integrity. The backfit discussion stated that no new staff positions are being established for compliance in this generic letter. Enclosure 1 contains a discussion of the regulatory requirements applicable to reactor vessel structural integrity.

Potential Impact Number: 21616 **SRP Section Number:** 5.3.2

Document: NRC GENERIC LETTER 92-01 **Revision:** 1 , 3/6/92
Title: REACTOR VESSEL STRUCTURAL INTEGRITY, 10 CFR 50.54(f)

Summary:

Generic Letter 92-01, Revision 1 is part of a program to evaluate reactor vessel integrity. The backfit discussion stated that no new staff positions are being established for compliance in this generic letter. Enclosure 1 contains a discussion of the regulatory requirements applicable to reactor vessel structural integrity.

Potential Impact Number: 21617 **SRP Section Number:** 5.3.3

Document: NRC GENERIC LETTER 92-01 **Revision:** 1 , 3/6/92
Title: REACTOR VESSEL STRUCTURAL INTEGRITY, 10 CFR 50.54(f)

Summary:

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Generic Letter 92-01, Revision 1 is part of a program to evaluate reactor vessel integrity. The backfit discussion stated that no new staff positions are being established for compliance in this generic letter. Enclosure 1 contains a discussion of the regulatory requirements applicable to reactor vessel structural integrity.

Potential Impact Number: 21621 **SRP Section Number:** 3.2.1

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The ABWR design eliminates the main steam isolation valve leakage control system for the ABWR plant design. 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 core damage. 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 position, which was discussed in Section II.E of SECY-93-087 and was approved by the Commission in its SRM dated July 21, 1993, is that the main steam piping beyond the outermost isolation valve up to the seismic interface restraint and connecting branch lines up to the first normally closed valve be classified as QG B (SC 2) and seismic Category I. The MSL from the seismic interface restraint up to but not including the turbine stop valve (including branch lines to the first normally closed valve) will be classified as QG B and inspected in accordance with the applicable portions of ASME Section XI. This portion of the steamline may be classified as non-seismic Category I if it has been analyzed using a dynamic seismic analysis method to demonstrate its structural integrity under SSE loading conditions. However, all pertinent QA requirements of Appendix B to 10 CFR Part 50 are applicable to ensure that the quality of the piping material is commensurate with its importance to safety during normal operational, transient, and accident conditions. To ensure the integrity of the remainder of GE's proposed alternative leakage path, the staff position is that (1) the main steam piping between the turbine stop valve and the turbine inlet, the turbine bypass line from the bypass valve to the condenser, and the main steam drain line from the first valve to the condenser are not required to be classified as safety-related or as seismic Category I, but should be analyzed using a dynamic seismic analysis to demonstrate their structural integrity under SSE loading conditions, and (2) the

condenser anchorage shall be seismically analyzed to demonstrate that it is capable of sustaining the SSE loading conditions without failure.

Potential Impact Number: 21626 **SRP Section Number:** 3.2.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Seismic Category I includes all SSC designated as Safety Class -1, -2, and -3; consistent with RG 1.29. Seismic Category II includes SSCs that perform no safety function and whose continued operation is not required, but whose structural failure or interaction could degrade the functioning of a Seismic Category I SSC to an unacceptable level. Staff required application of applicable parts of an Appendix B QA program and commitment to Positions C.2 and C.4 of RG 1.29. Non-Seismic includes those SSC not classified as Seismic Category I or II.

Potential Impact Number: 21627 **SRP Section Number:** 3.7.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

FSER requires that applicable non-seismic Category (Seismic Category II - those SSC whose structural failure or interaction could degrade the functioning of a seismic Category I structure, system, or component to an unacceptable safety level, or could result in incapacitating injury to occupants of the control room) structures and systems be analyzed using the same type of dynamic seismic analysis methodology as that used for seismic Category I structures and systems.

Potential Impact Number: 21634 **SRP Section Number:** 3.2.1

Document: NUREG 0800

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Potential Impacts

Title: STANDARD REVIEW PLAN

Summary:

Review Procedures (Section III.) and Evaluation Findings (IV.) provide for confirmation that Appendix B QA Program applies to Safety Related SSCs; no corresponding Acceptance Criterion exists.

Potential Impact Number: 21671 **SRP Section Number:** 5.2.3

Document: NRC GENERIC LETTER 91-17 Revision Date: 10/17/91
Title: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

Generic letter 91-17 provides information on the U.S. Nuclear Regulatory Commission's resolution of Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants." The NRC has resolved this issue without developing any new requirements. The bases for resolution are further documented in the enclosed NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants".

NUREG-1339 recommends that a new section of the SRP be prepared to provide guidance to the staff for the review of future plants. It is recognized that NUREG-1339 makes an exception for closure studs which are already addressed in SRP sections 5.3.1 and 5.2.3, however, the staff noted the absence of an SRP section on general reviews of bolting and recommended that one be prepared and issued.

Potential Impact Number: 21682 **SRP Section Number:** 3.6.1

Document: NUREG 1061 VOL 3
Title: REPORT OF U.S. NRC PIPING REVIEW COMMITTEE - EVALUATION OF POTENTIAL FOR PIPE BREAKS

Summary:

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NUREG-1061, Vol. 3 is referenced in GDC-4 rulemakings (52 FR 41288). Microfiche 27849:092. The Staff established a position that elimination of Double-Ended Guillotine Break (DEGB) at terminal ends of large primary pipes in PWRs and the control of the maximum flaw length in piping in general should permit an elimination of existing restraints or removal of restraints as a design requirement. Consequently, asymmetric RPV loads, jet impingement loads, and reactor cavity overpressurization that result from the postulated DEGB need not be considered. In BWRs where piping of safety significance has been replaced with a material resistant to IGSCC, the DEGB should not be a design criterion. Section 5.0 provides acceptance criteria for leak-before-break (LBB) submittals.

Potential Impact Number: 21686 **SRP Section Number:** 3.6.2

Document: NUREG 1061 VOL 3
Title: REPORT OF U.S. NRC PIPING REVIEW COMMITTEE - EVALUATION OF POTENTIAL FOR PIPE BREAKS

Summary:

NUREG-1061, Vol. 3 is referenced in GL 84-04 and in GDC-4 rulemakings. Microfiche 27849:092. The Staff established a position that elimination of Double-Ended Guillotine Break (DEGB) at terminal ends of large primary pipes in PWRs and the control of the maximum flaw length in piping in general should permit an elimination of existing restraints or removal of restraints as a design requirement. Consequently, asymmetric RPV loads, jet impingement loads, and reactor cavity overpressurization that result from the postulated DEGB need not be considered. In BWRs where piping of safety significance has been replaced with a material resistant to IGSCC, the DEGB should not be a design criterion. Section 5.0 provides acceptance criteria for leak-before-break (LBB) submittals. NUREG 1061, Vol. 3 referenced in GDC-4 rulemaking, 52 FR 41288.

Potential Impact Number: 21689 **SRP Section Number:** 3.6.2

Document: NUREG 1061 VOL 3
Title: REPORT OF U.S. NRC PIPING REVIEW COMMITTEE - EVALUATION OF POTENTIAL FOR PIPE BREAKS

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Summary:

NUREG-1061, Vol. 3 is referenced in GL 84-04 and in GDC-4 rulemakings. Microfiche 27849:092. The Staff has taken a position that arbitrary intermediate pipe breaks required by BTP MEB 3-1 should be eliminated as a design basis requirement and the BTP should be revised accordingly. NUREG 1061, Vol. 3 referenced in GDC-4 rulemaking 52 FR 41288. [Note: BTP revision promulgated with GL 87-11.]

Potential Impact Number: 21699 **SRP Section Number:** 3.6.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

To be consistent with the implementation of the LBB technology under the recent broad-scope amendment of GDC4 of Appendix A to 10 CFR Part 50, containment and ECCS function and performance requirements are maintained. The staff's current interpretation of this rule change is that the above local dynamic effects (pipe whip, jet impingement, rapid subcompartment pressurization, hydraulic system internal loads, and the motion of attached equipment) can be excluded from the containment design basis.

Potential Impact Number: 21708 **SRP Section Number:** 3.6.2

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff is currently using the methodology and acceptance criteria provided in SRP section 3.6.3 and NUREG-1061, Volume 3 for evaluation of Leak-Before-Break (LLB). The Staff recognizes that the LBB technology is continually evolving and, therefore, will review LBB requests on a case-by-case basis using the staff's methodology and acceptance criteria in effect at the time of the submittal. The analyses referred to in GDC 4 (Federal Register, Vol 52, No. 207, October 27, 1987, 41288-41295) should be based on specific plant data.

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Potential Impact Number: 21721 **SRP Section Number:** 3.6.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 50, p 27006, Federal Register, Vol. 51, p 12502, Federal Register, Vol. 51, p 26393, Federal Register, Vol. 52, p 41288, Federal Register, Vol. 52, p 32626

Summary:

"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 adapted NUREG 1061, Vol 3, Section 5 into a draft SRP Section 3.6.3 titled "Leak-Before-Break Evaluation Procedures" based on input from CRGR, ACRS, Staff and Industry. Application of LBB also eliminates requirement to consider asymmetric loads per NUREG-0609.

Potential Impact Number: 21722 **SRP Section Number:** 3.6.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 50, p 27006, Federal Register, Vol. 51, p 12502, Federal Register, Vol. 51, p 26393, Federal Register, Vol. 52, p 41288, Federal Register, Vol. 52, p 32626

Summary:

"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 adapted NUREG 1061, Vol 3, Section 5 into a draft SRP Section 3.6.3 titled "Leak-Before-Break Evaluation Procedures." Application of LBB also eliminates requirement to consider asymmetric loads per NUREG-0609.

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Potential Impact Number: 21725 **SRP Section Number:** 3.5.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 52, p 41288

Summary:

Rule making in 1987 revised GDC 4, including the title - now reads "Environmental and dynamic effects design bases."

Potential Impact Number: 21730 **SRP Section Number:** 3.5.1.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 52, p 41288

Summary:

Rule making in 1987 revised GDC 4, including the title - now reads "Environmental and dynamic effects design bases."

Potential Impact Number: 21731 **SRP Section Number:** 3.8.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 52, p 41288

Summary:

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Rule making in 1987 revised GDC 4, including the title - now reads "Environmental and dynamic effects design bases."

Potential Impact Number: 21732 **SRP Section Number:** 3.8.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 52, p 41288

Summary:

Rule making in 1987 revised GDC 4, including the title - now reads "Environmental and dynamic effects design bases."

Potential Impact Number: 21733 **SRP Section Number:** 3.8.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 50, p 27006, Federal Register, Vol. 51, p 12502, Federal Register, Vol. 51, p 26393, Federal Register, Vol. 52, p 41288, Federal Register, Vol. 52, p 32626

Summary:

"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 adapted NUREG 1061, Vol 3, Section 5 into a draft SRP Section 3.6.3 titled "Leak-Before-Break Evaluation Procedures." Application of LBB also eliminates requirement to consider asymmetric loads per NUREG-0609.

Potential Impact Number: 21734 **SRP Section Number:** 3.8.4

APPENDIX II
Potential Impacts

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 50, p 27006, Federal Register, Vol. 51, p 12502, Federal Register, Vol. 51, p 26393, Federal Register, Vol. 52, p 41288, Federal Register, Vol. 52, p 32626

Summary:

"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 adapted NUREG 1061, Vol 3, Section 5 into a draft SRP Section 3.6.3 titled "Leak-Before-Break Evaluation Procedures." Application of LBB also eliminates requirement to consider asymmetric loads per NUREG-0609.

Potential Impact Number: 21736 **SRP Section Number:** 3.9.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 50, p 27006, Federal Register, Vol. 51, p 12502, Federal Register, Vol. 51, p 26393, Federal Register, Vol. 52, p 41288, Federal Register, Vol. 52, p 32626

Summary:

"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 adapted NUREG 1061, Vol 3, Section 5 into a draft SRP Section 3.6.3 titled "Leak-Before-Break Evaluation Procedures." Application of LBB also eliminates requirement to consider asymmetric loads per NUREG-0609.

Potential Impact Number: 21739 **SRP Section Number:** 3.10

Document: CODE OF FED. REGS 10CFR50

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Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 50, p 27006, Federal Register, Vol. 51, p 12502, Federal Register, Vol. 51, p 26393, Federal Register, Vol. 52, p 41288, Federal Register, Vol. 52, p 32626

Summary:

"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 adapted NUREG 1061, Vol 3, Section 5 into a draft SRP Section 3.6.3 titled "Leak-Before-Break Evaluation Procedures." Application of LBB also eliminates requirement to consider asymmetric loads per NUREG-0609.

Potential Impact Number: 21742 **SRP Section Number:** 5.3.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 52, p 41288

Summary:

Rule making in 1987 revised GDC 4, including the title - now reads "Environmental and dynamic effects design bases."

Potential Impact Number: 21743 **SRP Section Number:** 5.3.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 52, p 41288

Summary:

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Rule making in 1987 revised GDC 4, including the title - now reads "Environmental and dynamic effects design bases."

Potential Impact Number: 21745 **SRP Section Number:** 5.4.7

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 52, p 41288

Summary:

Rule making in 1987 revised GDC 4, including the title - now reads "Environmental and dynamic effects design bases."

Potential Impact Number: 21747 **SRP Section Number:** 5.4.1.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 52, p 41288

Summary:

Rule making in 1987 revised GDC 4, including the title - now reads "Environmental and dynamic effects design bases." The revision to GDC 4 allows for the application of Leak-Before-Break methods in certain situations in plant design.

Potential Impact Number: 21753 **SRP Section Number:** 6.2.1.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 50, p 27006, Federal Register, Vol. 51, p 12502, Federal Register, Vol. 51, p 26393, Federal Register, Vol. 52, p 41288, Federal Register, Vol. 52, p 32626

APPENDIX II
Potential Impacts

Summary:

"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 adapted NUREG 1061, Vol 3, Section 5 into a draft SRP Section 3.6.3 titled "Leak-Before-Break Evaluation Procedures." Application of LBB also eliminates requirement to consider asymmetric loads per NUREG-0609.

Potential Impact Number: 21754 **SRP Section Number:** 6.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 52, p 41288

Summary:

Rule making in 1987 revised GDC 4, including the title - now reads "Environmental and dynamic effects design bases."

Potential Impact Number: 21767 **SRP Section Number:** 9.3.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 52, p 41288

Summary:

Rule making in 1987 revised GDC 4, including the title - now reads "Environmental and dynamic effects design bases."

Potential Impact Number: 21777 **SRP Section Number:** 10.3

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Potential Impacts

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 52, p 41288

Summary:

Rule making in 1987 revised GDC 4, including the title - now reads "Environmental and dynamic effects design bases."

Potential Impact Number: 21778 **SRP Section Number:** 10.4.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: Appendix A, GDC4, Appendix A, GDC4, Federal Register, Vol. 52, p 41288

Summary:

Rule making in 1987 revised GDC 4, including the title - now reads "Environmental and dynamic effects design bases."

Potential Impact Number: 21786 **SRP Section Number:** 3.6.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The elimination of dynamic effects of postulated high energy pipe ruptures from the design basis of ALWRs using advanced fracture mechanics analyses (leak-before-break approach) is permitted in the revised GDC 4. The staff recommends that the Commission approve the application of the LBB approach to both evolutionary and passive ALWRs seeking design certification under 10 CFR Part 52 when appropriate bounding limits are established during the design certification phase using preliminary analyses results and verified during the combined license phase by performing the appropriate ITAAC discussed herein.

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Potential Impact Number: 21787 **SRP Section Number:** 15.8

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

EPRI specifies that analysis and acceptance criteria for events involving failures of multiple active components associated with anticipated transients without scram (ATWS) will be in accordance with 10 CFR 50.62. However, 10 CFR 50.62 does not specify analysis and acceptance criteria, except for the prescriptive equipment design requirements that were based on analyses of the current generation of LWRs.

The staff concludes that EPRI has not provided adequate requirements for analyses for ATWS events. Therefore, the staff concludes that it will be necessary for designers of ALWRs to demonstrate that the bases for which the ATWS rule was developed remain valid for their proposed design, and, therefore, the ATWS rule requirements are still appropriate. Each evolutionary LWR designer must demonstrate that the new plants will not experience ATWS behavior unexpectedly more severe than that experienced in current plants. The staff will evaluate this matter during its review of an individual application for FDA/DC.

Potential Impact Number: 21789 **SRP Section Number:** 15.8

Document: FINAL SER EPRI CH 4
Title: EPRI URD-EPD: REACTOR SYSTEMS

Summary:

The Evolutionary Requirements Document contains requirements for core characteristics to ensure stable operation for all expected operating conditions. In the DSER for Chapter 4, the staff raised several concerns related to the use of analytic methods to prevent thermal-hydraulic instabilities. The staff recommended that attempts be made to prevent unstable operation by design, even in the "excluded region" and that current methods should be used to establish margins for thermal-hydraulic stability after calculational uncertainties have been included. This was identified as an open issue in the DSER. EPRI's revisions to Chapter 4 to address

this open issue have resulted in requirements that are considerably improved in this area. Section 4.2.1.2.2 now requires that operation beyond stability limits be prevented by an operationally proven, reliable control and instrumentation system. This control and instrumentation system is required to be capable of detecting all expected modes of instability and automatically reducing the reactor power sufficiently to return the reactor to a stable condition before fuel safety limits are exceeded. The staff concludes that these requirements are acceptable. Therefore, this DSER open issue is closed.

As part of its continuing generic study of thermal-hydraulic stability, the staff has raised concerns about thermal-hydraulic stability during an anticipated transient without scram (ATWS). The Evolutionary Requirements Document does not address stability performance during an ATWS. Therefore, the staff will review applications for FDA/DC to ensure that designers demonstrate that the thermal-hydraulic stability performance of the core during an ATWS is acceptable.

Potential Impact Number: 21792 **SRP Section Number:** 15.8

Document: FINAL SER EPRI CH 10
Title: EPRI URD-EPD: MAN-MACHINE INTERFACE SYSTEMS

Summary:

The EPRI Evolutionary Requirements Document required that the PWR designer either provide a diverse backup reactor trip system or demonstrate that the design of the plant has enough features or margin so that the backup system is not warranted. The staff questioned the use of a design margin analysis in lieu of the backup scram system and also asked EPRI to address the use of diverse sensors. EPRI proposed a revision requiring that the designer retain the option of complying with the ATWS rule or demonstrating that the plant can withstand an ATWS event. This was acceptable to the staff (optional compliance with the ATWS rule, and the staff closed a confirmatory item based upon verification that EPRI incorporated this proposed revision).

EPRI also stated that the rule does not require diverse sensors and the Evolutionary Requirements Document does not require them unless the designer determines they are necessary. The staff concludes that this is acceptable.

Analyst Note:

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Annex A to EPRI FSER Chapter 1 includes documents related to the Commission's review and approval of the staff position concerning future evolutionary design compliance with the ATWS rule as stated in SECY 90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements." In response to ACRS comments, the staff changed its position as stated in SECY 90-016 to either provide a diverse scram system or allow the designer to demonstrate that the consequences of an ATWS event are acceptable. In the Staff Requirements Memorandum (SRM) related to SECY 90-016, the Commission approved the position that ALWRs should either include a diverse scram system or a demonstration that the consequences of an ATWS are acceptable.

Potential Impact Number: 21793 **SRP Section Number:** 15.8

Document: NUREG 0460 VOL 1
Title: ANTICIPATED TRANSIENTS WITHOUT SCRAM FOR LIGHT WATER REACTORS

Summary:

Anticipated Transients Without Scram for Light Water Reactors. Vols. 1 and 2 April 1978, Vol. 3 December 1978, Vol. 4 March 1980. This is the NRC staff's review of anticipated transient without scram (ATWS). Studies, findings and recommendations are presented. Volumes 1, 2 and 3 contain preliminary studies, findings, resolution options, criteria (where applicable for resolution options), and recommendations pursuant to regulatory action (e.g. orders, rulemaking, regulatory guide development, etc.). Volume 4 contains the actual proposed resolution of unresolved safety issue (TAP A-9) in the form of requirements recommended to be imposed on licensees and applicants (via Commission Orders and via rulemaking, see 10 CFR 50.62 for ultimate results of rulemaking). Volume 4 describes the proposed requirements and their phased implementation, gives the staff's technical basis, and considers the values and impacts. The appendices describe the recently submitted industry information including analyses and the staff's evaluation of that information (concerns with analyses techniques, computer codes, analyses validity as an ATWS event progresses, etc.), technical details, and some related risk estimates associated with the revised requirements. Guidance and insight is provided with respect to ATWS analyses techniques, acceptable levels of risk, etc. Many of the hardware modification options evaluated for PWR diverse scram systems, PWR diverse logic for AFW initiation and turbine trip, BWR ARI systems, etc. were recommended/assumed to be compliant with IEEE Std 279 for Protection Systems.

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This NUREG (Section IV-4 of Vol 2) is currently cited as Acceptance Criteria for BWR Recirculation Pump Trip in SRP Section 15.8. Information from the NUREG is also included in the Review Procedures for SRP Section 15.8.

Potential Impact Number: 21794 **SRP Section Number:** 15.8

Document: NUREG 0460 VOL 2
Title: ANTICIPATED TRANSIENTS WITHOUT SCRAM FOR LIGHT WATER REACTORS

Summary:

Anticipated Transients Without Scram for Light Water Reactors. Vols. 1 and 2 April 1978, Vol. 3 December 1978, Vol. 4 March 1980. This is the NRC staff's review of anticipated transient without scram (ATWS). Studies, findings and recommendations are presented. Volumes 1, 2 and 3 contain preliminary studies, findings, resolution options, criteria (where applicable for resolution options), and recommendations pursuant to regulatory action (e.g. orders, rulemaking, regulatory guide development, etc.). Volume 4 contains the actual proposed resolution of unresolved safety issue (TAP A-9) in the form of requirements recommended to be imposed on licensees and applicants (via Commission Orders and via rulemaking, see 10 CFR 50.62 for ultimate results of rulemaking). Volume 4 describes the proposed requirements and their phased implementation, gives the staff's technical basis, and considers the values and impacts. The appendices describe the recently submitted industry information including analyses and the staff's evaluation of that information (concerns with analyses techniques, computer codes, analyses validity as an ATWS event progresses, etc.), technical details, and some related risk estimates associated with the revised requirements. Guidance and insight is provided with respect to ATWS analyses techniques, acceptable levels of risk, etc. Many of the hardware modification options evaluated for PWR diverse scram systems, PWR diverse logic for AFW initiation and turbine trip, BWR ARI systems, etc. were recommended/assumed to be compliant with IEEE Std 279 for Protection Systems.

This NUREG (Section IV-4 of Vol 2) is currently cited as Acceptance Criteria for BWR Recirculation Pump Trip in SRP Section 15.8. Information from the NUREG is also included in the Review Procedures for SRP Section 15.8.

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Potential Impact Number: 21795 **SRP Section Number:** 15.8

Document: NUREG 0460 VOL 3
Title: ANTICIPATED TRANSIENTS WITHOUT SCRAM FOR LIGHT WATER REACTORS

Summary:

Anticipated Transients Without Scram for Light Water Reactors. Vols. 1 and 2 April 1978, Vol. 3 December 1978, Vol. 4 March 1980. This is the NRC staff's review of anticipated transient without scram (ATWS). Studies, findings and recommendations are presented. Volumes 1, 2 and 3 contain preliminary studies, findings, resolution options, criteria (where applicable for resolution options), and recommendations pursuant to regulatory action (e.g. orders, rulemaking, regulatory guide development, etc.). Volume 4 contains the actual proposed resolution of unresolved safety issue (TAP A-9) in the form of requirements recommended to be imposed on licensees and applicants (via Commission Orders and via rulemaking, see 10 CFR 50.62 for ultimate results of rulemaking). Volume 4 describes the proposed requirements and their phased implementation, gives the staff's technical basis, and considers the values and impacts. The appendices describe the recently submitted industry information including analyses and the staff's evaluation of that information (concerns with analyses techniques, computer codes, analyses validity as an ATWS event progresses, etc.), technical details, and some related risk estimates associated with the revised requirements. Guidance and insight is provided with respect to ATWS analyses techniques, acceptable levels of risk, etc. Many of the hardware modification options evaluated for PWR diverse scram systems, PWR diverse logic for AFW initiation and turbine trip, BWR ARI systems, etc. were recommended/assumed to be compliant with IEEE Std 279 for Protection Systems.

This NUREG (Section IV-4 of Vol 2) is currently cited as Acceptance Criteria for BWR Recirculation Pump Trip in SRP Section 15.8. Information from the NUREG is also included in the Review Procedures for SRP Section 15.8.

Potential Impact Number: 21796 **SRP Section Number:** 15.8

Document: NUREG 0460 VOL 4
Title: ANTICIPATED TRANSIENTS WITHOUT SCRAM FOR LIGHT WATER REACTORS

APPENDIX II
Potential Impacts

Summary:

Anticipated Transients Without Scram for Light Water Reactors. Vols. 1 and 2 April 1978, Vol. 3 December 1978, Vol. 4 March 1980. This is the NRC staff's review of anticipated transient without scram (ATWS). Studies, findings and recommendations are presented. Volumes 1, 2 and 3 contain preliminary studies, findings, resolution options, criteria (where applicable for resolution options), and recommendations pursuant to regulatory action (e.g. orders, rulemaking, regulatory guide development, etc.). Volume 4 contains the actual proposed resolution of unresolved safety issue (TAP A-9) in the form of requirements recommended to be imposed on licensees and applicants (via Commission Orders and via rulemaking, see 10 CFR 50.62 for ultimate results of rulemaking). Volume 4 describes the proposed requirements and their phased implementation, gives the staff's technical basis, and considers the values and impacts. The appendices describe the recently submitted industry information including analyses and the staff's evaluation of that information (concerns with analyses techniques, computer codes, analyses validity as an ATWS event progresses, etc.), technical details, and some related risk estimates associated with the revised requirements. Guidance and insight is provided with respect to ATWS analyses techniques, acceptable levels of risk, etc. Many of the hardware modification options evaluated for PWR diverse scram systems, PWR diverse logic for AFW initiation and turbine trip, BWR ARI systems, etc. were recommended/assumed to be compliant with IEEE Std 279 for Protection Systems.

This NUREG (Section IV-4 of Vol 2) is currently cited as Acceptance Criteria for BWR Recirculation Pump Trip in SRP Section 15.8. Information from the NUREG is also included in the Review Procedures for SRP Section 15.8.

Potential Impact Number: 21798 **SRP Section Number:** 5.2.4

Document: NRC GENERIC LETTER 88-01 SUP 1
Title: NRC POSITION ON INTERGRANULAR STRESS CORROSION CRACKING (IGSCC) IN BWR AUSTENITIC STAINLESS STEEL PIPING

Summary:

This supplement to Generic Letter (GL) 88-01 provides acceptable alternative positions to some of those delineated in GL 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,". The alternatives are with regard to the inspection of reactor water cleanup (RWCU) system piping outboard of the containment isolation valves and

the leak detection requirements pertaining to the operability of leakage measurement instruments and the frequency of monitoring leakage rates. The supplement also provides clarification or guidance on the staff's positions regarding the sample expansion for Category D welds, the effect of shrinkages resulting from weld overlay repairs or stress improvement (SI) on the piping system and its supports and pipe whip restraints and the technical specification (TS) amendments for incorporating the inservice inspection statement and leak detection requirements as delineated in GL 88-01.

Potential Impact Number: 21799 **SRP Section Number:** 6.6

Document: NRC GENERIC LETTER 88-01 SUP 1
Title: NRC POSITION ON INTERGRANULAR STRESS CORROSION CRACKING (IGSCC) IN BWR AUSTENITIC STAINLESS STEEL PIPING

Summary:

This supplement to Generic Letter (GL) 88-01 provides acceptable alternative positions to some of those delineated in GL 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,". Position (2) establishes alternatives for inspection of reactor water cleanup system piping. Position (4) provides clarification of staff's positions regarding expansion of sample size for Category D welds.

Potential Impact Number: 21815 **SRP Section Number:** 5.4.7

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

Issue: Either human errors, valve leakage, multiple valve failures, or a combination of theses could permit overpressurization of low-pressure systems connected to high-pressure systems. Pipe or component rupture resulting from overpressurization could result in loss of decay heat removal if the rupture was outside of the containment. The staff position regarding intersystem LOCA protection is that future ALWR designs should reduce the possibility of a 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. The "extent practicable" phrase indicates a realization that all systems must eventually interface with atmospheric pressure and that for certain large tanks and heat exchangers it would be difficult or prohibitively expensive to design such systems to the URS equal to full RCS pressure.

The degree of isolation or number of barriers (e.g., three isolation valves) is not sufficient justification for using low-pressure components that can be practically designed to the URS criteria. For example, piping runs should always be designed to meet the URS criteria, as should all associated flanges, connectors, and packings (including valve stem seals, pump seals, heat exchanger tubes, valve bonnets and RCS drain and vent lines). The designer should make every effort to reduce the level of pressure challenge to all systems and subsystems connected to the RCS.

Potential Impact Number: 21820 **SRP Section Number:** 6.3

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

Issue: Either human errors, valve leakage, multiple valve failures, or a combination of theses could permit overpressurization of low-pressure systems connected to high-pressure systems. Pipe or component rupture resulting from overpressurization could result in loss of decay heat removal if the rupture was outside of the containment.

The staff position regarding intersystem LOCA protection is that future ALWR designs should reduce the possibility of a 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. The "extent practicable" phrase indicates a realization that all systems must eventually interface with atmospheric pressure and that for certain large tanks and heat exchangers it would be difficult or prohibitively expensive to design such systems to the URS equal to full RCS pressure.

The degree of isolation or number of barriers (e.g., three isolation valves) is not sufficient justification for using low-pressure components that can be practically designed to the URS criteria. For example, piping runs should always be designed to meet the URS criteria, as should all associated flanges, connectors, and packings

(including valve stem seals, pump seals, heat exchanger tubes, valve bonnets and RCS drain and vent lines). The designer should make every effort to reduce the level of pressure challenge to all systems and subsystems connected to the RCS.

Potential Impact Number: 21821 **SRP Section Number:** 9.3.4

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

Issue: Either human errors, valve leakage, multiple valve failures, or a combination of theses could permit overpressurization of low-pressure systems connected to high-pressure systems. Pipe or component rupture resulting from overpressurization could result in loss of decay heat removal if the rupture was outside of the containment.

The staff position regarding intersystem LOCA protection is that future ALWR designs should reduce the possibility of a 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. The "extent practicable" phrase indicates a realization that all systems must eventually interface with atmospheric pressure and that for certain large tanks and heat exchangers it would be difficult or prohibitively expensive to design such systems to the URS equal to full RCS pressure.

The degree of isolation or number of barriers (e.g., three isolation valves) is not sufficient justification for using low-pressure components that can be practically designed to the URS criteria. For example, piping runs should always be designed to meet the URS criteria, as should all associated flanges, connectors, and packings (including valve stem seals, pump seals, heat exchanger tubes, valve bonnets and RCS drain and vent lines). The designer should make every effort to reduce the level of pressure challenge to all systems and subsystems connected to the RCS.

Potential Impact Number: 21826 **SRP Section Number:** 3.9.3

Document: NRC GENERIC LETTER 91-17 Revision Date: 10/17/91
Title: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

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Summary:

Generic letter 91-17 provides information on the U.S. Nuclear Regulatory Commission's resolution of Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants." The NRC has resolved this issue without developing any new requirements. The bases for resolution are further documented in the enclosed NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants".

NUREG-1339 recommends that a new section of the SRP be prepared to provide guidance to the staff for the review of future plants. It is recognized that NUREG-1339 makes an exception for closure studs which are already addressed in SRP sections 5.3.1 and 5.2.3, however, the staff noted the absence of an SRP section on general reviews of bolting and recommended that one be prepared and issued.

Potential Impact Number: 21828 **SRP Section Number:** 5.4.7

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

The SER for the EPRI evolutionary ALWR contained a review relating to potential loss of decay heat removal when the RCS level is lowered for maintenance during shutdown operations. The NRC staff concluded that the set of conditions contained in chapter 5 section 5.2.3.1.3 of the ALWR Utility Requirements Document were an adequate set of conditions to design the system to meet the criteria for midloop operation contained in SECY 90-016.

Potential Impact Number: 21829 **SRP Section Number:** 3.2.1

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

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Potential Impacts

The seismic interface restraint shall provide a structural barrier between the seismic Category I portion of the main steam lines (MSL) in the reactor building and the non-seismic Category I portions of the MSL in the turbine building and shall be located inside the seismic Category I building. The classification of the MSL in the turbine building as non-seismic Category I is needed for consistency with the classification of the turbine building. Therefore, the staff positions related to the quality and safety guidelines imposed on the MSL from the outermost isolation valve up to the turbine stop valve are equivalent to the staff guidelines in SRP Section 3.2.2, Appendix A to 10 CFR Part 50, and RG 1.29.

Affected main steam system non-seismic Category I components are designed by using appropriate dynamic seismic analyses to withstand the SSE design loads in combination with other appropriate loads, within the limits specified. The mathematical model for the dynamic seismic analyses of the MSL and the branch line piping includes the turbine stop valves and piping to the turbine casing. The dynamic input for the analysis and design of the MSL are derived from a time history analysis (or an equivalent method) of the turbine building as described in SSAR Section 3.7.

The staff noted that, to demonstrate the structural integrity of the main steam piping under SSE loading conditions, a dynamic analysis method should be used for this portion of the main steam piping. In addition, GE should discuss how the turbine building response spectra input to the main steam piping analysis will be generated when no dynamic analysis method of the turbine building is proposed. The staff position as delineated in SRP Section 3.7.2.5 for the generation of floor response spectra (FRS) is that the development of the FRS is acceptable if a time history approach is used. Alternative methods, other than the time history approach, used for generating FRS shall be submitted to the staff for review and approval on a case-by-case basis. The resolution of this issue is discussed in the "Turbine Building" portion of Section 3.7.2 in this report.

Equipment, structures, and piping in the ABWR that are non-seismic Category I but that could damage seismic Category I items if their structural integrity failed, are analyzed and designed to ensure their integrity is maintained under seismic loading from the SSE. At the interface between seismic and non-seismic Category I piping systems, the seismic analysis for the seismic Category I system 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. This commitment is in conformance with RG 1.29. The staff noted that the turbine building and any other applicable structure or equipment should be seismically analyzed for the SSE (RG 1.60 ground response spectrum anchored to 0.3g peak acceleration) to ensure that an earthquake will not adversely affect the structural integrity of the main

steam piping, bypass line, valves, and instruments mounted on these pipes, and the main condenser. GE added SSAR Section 3.7.3.16, which described a static, in lieu of dynamic, analysis for the turbine building. This proposed analysis was in accordance with the Uniform Building Code (UBC) Zone 2A, but was substantially less conservative than the specified SSE for which the main steam and bypass lines should be analyzed. The staff asked GE to provide a clear set of criteria for the turbine building, which is designed to UBC criteria, so that it will neither suffer no loss of function at the specified SSE nor adversely affect safety-significant piping, pipe-mounted equipment, or the condenser itself. The resolution of this issue is discussed in the "Turbine Building" portion of Section 3.7.2 in this report.

Potential Impact Number: 21830 **SRP Section Number:** 3.2.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

For the ABWR, to account for reliance on main steam system and condenser to contain fission products following an accident, the staff has taken the position that main steam piping is subject to in-service inspection, QA requirements, and may be classified as non-seismic Category I if it has been analyzed using a dynamic seismic analysis to SSE loading conditions. The turbine building should also be seismically analyzed for the SSE. The staff requires that the condenser be seismically analyzed to the SSE.

Although the staff has identified its position on this matter in the draft Commission paper issued on February 27, 1992, its approach to resolving this issue has not been reviewed by the Commission and, therefore, does not represent an agency position.

Potential Impact Number: 21832 **SRP Section Number:** 3.7.2

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

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In light of reliance on main steam piping and condenser for containment of fission products following core damage in the event of MSIV leakage, the staff required seismic analysis of main steam piping, turbine building and condenser anchorages. Although GE's use of UBC methods for analysis of the Turbine Building was under review by the staff at the time of the final SER, the staff recommended dynamic analysis of these non-seismic Category I structures and systems.

Potential Impact Number: 21834 **SRP Section Number:** 3.6.2

Document: NRC POLICY STATEMENT 54 FR 18649
Title: POLICY STATEMENT ON ADDITIONAL APPLICATIONS OF LEAK-BEFORE-BREAK TECHNOLOGY

Summary:

Having considered all public comments received, the Commission has decided not to undertake any rulemaking to extend the applicability of LBB to ECCS or EQ at this time. In large part, any safety benefits associated with ECCS can presently be more readily obtained under the recent ECCS rule. The use of exemptions for applying LBB to environmental qualification was permitted in the revision to General Design Criterion 4 (52 FR 41288).

Potential Impact Number: 21835 **SRP Section Number:** 3.6.2

Document: DRAFT SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The application of the leak-before-break (LBB) approach to ALWRs seeking design certification under 10 CFR Part 52 is acceptable when appropriate bounding limits are established during the design certification phase using preliminary analyses results and verified during the combined license phase. Bounding analyses must be performed by the applicants requesting the application of LBB for ALWR piping. NUREG-1061, Volume 3, constitutes the NRC-approved LBB methodology and acceptance criteria.

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Potential Impact Number: 21836 **SRP Section Number:** 3.6.2

Document: DRAFT SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The staff has established certain limitations on the use of the LBB approach for excluding piping that is likely to be susceptible to failure from various degradation mechanisms during service. The LBB approach cannot be applied to piping that can fail in service from such effects as water hammer, creep, erosion, corrosion, erosion-corrosion, fatigue, thermal stratification, and environmental conditions.

Potential Impact Number: 21837 **SRP Section Number:** 3.6.2

Document: DRAFT SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The application of LBB technology eliminates the local dynamic effects of postulated pipe ruptures from the design basis. However, global effects may still be caused by something other than the postulated pipe rupture. Since the global effects from the postulated pipe rupture provide a reasonably conservative design envelope, the NRC staff will continue to require the consideration of global effects for various aspects of the plant design such as EQ, ECCS, and the containment.

Potential Impact Number: 21839 **SRP Section Number:** 3.6.2

Document: DRAFT SER CE80 CH 3

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Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

If the bounding Leak-Before-Break analyses are not based on actual pipe routing, the building subcompartments should be designed to accommodate a postulated pipe break of the LBB candidate piping.

Potential Impact Number: 21858 **SRP Section Number:** 3.7.2

Document: RG 1.29 **Revision:** 3
Title: SEISMIC DESIGN CLASSIFICATION

Summary:

Those portions of structures, whose continued function is not required but whose failure could reduce the functioning of any seismic Category I plant feature to an unacceptable safety level or could result in incapacitating injury to occupants of the control room should be designed and constructed so that the SSE would not cause such failure.

Potential Impact Number: 21876 **SRP Section Number:** 3.7.2

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

The staff required the applicant to address the effects of concrete cracking on seismic analysis of all seismic Category I structures.

Potential Impact Number: 21891 **SRP Section Number:** 3.7.2

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Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

The staff has required the applicant to specify the analysis methods and the design criteria that will be applied to the non-safety-related structures to ensure protection of adjacent safety-related structures.

Potential Impact Number: 21893 **SRP Section Number:** 3.2.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 1

Summary:

A quality assurance program shall be established and implemented in order to provide adequate assurance that structures, systems, and components important to safety will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Potential Impact Number: 21905 **SRP Section Number:** 5.4.7

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

To demonstrate adequate treatment of shutdown risk for ALWRs, the staff will require: 1) adequate vendor assessment of shutdown and low-power risk, identifying design-specific vulnerabilities and weaknesses; 2) documentation showing consideration and incorporation of design features that minimize shutdown and

low-power risk vulnerabilities. The shutdown and Low-power risk issue is currently under review by the staff, as discussed in NUREG-1449. The staff plans to assess current regulatory requirements and, where necessary, will develop and implement appropriate regulatory actions to address the issues, including new guidance and new requirements.

Potential Impact Number: 21916 **SRP Section Number:** 11.2

Document: NRC BULLETIN 80-05
Title: VACUUM CONDITION RESULTING IN DAMAGE TO CHEMICAL VOLUME CONTROL SYSTEM (CVCS) HOLDUP TANKS (SOMETIMES CALLED 'CLEAN WASTE RECEIVER TANKS')

Summary:

Incidents have occurred involving holdup tanks that may have potential for tank ruptures and a release path that bypasses normal containment barriers.

Potential Impact Number: 21917 **SRP Section Number:** 5.4.7

Document: NRC BULLETIN 88-08
Title: THERMAL STRESSES IN PIPING CONNECTED TO REACTOR COOLANT SYSTEMS

Summary:

The purpose of this bulletin is to request that licensees (1) review systems connected to their reactor coolant systems to identify any unisolable piping that could be subjected to temperature distributions which would result in unacceptable thermal stresses and (2) take action, where such piping is identified, to ensure that the piping will not be subjected to unacceptable thermal stresses.

Potential Impact Number: 22050 **SRP Section Number:** 3.7.1

Document: FINAL SER EPRI CH 1

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Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The elimination of the OBE from design would require all current OBE design-related checks to be performed for the SSE. The staff is developing alternatives with the industry to revise the codes and standards when design-related checks are based on the OBE. Resolution of these issues may result in staff recommendations for changes in applicable ASME Code, Section III rules. Therefore, the staff concludes that the elimination of the OBE from design is acceptable. However, the details of how current OBE- related design checks will be performed using the SSE will be resolved between industry and the staff through the appropriate code-related activities or supplemental regulatory guidance. The supplemental regulatory guidance could be in the form of revised SRP sections or the ITAAC (inspections, tests, analyses, and acceptance criteria). The elimination of the OBE from design would require an exemption from the current regulations until the final rulemaking pertaining to Appendix A to 10 CFR Part 100 is approved.

Elimination of the OBE from design would require a change to SRP 3.7.1 as it reviews and accepts OBE and SSE design basis response spectras, time histories and damping values. Final rulemaking on 10 CFR 100 App A could have an affect on SRP 3.7.1.

Potential Impact Number: 22052 **SRP Section Number:** 3.7.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

ASME Code Case N-411-1 is recommended to be used only within the conditions provided in Regulatory Guide 1.84. Code Case N-411-1 provides an alternate method for determining structural damping values to be used in the seismic analysis of specific piping types, under specific design conditions. This item has the potential to apply to SRP Section 3.7.1.

Potential Impact Number: 22053 **SRP Section Number:** 3.7.1

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Document: FINAL SER CE80 CH 2
Title: SITE CHARACTERISTICS

Summary:

10 CFR 100 App A and SRP 2.5.2 define OBE as at least 1/2 the SSE. For the CE80, SSE is anchored at 0.3 g, however, OBE is anchored at 0.1 g. In accordance with 10 CFR 52.48, the staff used the current regulations in 10 CFR Parts 20, 50, 73, and 100 in reviewing ABB-CE's application for design certification. During this review, the staff recognized that if certain regulations were applied to the System 80+ design, the use of those regulations would not serve the underlying purpose of the rule or would not be needed for achieving the fundamental intent of the rule. These exemptions are discussed in the sections of this report listed below.

In the SRM pertaining to SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor Designs," the Commission approved the staff's recommendation to proceed with design-specific rulemakings as part of individual design certifications to resolve selected technical and severe accident issues as "applicable regulations" for the advanced boiling water reactor and System 80+ standard designs. Among these issues are staff positions that deviate from or are not embodied in current regulations applicable to the System 80+ design.

These issues were addressed in various Commission papers, such as SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced LWR designs." The "applicable regulations" that are specific to the System 80+ design are identified and evaluated in this report. The proposed design certification rule will then include these "applicable regulations" for the System 80+ design for the purposes of 10 CFR 52.48, 52.54, 52.59, and 52.63. These "applicable regulations" are discussed in the sections of this report, including Section 3.1.1, "Exemption from operating-basis earthquake design requirement."

Potential Impact Number: 22054 **SRP Section Number:** 3.7.1

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

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The staff evaluated the use of a single-earthquake design (i.e., elimination of operating-basis earthquake (OBE)) for structures systems, and components (SSCs). Originally, implementation of the OBE was included in applicable sections of the ABWR SSAR. After many discussions with the staff, General Electric (GE) stated that it might opt to use the single-earthquake design approach. In a letter to GE dated September 11, 1992, the staff provided preliminary guidance concerning what types of analyses and information would be required in the SSAR for the staff to approve design of SSCs for the ABWR without the OBE. This was Open Item 3.1-1 in the DFSER. The staff's evaluation of this issue is discussed in Chapter 3.1 of the ABWR FSER.

Approval to eliminate the OBE from the design basis would represent a staff position which would require modifying SRP 3.7.1 to modify references to the OBE appropriately.

Potential Impact Number: 22055 **SRP Section Number:** 3.7.1

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The original SSAR specified the SSE as 0.3 g. It also set the design basis OBE as one-third of the SSE (i.e., 0.1 g). For the standard plant design, a more conservative OBE that is equal to one-half of the SSE (i.e., 0.15 g), was employed to evaluate the structural and component responses. Under these conditions, the staff found that specifying the OBE as one-third of the SSE instead of one-half of the SSE was not acceptable.

In the DFSER, the staff's evaluation covered both OBE and SSE. However, as discussed in Section 3.1.1 of the ABWR FSER, by implementing the Commission-approved staff recommendations on OBE elimination, unless otherwise denoted, only the SSE seismic design is evaluated in the ABWR FSER.

Potential Impact Number: 22056 **SRP Section Number:** 4.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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Location: 50.62(c)(3)

Summary:

Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

Potential Impact Number: 22058 **SRP Section Number:** 3.7.1

Document: C&S: ASME N-411 **Revision:** 1 , 2/20/89
Title: ALTERNATIVE DAMPING VALUES FOR RESPONSE SPECTRA ANALYSIS OF CLASSES 1, 2, AND 3 PIPING, SECTION III, DIVISION 1

Summary:

This code case allows damping values which are slightly less restrictive (i.e., slightly higher damping values) than the ones provided in Appendix N to ASME Code Section III, Division 1. The Code Case has not been completely accepted in its entirety, but if certain restrictions, provided in Regulatory Guide 1.84, are applied, Code Case N-411-1 may be used as an alternate source of damping values for seismic analysis of Class 1, 2, and 3 piping sections under Section III, Division 1. Neither this Code Case nor Regulatory Guide 1.84 appear in SRP 3.7.1 Revision 2. The SRP only allows damping values from Regulatory Guide 1.61 or from test data, approved on a case-by-case basis.

Potential Impact Number: 22060 **SRP Section Number:** 2.5.2

Document: DRAFT SER CE80 CH 2
Title: SITE ENVELOPE CHARACTERISTICS

Summary:

10 CFR Part 100 Appendix A and SRP 2.5.2 both require the magnitude of the OBE be at least one-half the SSE. Industry experience has shown such an OBE controls the

design of some safety systems. The staff has agreed in SECY 90-016 Section IV.A that the OBE should not control safety system design.

The CE80 SSE is anchored at 0.30g per RG 1.60, however, the OBE is anchored at 0.10g. CE DSER Section 2.5.2.7 accepts this condition based on SECY-90-016. Specifically, "Since the Commission has approved decoupling of the OBE from the SSE for evolutionary reactors, per SECY-90-016, the proposed OBE for the ABB-CE System 80+ is acceptable."

Potential Impact Number: 22062 **SRP Section Number:** 13.6

Document: FINAL SER EPRI CH 9
Title: EPRI URD-EPD: SITE SUPPORT SYSTEMS

Summary:

In the EPRI Evolutionary Plant FSER, the staff indicates their intention to ensure that final design approval/design certification applicants include sabotage vulnerability analyses that address insider tampering and the external assault threat. The SRP currently does not address sabotage vulnerability analyses.

Potential Impact Number: 22063 **SRP Section Number:** 13.6

Document: FINAL SER EPRI CH 3
Title: EPRI URD-EPD: REACTOR COOLANT SYSTEM AND REACTOR NON-SAFETY AUXILIARY SYSTEMS

Summary:

In the EPRI Evolutionary Plant FSER, the staff indicates the acceptability of the linkage between equipment designated as seismic and equipment designated as vital. This linkage is also noted in Chapter 9, Section 5.2.1 of the EPRI FSER. The SRP currently requires that vital areas be identified, but does not require that the list include all Seismic Category I Equipment.

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Potential Impact Number: 22064 **SRP Section Number:** 13.6

Document: FINAL SER CE80 CH 13
Title: CONDUCT OF OPERATIONS

Summary:

In CESSAR-DC Section 13.6.1(A), ABB-CE requires the physical security system design to conform to the eight regulatory documents listed in Part II of SRP Section 13.6, Revision 2 (NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981); however, one item did not reflect a recent regulatory change. For employee screening requirements, ABB-CE should reference 10 CFR 73.56 and RG 5.66, "Access Authorization Program for Nuclear Power Plants" (June 1991), in lieu of ANSI N18.17. Updating access authorization references in CESSAR-DC Section 13.6.1(A) was designated as Open Item 13.6-1. By CESSAR-DC Amendment Q, ABB-CE changed the reference for authorization accession in CESSAR-DC Section 13.6.1(A) to 10 CFR 73.56 and RG 5.66. These are the appropriate references and DSER Open Item 13.6-1 is resolved.

Potential Impact Number: 22067 **SRP Section Number:** 13.6

Document: RG 5.66 **Revision:** 0 , 6/1/91
Title: ACCESS AUTHORIZATION PROGRAM FOR NUCLEAR POWER PLANTS

Summary:

Provides guidance for meeting the requirements of 10 CFR 73.56, "Personnel Access Authorization Requirements for Nuclear Power Plants." Describes the elements of an acceptable program for granting unescorted access to protected and vital areas of a nuclear power plant.

Potential Impact Number: 22091 **SRP Section Number:** 3.7.1

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

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Summary:

In section 3.9.2.2 , "Seismic Subsystem Analysis," the staff found the use of ASME Code Case N-411-1, as conditioned by Regulatory Guide 1.84, to be acceptable. Specifically:

Regulatory Guide 1.61 contains recommended values of damping to be used in the seismic analysis of structures, systems, and components. In addition, Regulatory Guide 1.84, Revision 25 (May 1988), conditionally endorses ASME Code Case N-411-1. The damping values used by General Electric are the same as those specified in either Regulatory Guide 1.61 or ASME Code Case N-411-1, pursuant to Regulatory Guide 1.84. These criteria are acceptable.

Potential Impact Number: 22092 **SRP Section Number:** 3.7.1

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

References to usage of ASME Code Case N-411-1 are contained in Sections 3.7.1, 3.7.2, and 3.7.3 (via reference to Table 3.7-1). The staff found usage of this Code Case acceptable, as long as a commitment to all of the conditions of Regulatory Guide 1.84 on the use of ASME Code Case N-411-1 was included.

Potential Impact Number: 22096 **SRP Section Number:** 3.7.3

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff noted: "Note f in SSAR Table 3.2-1 and SSAR Sections 3.7.2.8 and 3.7.3.13 state that equipment, structures, and piping in the ABWR that are non-seismic Category I but that could damage seismic Category I items if their structural integrity failed, are analyzed and designed to ensure their integrity is maintained under seismic loading

from the SSE. 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. In the DFSER, the staff noted that this commitment is in conformance with RG 1.29..."

However, SRP 3.7.3.II.8 states: "The attached non-Category I systems, up to the first anchor beyond the interface, should also be designed in such a manner that during an earthquake of SSE intensity it will not cause a failure of the Category I system."

Potential Impact Number: 22097 **SRP Section Number:** 3.7.3

Document: DRAFT SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The staff found in sections 3.2.1, 3.7.2, and 3.7.3 that the boundary between interconnected non-seismic and seismic Category I piping systems was adequately design/defined with two exceptions - both dealing with how the piping was anchored or restrained. Specifically,

"The applicant states that, at the boundary between interconnected non-seismic and seismic Category I piping systems, the seismic Category I boundary extends to the first seismic restraint beyond the boundary. The staff interprets the term 'seismic restraint' to mean that the pipe is restrained in six degrees-of-freedom. In CESSAR Section 3.7.3.13(c), the applicant states that pressure boundary valves are protected by restraining or anchoring the non-seismic portion of the system as required. This is not completely acceptable. While the term 'anchoring', as it is interpreted by the staff is acceptable, it should be clarified by the applicant that the term 'restraining as required' means that the non-seismic portion of the piping system will be restrained in such a manner so as not to invalidate the analysis for the seismic portion of the system. The staff has concluded that, except as noted above, these commitments are in conformance with RG 1.29."

Review note: The CEFSER states that "By DSER Open Item 3.2.1-1, the staff requested that ABB-CE clarify the use of the term 'restrained as required.' Amendment

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N to CESSAR-DC Sections 3.7.3.13(A) and 3.7.3.13(C) stated that the attached non-Category I piping, up to the first anchor beyond the interface, is designed in such a manner that during an earthquake of SSE intensity it will not cause a failure of the Category I piping. These commitments are in conformance with RG 1.29, and thus, acceptable. On this basis, DSER Open Item 3.2.1-1 is resolved."

Potential Impact Number: 22099 **SRP Section Number:** 3.7.3

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

The staff required a commitment to a NUREG not referenced in 3.7.3 in order for the applicants approach for multiple support equipment and components with distinct inputs to be acceptable. Specifically;

"In CESSAR Section 3.7.3.9, the applicant states that the seismic qualification of equipment or components supported at different elevations is based on the envelope of the elevation response spectra or 'multiple support excitation.' The staff asked the applicant to specify that for acceptability, multiple support excitation methods should be in accordance with the staff recommendations on response combinations given in NUREG-1061, Volume 4, Section 2."

Potential Impact Number: 22100 **SRP Section Number:** 3.7.3

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Revision 3 of the Evolutionary Requirements Document, Section 4.4.3.3.4 of Chapter 1 states that the plant designer may use approved independent support motion response spectrum analyses techniques as a basis for seismic design.

The staff's position regarding a definition of "approved techniques" is that this method is only acceptable when used in accordance with the information and recommendations in Sections 2.3 and 2.4 of NUREG-1061, "Report of the U.S. NRC Piping Review Committee," Volume 4. As a part of this position, a support group is defined by supports that have the same time history input. This usually means all supports located on the same floor (or portions of a floor) of a structure. The staff concluded that Sections 4.4.3.3.4 and 4.7.3.4 of Chapter 1 should be revised to provide this commitment.

Potential Impact Number: 22102 **SRP Section Number:** 6.7

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

2.3.1 BWR Main Steamline Isolation Valves and Leakage Control and Classification of Main Steamline of Boiling-Water Reactor

In Section 2.3.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI proposes to eliminate the BWR main steam isolation valve leakage control system and provide an alternative leakage pathway (i.e., the main steamline and the condenser) to the main condenser downstream of the isolation valves in the event of a loss-of-coolant accident (LOCA). This issue is also related to Section 3.3.2 in Appendix B to Chapter 1, Section 3.4.1.5 in Chapter 2, Sections 5.3.3 and 5.4.1.5 in Chapter 3, Section 1.2.3 in Chapter 5, and Section 3 in Chapter 13 of the Evolutionary Requirements Document. In its letters dated May 17 and August 29, 1991, the staff stated its positions on the quality group and seismic classifications of the BWR main/extraction steam system, main steamline, main turbine system, condenser, and related non-seismic structures that resulted from the proposed elimination of the main steam isolation valve leakage control system. The objective of these staff positions is to establish adequate measures to ensure that the above systems and structures will maintain their structural integrity during and following an SSE. Although this issue involves information in Chapters 1, 2, 3, 5, 6, and 13, for the sake of continuity, the basis for the staff's positions and the staff's evaluation of all of EPRI's responses are discussed [in this section].

Potential Impact Number: 22103 **SRP Section Number:** 6.7

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Document: DRAFT FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

GE offers an alternative plan for plateout and holdup of radionuclides following a LOCA and intends not to include an MSIV leakage control system in ABWR designs. Staff presents their review of this alternative in ABWR SER section 15.4.4.2. 'Staff concerns related to the GE proposal were identified as Open Item 142 in the DSER. Subsequently the staff has completed its evaluation and has accepted the BWROG and GE proposals. In order to implement these proposals, the staff has had to deviate from the current guidelines provided in the SRP. The bases for these deviations by the staff are given in [15.4.4.2.1 and 15.4.4.2.2].'

Potential Impact Number: 22104 **SRP Section Number:** 6.7

Document: DRAFT FINAL SER ABWR CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

GE offers an alternative plan for plateout and holdup of radionuclides following a LOCA and intends not to include an MSIV leakage control system in ABWR designs. Staff has identified this issue as COL Action Item 10.3.1-1 and commits to review this proposed method to maintain offsite exposure within 10CFR100 limits on a case by case basis.

Potential Impact Number: 22105 **SRP Section Number:** 3.7.3

Document: NUREG 1061 VOL 4
Title: REPORT OF THE U.S. NRC PIPING REVIEW COMMITTEE - EVALUATION OF OTHER LOADS AND LOAD COMBINATIONS

Summary:

Microfiche 28224:013 - Referenced in the CE80 Chapter 3 Draft SER. The applicant had stated the design would use 'multiple support excitation' and the staff stipulated this was acceptable as long as multiple support excitation methods would be in accordance

with the staff recommendations on response combinations given in NUREG-1061, Volume 4, Section 2.

NUREG-1061, 'Evaluation of Other Loads and Load Combinations' Volume 4 Section 2 deals with 'Staff Recommendations on Response Combinations.' It provides guidance on Independent Support Motion Method, Sequence of Combinations, and High Frequency Modes. This information is relevant to SRP Sections 3.7.3 and 3.9.2.

Potential Impact Number: 22110 **SRP Section Number:** 9.2.1

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

Provision should be made for periodic analyses of water and substrate samples and/or visual inspection of intake structures for biofouling and for its removal. Generic Letter 89-13 provides guidance for resolution of GSI 51, which addressed this problem. Continuous biocide treatment should be specified for service water systems at sites with a potential for macroscopic biofouling. Systems using raw water as a source should make provision for biocide treatment prior to wet layup.

Potential Impact Number: 22111 **SRP Section Number:** 9.2.1

Document: FINAL SER EPRI CH 8
Title: EPRI URD-EPD: PLANT COOLING WATER SYSTEMS

Summary:

The EPRI URD requires that attention be paid to access provided for cleaning out trash racks and traveling screens in order to reduce the potential for bypassing the intrusion detection system at intake areas. The staff concluded that ABB-CE should address this URD requirement. Coordination of the SSWS intake design with the security system design was designated as DSER Open Item 9.2.1-2.

By CESSAR-DC Amendment L, ABB-CE added the following paragraph to amplify the system description to CESSAR-DC Section 9.2.1.1.4:

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"Trash racks shall be provided upstream of the station service water pumps susceptible to damage due to large debris. A safety grade traveling screen shall be installed downstream of the trash racks and upstream of the station service water pumps. Provisions to manually clear debris from trash racks should be coordinated with the design of security barriers and intrusion detection systems so that provisions for physical access do not provide a potential path for covert penetration from the water into the plant's protected area."

Potential Impact Number: 22112 **SRP Section Number:** 9.2.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

An ongoing program of surveillance and control shall be implemented and maintained to reduce fouling problems in open-cycle service water systems. Control of biofouling by the addition of biocides or chemicals should be included in this section, together with surveillance, as the two main elements for reducing fouling. Biocide injection capability for wet layup of service water should also be provided.

Potential Impact Number: 22113 **SRP Section Number:** 9.2.1

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The station service water system (SSWS) is designated as a vital area and all SSWS piping and cabling shall be located within the vital protection area that is common to the main plant.Equipment within the SSWS pump intake structure must be protected as vital equipment. Vital equipment, including cables and piping, must not be accessible outside protected areas.

Potential Impact Number: 22117 **SRP Section Number:** 3.9.6

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Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: 50.55a (f), 50.55a (f), 50.55a (g), 50.55a (b)(2), 57 FR 34666

Summary:

The regulations under 10 CFR 50.55a have been amended (effective 9/8/92, published 8/6/92 at 57 FR 34666) to add paragraph (f) which provides detailed ASME Section XI inservice testing requirements for pumps and valves. The amendment separated requirements for inservice 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.

Potential Impact Number: 22120 **SRP Section Number:** 9.5.1

Document: NRC GENERIC LETTER 92-08
Title: THERMO-LAG 330-1 FIRE BARRIERS

Summary:

The purpose of Generic Letter 92-08 was to require licensees to take action to verify that Thermo-Lag 330-1 fire barrier systems installed in their plants comply with NRC requirements. The NRC staff has found specific problems regarding Thermo-Lag 330-1 barriers which are detailed in this generic letter. The generic letter presented qualification requirements and guidance for fire barriers and detailed why the NRC is concerned that plants using Thermo-Lag 330-1 may not be meeting the requirements of 10 CFR 50.48 and GDCs 3 and 17. This generic letter provided guidance for complying with existing NRC requirements no new staff positions were created.

Potential Impact Number: 22125 **SRP Section Number:** 5.2.1.2

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Document: RG 1.84 **Revision:** 28 , 4/1/92
Title: DESIGN AND FABRICATION CODE CASE ACCEPTABILITY ASME SECTION III DIVISION 1

Summary:

This regulatory guide lists those Section III, Division 1, ASME Code Cases oriented to design and fabrication that are generally acceptable to the NRC staff for implementation in the licensing of light-water-cooled nuclear power plants.

Potential Impact Number: 22132 **SRP Section Number:** 5.2.1.2

Document: RG 1.85 **Revision:** 28 , 4/1/92
Title: MATERIALS CODE CASE ACCEPTABILITY ASME SECTION III DIVISION 1

Summary:

This regulatory guide (revision 28) lists those Section III, Division 1, ASME Code Cases oriented to Materials and Testing that are generally acceptable to the NRC staff for implementation in the licensing of light-water-cooled nuclear power plants.

Potential Impact Number: 22137 **SRP Section Number:** 13.3

Document: RG 1.101 **Revision:** 3 , 8/1/92
Title: EMERGENCY PLANNING AND PREPAREDNESS FOR NUCLEAR POWER REACTORS

Summary:

This regulatory guide provides guidance to licensees and applicants on methods acceptable to the NRC staff for complying with the Commission's regulations for emergency response plans and preparedness at nuclear power reactors.

Potential Impact Number: 22139 **SRP Section Number:** 5.2.1.2

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Document: RG 1.147 **Revision:** 9 , 4/1/92
Title: INSERVICE INSPECTION CODE CASE ACCEPTABILITY ASME SECTION XI DIVISION 1

Summary:

This regulatory guide (revision 9) lists those Section XI, Division 1, ASME Code Cases that are generally acceptable to the NRC staff for implementation in the inservice inspection of light-water-cooled nuclear power plants.

Potential Impact Number: 22141 **SRP Section Number:** 12.5

Document: RG 8.35 **Revision:** 0
Title: PLANNED SPECIAL EXPOSURES

Summary:

This regulatory guide provides guidance on the conditions and prerequisites for permitting Planned Special Exposures (PSEs) allowed by the revision to 10 CFR Part 20, the associated specific monitoring and reporting requirements, and examples of acceptable means of satisfying these requirements. The revised rules in 10 CFR Part 20 allow for 5 rems of occupational exposure and 5 rems of PSE per year, the previous rule using 5(N-18) to establish a limit is no longer applicable.

Potential Impact Number: 22142 **SRP Section Number:** 12.5

Document: RG 8.36 **Revision:** 0
Title: RADIATION DOSE TO THE EMBRYO/FETUS

Summary:

This regulatory guide provides guidance and acceptable methods for calculating the radiation dose to the embryo/fetus of a declared pregnant woman.

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Potential Impact Number: 22143 **SRP Section Number:** 12.5

Document: RG 8.34 **Revision:** 0
Title: MONITORING CRITERIA AND METHODS TO CALCULATE OCCUPATIONAL RADIATION DOSES

Summary:

This regulatory guide provides criteria acceptable to the NRC staff that may be used by licensees to determine when monitoring is required, and it describes methods acceptable to the NRC staff for calculating occupational doses when the intake is known.

Potential Impact Number: 22155 **SRP Section Number:** 9.2.1

Document: FINAL SER EPRI CH 8
Title: EPRI URD-EPD: PLANT COOLING WATER SYSTEMS

Summary:

An ongoing program of surveillance and control shall be implemented and maintained to reduce fouling problems in open-cycle service water systems. Control of biofouling by the addition of biocides or chemicals should be included in this section, together with surveillance, as the two main elements for reducing fouling. Biocide injection capability for wet layup of service water should also be provided.

Potential Impact Number: 22160 **SRP Section Number:** 6.1.1

Document: NRC GENERIC LETTER 88-01 SUP 1
Title: NRC POSITION ON INTERGRANULAR STRESS CORROSION CRACKING (IGSCC) IN BWR AUSTENITIC STAINLESS STEEL PIPING

Summary:

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The supplement provides acceptable alternative staff positions and clarifications to some of those delineated in Generic Letter (GL) 88-01 based upon licensee responses to GL 88-01.

Potential Impact Number: 22161 **SRP Section Number:** 6.1.1

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

During staff review of the ABWR design certification application, the staff determined that to meet the requirements of GDC 4, 14, and 41; the water used in the ESF systems should be controlled to provide assurance that stress corrosion cracking of unstabilized austenitic stainless steel components will not occur. The staff made the following recommendations:

GE should follow Electric Power Research Institute (EPRI) report, "BWR Water Chemistry Guidelines," NP-3589-SR-LD, April 1985, for the water used for emergency core cooling systems and spray systems.

GE should follow EPRI reports, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations: 1987 Revision," NP-5283-SR-A, September 1987, for the design, construction, and operation of hydrogen water chemistry installations.

GE should follow EPRI report, "BWR Hydrogen Water Chemistry Guidelines: 1987 Revision," NP-4947-SR, October 1988, for water chemistry limits, responses to out-of-specification water chemistry, chemical analysis methods, and data management and surveillance and management philosophy required to establish and implement a successful water chemistry control program.

GE responded to this item by revising SSAR Section 9.3.9 to state that the hydrogen water chemistry system will conform to the guidelines in EPRI reports NP-5283-SR-A and NP-4947-SR. This response was acceptable; however, because GE also might use pure water chemistry, the staff concluded that GE should commit to follow EPRI Report NP-3589-SR-LD. GE advised the staff that compliance with EPRI report NP-3589-SR-LD was no longer appropriate because it had committed to use hydrogen water chemistry and was following the guidelines of EPRI report NP-4947-SR. The staff finds this commitment acceptable.

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Potential Impact Number: 22162 **SRP Section Number:** 6.1.1

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

FSER Section 5.2.3:

The materials used for the construction of RCPB components have been identified by specification and are in conformance with Section III of the ASME Code and NUREG-0313, Revision 2. GE's compliance with the provisions of the Code for material specifications and conformance with NUREG-0313, Revision 2, satisfy the quality standards of GDC 1 and 30 and 10 CFR 50.55a.

The RCP materials of construction identified in the SSAR that will be exposed to the reactor coolant have been identified and are compatible with the primary coolant water, which will be chemically controlled to maintain adequate water purity. This compatibility has been proven by extensive testing and satisfactory performance. This includes conformance with the recommendations of RG 1.44, Revision 0, and with the staff guidelines of NUREG-0313, Revision 2. General corrosion of all materials, except unclad carbon and low-alloy steels will be negligible. For unclad carbon and low-alloy steel, GE has provided conservative corrosion allowances for all exposed surfaces in accordance with ASME Code, Section III. This compatibility with the reactor coolant and compliance with the ASME Code satisfy the requirements of GDC 4 as related to the compatibility of components environmental conditions.

The controls to avoid stress corrosion cracking in RCPB components constructed of austenitic stainless steels conform to the recommendations of RGs 1.44 and 1.37, and NUREG-0313, Revision 2. Cold work of austenitic stainless steel is controlled by applying limits on hardness, bend radii, and surface finish on ground surfaces. The controls to be followed during material selection, fabrication, examination, and protection, in accordance with these recommendations, in order to prevent excessive yield strength, sensitization, and contamination provide reasonable assurance that the RCPB components of austenitic stainless steels will be in a metallurgical condition that minimizes susceptibility to stress corrosion cracking during service. These controls meet the requirements of GDC 4 pertaining to the compatibility of components with environmental conditions and those of GDC 14 pertaining to the prevention of leakage and failure of the RCPB.

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FSER Section 5.3.1:

As discussed in SSAR Section 5.3.1, 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 recommendations of RGs 1.37, Revision 0, and 1.44, Revision 0, and NUREG-0313, Revision 2. These controls will provide reasonable assurance (1) that austenitic stainless steel components will be properly cleaned on site, thus satisfying Appendix B to 10 CFR Part 50, and (2) that welded components will not be contaminated or excessively sensitized before or during the welding process, thus satisfying GDC 1, 4, and 30 and 10 CFR 50.55a. This is acceptable.

When components of austenitic stainless steels are welded, ASME Code controls will be supplemented by conformance with the recommendations of RG 1.31, Revision 3, and NUREG-0313, Revision 2. These controls will provide reasonable assurance that the welds will not contain microcracks and will be resistant to IGSCC and satisfy GDC 1, 14, and 30 and 10 CFR 50.55a as related to the integrity of the RCPB.

Potential Impact Number: 22169 **SRP Section Number:** 5.4.2.2

Document: NUREG 1430
Title: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS

Summary:

Standard Technical Specifications for Babcock and Wilcox PWRs. Contains description of, and references to, a steam generator tube ISI program. This NUREG is an improved version of NUREG 0103 STS.

Potential Impact Number: 22170 **SRP Section Number:** 5.4.2.2

Document: NUREG 1431
Title: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS

Summary:

Standard Technical Specifications for Westinghouse PWRs. Contains description of, and references to, a steam generator tube ISI program. This NUREG is an improved version of NUREG 0452 STS.

Potential Impact Number: 22171 **SRP Section Number:** 5.4.2.2

Document: NUREG 1432
Title: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS

Summary:

Standard Technical Specifications for Combustion Engineering PWRs. Contains description of, and references to, a steam generator tube ISI program. This NUREG is an improved version of NUREG 0212 STS.

Potential Impact Number: 22172 **SRP Section Number:** 5.4.2.2

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

A subissue in GSI-135 is the improved eddy current testing of steam generator tubes. The staff deferred this subissue to the development of a revision to RG 1.83, 'Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes.'

DSER Open Item 20.2-18 in the DSER is considered resolved as the COL applicant will be required by the System 80+ Technical Specification Surveillance Requirement 3.4.4.2 to develop an ISI program for steam generator tubes that will be based on eddy current testing techniques that have been demonstrated to have adequate detection and sizing capabilities. Thus, Issue 135 is resolved for the System 80+ design.

Potential Impact Number: 22173 **SRP Section Number:** 5.4.2.2

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Document: FINAL SER EPRI CH 3
Title: EPRI URD-EPD: REACTOR COOLANT SYSTEM AND REACTOR NON-SAFETY AUXILIARY SYSTEMS

Summary:

In Generic Issue (GI) 67.7.0, 'Steam Generator Staff Actions - Improved Eddy Current Tests,' the staff considered eddy current inspection procedures that will prevent steam generator tube ruptures as part of its effort to resolve Unresolved Safety Issues A-3, A-4, and A-5 concerning steam generator tube integrity. This was identified as a confirmatory issue in the DSER for Chapter 3. On the basis of the staff's recommendations, EPRI has instituted requirements for the eddy current inspection procedures in Sections 4.4.1.4 and 4.6.2 of Chapter 3.

Potential Impact Number: 22174 **SRP Section Number:** 5.4.2.2

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

Steam generator tube integrity was designated an unresolved safety issue (USI) in 1978, and Task Action Plans for Issues A-3, A-4, and A-5 were established to evaluate the safety significance of tube degradation in Westinghouse, Combustion Engineering, and Babcock & Wilcox steam generators, respectively. These studies were later combined into one effort because of the similarity of many problems among the PWR vendors.

This issue was resolved, and no new requirements were established because steam generator tube rupture events are not dominant contributors to risk. However, the staff recommended actions in Enclosure 1 of GL-85-02, "Staff Recommended Actions Stemming From NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity." In reviewing the response to GL-85-02, the staff concluded that the large majority of licensees and applicants are following programs, practices, and procedures that are partially to fully consistent with, and/or equivalent to, the staff-recommended actions.

Development of the steam generator ISI program is the responsibility of the COL applicant. The program is plant specific and will be reviewed by the staff individually for

each license application referencing the System 80+ design certification. Therefore, submittal of the ISI program is designated as COL Action Item 5.F-1 (FSER 20.2, Task Action Plan Items, Issue A-4).

Potential Impact Number: 22175 **SRP Section Number:** 6.1.1

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

The COL applicant will perform grinding with resin-bonded or rubber-bonded aluminum oxide or silicon carbide wheels that have been used only on Type 300-series stainless alloys. Grinding wheels bonded with rubber compositions that include halides or sulfur will not be used on austenitic stainless steels.

The staff apparently found these COL Actions to be adequate.

Potential Impact Number: 22176 **SRP Section Number:** 6.1.1

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In CESSAR-DC Table 6.1-1, ABB-CE indicates that Inconel 690 will be used in place of Inconel 600 as an ESF pressure-retaining material. Operating experience indicates that Inconel 600 is susceptible to cracking as discussed at a 1987 workshop on secondary-side intergranular corrosion mechanisms and documented in EPRI NP-5971. ABB-CE has considered using alternate materials that resist stress corrosion cracking. ABB-CE will use Inconel 690 wherever Inconel 600 was to be used. The staff views the Inconel 690 alloy as the preferred nickel alloy in the primary and secondary coolant loops because it is more resistant to corrosion than Inconel 600. The use of Inconel 690 will provide reasonable assurance of the material integrity of the components and tubing in contact with reactor coolant and secondary water chemistries thus meeting the EPRI guidelines committed to by ABB-CE.

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ASTM STP 1074 presents general corrosion results for Alloy 690 and Alloy 600 in flowing (5.5 m/sec) ammoniated and borated water at 316 C (600 F). This study shows a significantly improved general corrosion resistance for Alloy 690 over Alloy 600. Under the test conditions, general corrosion for Alloy 690 was one fourth to one tenth that of Alloy 600. Thus, a conservative corrosion allowance for Alloy 690 is 0.000025 cm/year (0.01 mils/yr) of operation.

On the secondary side, outside surfaces of the tubes (Alloy 690) are exposed to the secondary environment. The same corrosion allowances used for the primary-side applications are appropriate for the secondary side for Alloy 690.

Potential Impact Number: 22180 **SRP Section Number:** 6.1.1

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In the DSER, the staff indicated that the applicant should include a corrosion allowance for a 60-year plant design life and its technical basis.

ABB-CE has provided a corrosion allowance for a 60-year plant design life and its technical basis as follows: The 60-year minimum corrosion allowance for the materials exposed to process fluids are:

0.00015 cm (0.0006 in) for Inconel 690
0.0076 cm (0.003 in) for austenitic stainless steel
0.015 cm (0.006 in) for ferritic stainless steel
0.15 cm (0.060 in) for unclad carbon steel

Among the materials of construction for the primary-side components are a variety of nickel alloys, stainless steels, and low alloy steels clad with corrosion resistant material (i.e., nickel alloy or stainless steel). All of these materials are resistant to general corrosion in high temperature aqueous environments, and numerous studies have documented this resistance (EPRI URD and NUREG-0313). For example, 300 series stainless steels, low-chromium stainless steels (Types 405 and 409), and Alloy 600 (a nickel alloy used for steam generator (SG) tubes and penetrations in older pressurized-water reactors (PWRs) have been exposed to low-level faulted secondary-side environments for up to 249 days. After exposure, the following general

corrosion rates were determined: for Type 347 stainless steel, 0.000033 - 0.00012 cm/year (0.013 - 0.048 mils/yr); for Type 405/409 stainless steel, 0.00007 - 0.00015 cm/year (0.028 - 0.061 mils/yr); and for Alloy 600, 0.000015 - 0.000058 cm/year (0.006 - 0.023 mils/yr). ASTM STP 1074 presents general corrosion results for Alloy 690 and Alloy 600 in flowing (5.5 m/sec) ammoniated and borated water at 316 C (600 F). This study shows a significantly improved general corrosion resistance for Alloy 690 over Alloy 600. Under the test conditions, general corrosion for Alloy 690 was one fourth to one tenth that of Alloy 600. Thus, a conservative corrosion allowance for Alloy 690 is 0.000025 cm/year (0.01 mils/yr) of operation. For austenitic stainless steel in the primary circuit, a corrosion allowance of 0.00013 cm/year (0.05 mils/yr) is appropriate; and for ferritic stainless steels, a corrosion allowance of 0.00025 cm/year (0.1 mils/yr) is appropriate.

On the secondary side, unclad carbon and low-alloy steels used for the steam generator shells are exposed to the secondary environment, as are the outside surfaces of the tubes (Alloy 690), the tube supports (Type 409 stainless steel), and the flow distribution plate (Type 405 stainless steel). The same corrosion allowances used for the primary-side applications are appropriate for the secondary side for Alloy 690 and the Types 405/409 stainless steels. References (1) and (2), along with other sources, show relatively low corrosion rates for carbon and low-alloy steels. For Type 1010 carbon steel, corrosion rates after up to 440 days of exposure to low-level faulted conditions ranged from 0.00034 - 0.002 cm/year (0.135 - 0.77 mils/yr). For SA 508 Class 3 steel, the corrosion rates for the same test period varied from 0.00031 - 0.0017 cm/year (0.122 - 0.650 mils/yr). Thus, for secondary-side applications, a corrosion allowance of 0.0025 cm/year (1.0 mils/year) of operation is reasonable, and an overall corrosion allowance of 0.15 cm (0.060 in.) will provide sufficient material for 60 years of operation.

Potential Impact Number: 22181 **SRP Section Number:** 6.1.1

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

Recent fatigue test data indicates that the effects of the environment could significantly reduce the fatigue resistance of materials (see the following references). (1) K. Iida, J. Fukakura, M. Higuchi, H. Kobayashi, S. Miyazono, and M. Nakao, "Survey of Fatigue Strength Data of Nuclear Structural Materials in Japan," Abstract of DBA Committee

Report, 1988. (Presented to the ASME, Subgroup on Fatigue Strength, on December 5, 1988, in New York City, NY.) (Enclosure in letter, from John W. Craig of NRC to Edward Griffing of the Nuclear Management and Resources Council, Washington, D.C., dated July 2, 1991.). (2) M. Higuchi and K. Iida, "Fatigue Strength Correction Factors of Carbon and Low-Alloy Steels in Oxygen-Containing High-Temperature Water," Nuclear Engineering and Design, Volume 129, 1992, pp. 293-306. (3) J.B. Terrell, "Effect of Cyclic Frequency on the Fatigue Life of ASME SA-106-C Piping Steel in PWR Environments," Journal of Materials Engineering, Volume 10, Number 3, 1988, pp. 193-203). The specific concerns relate to the water and temperature environment and its synergistic interactions with the strain rate.

The design of System 80+ components will address the potential influence of environmental effects on the fatigue life of materials over the 60-year design life. The issue of environmental effects on fatigue is currently under consideration by a special Steering Committee for Cyclic Life and Environmental Effects in Nuclear Applications of the Pressure Vessel Research Council (PVRC). These activities were initiated based on requests from the ASME Boiler and Pressure Vessel (B&PV) Code Committee and the Board on Nuclear Codes and Standards (BNCS). The charter of the PVRC Steering Committee is to provide guidance and directions related to determining the effects of light water reactor (LWR) service environments on the cyclic life properties of applicable materials.

The Steering Committee is also evaluating application methodologies that include these effects in the fatigue analysis process. Preliminary recommendations were provided to the BNCS in September 1992. The initial findings reported to BNCS were that the current S/N curves should be appropriate for PWR environments. There was not a complete agreement of the Steering Committee on this position and the issue is not yet finally resolved. ABB-CE will continue to monitor the industry activities on the fatigue curves and fatigue analysis methodology.

System 80+ components will be designed to ASME B&PV Code rules. If the influence of environmental effects has not been incorporated into the Code rules at the time of the design, the potential effects will be addressed based on the technical understanding of the materials data and anticipated operating conditions.

Potential Impact Number: 22182 **SRP Section Number:** 6.1.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

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Summary:

The staff considers Alloy 600 undesirable for use in the ALWR because of the stress corrosion cracking experienced in existing nuclear plants. It specifically discourages the use of Alloy 600 in steam generators and pressurizers. Therefore, it will require that the applicant for any standard design application identify the use of Alloy 600 and provide information concerning its use. Those applications will be reviewed and approved by the staff on a case-by-case basis. In addition, the use of other Ni-Cr-Fe alloys such as Alloy 690 or 800 should be considered in applications for which primary water stress corrosion cracking is a concern. These applications also will be reviewed on a case-by-case basis.

Also, where Iconel Alloy-X-750 is used, the application will be in accordance with EPRI NP-6202, 'Material specification for Alloy-X-750 in LWR Internal Components. The staff considers these requirements acceptable because improperly heat treated Alloy-X-750 is susceptible to cracking and EPRI NP- 6202 provides the technical basis for the use of this alloy.

Potential Impact Number: 22183 **SRP Section Number:** 6.1.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The staff considers previously accepted standard corrosion allowances to be inadequate. The plant designer should consider all types of corrosion (i.e., microbiological, pitting, and crevice) in the design. The staff concluded that the plant designer should control corrosion mechanisms, such as general, pitting, crevice, and microbiological corrosion, in the design of piping systems of primary and secondary systems. The general corrosion allowance should comply with the corrosion allowance specified in Section III of the ASME Code and ANSI/ASME B.31.1, Power Piping. For the specific corrosion allowance (such as that for microbiological corrosion), the plant designer may use industry methodologies such as the EPRI computer code CHECMATE. However, the staff will have to review and approve any industry methodology before it is used in the ALWR design. The staff will evaluate this matter during its review of an individual application for FDA/DC.

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Potential Impact Number: 22184 **SRP Section Number:** 6.1.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The staff has recommended that licensees and applicants follow Revision 2 of NUREG-0313, Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping, to prevent IGSCC in stainless steel. In addition, it is important that adequate field and shop fabrication processes be used to minimize the sensitization of materials to IGSCC. Therefore, the staff will evaluate this matter during its review of an individual application for a combined license.

Potential Impact Number: 22185 **SRP Section Number:** 6.1.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The EPRI proposed water chemistry design basis requirements for BWR plant systems in accordance with EPRI NP-4947-SR, BWR Hydrogen Water Chemistry (HWC) Guidelines, 1987 Revision, and its subsequent revisions, as supplemented by added guidelines in Table 1.2 of the FSER were acceptable to the staff. The FSER also discussed problems associated with HWC control which have been identified through recent operating experience. The staff took no position with respect the identified problems.

Potential Impact Number: 22186 **SRP Section Number:** 6.1.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

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The EPRI proposed water chemistry design basis requirements for PWR plant systems in accordance with EPRI NP-5960, PWR Primary Water Chemistry Guidelines, Revision 1, and its subsequent revisions; EPRI NP-6239, PWR Secondary Water Chemistry Guidelines, Revision 2, and its subsequent revisions; as supplemented by the guidelines in Table 1.3 of the FSER are acceptable. The applicant for FDA/DC or a combined license should reference EPRI NP-7077, PWR Primary Water Chemistry Guidelines: Revision 2, November 1990 instead of EPRI NP-5960, Revision 1. The staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license.

Potential Impact Number: 22187 **SRP Section Number:** 6.1.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: 50.44, 50.44, 50.44(a)

Summary:

Provides requirements for the control of combustible gasses in light water cooled reactors including a general requirement in paragraph (a) that each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy or ZIRLO cladding, must include means for control of hydrogen gas that may be generated, following a postulated loss-of-coolant accident (LOCA), by metal-water reaction involving the fuel cladding and the reactor coolant, radiolytic decomposition of the reactor coolant, and corrosion of metals.

SRP Section 6.1.1 currently includes review of hydrogen control, which explicitly considers the hydrogen generated by corrosion of metals in containment by containment spray systems. Metal-water reaction between ECCS fluids and fuel cladding and radiolytic decomposition of ECCS fluids are not explicitly addressed in SRP Section 6.1.1. GDC 41 is currently cited as Acceptance Criteria for such reviews although GDC 41 provides design criteria for containment atmosphere cleanup systems. 10 CFR 50.44 provides requirements applicable to reviews of means to control hydrogen gas which may be generated following a LOCA. Such means should include minimization of gas generated as a result of material and ESF fluid properties (See Reg. Guide 1.7 Position C.6).

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Potential Impact Number: 22192 **SRP Section Number:** 3.9.2

Document: RG 1.84 **Revision:** 28 , 4/1/92
Title: DESIGN AND FABRICATION CODE CASE ACCEPTABILITY ASME SECTION III DIVISION 1

Summary:

Allows conditional use of Design-oriented Code Case N-411-1, 'Alternate Damping Values for Response Spectra Analysis of Classes 1, 2, and 3 Piping, Section III, Division 1,' 02-20-89.

The following P.I.s were originally assigned to this section. This P.I. and 22193 were used to replace these: 3065, 3068, 3071, 3079, 3081, 3087, 3089, 3096, 3098, 3104, 3105, 3127, 3128, 3131, 3134, 3135, 3138, 3140, 3143, 3144, 3146, 3147, 3148, 3191, 3193, 3195, 3197, 3199, 3200, 3201, 3203, 3205, 3206, 3209, 3211, 3212, 3213, 3214, 3215, 3216, 3217, 3219, 3220, 3222, 3223, 3225, 3410, 3412, 3414, 3416, 3417, 3419, 3420, 3421, 3423, 3424, 3425, 3426, 3427, 3429, 3430, 3431, 3432, 3433, 3434, 3436, 3437, 3557, 3559, 3569.

Potential Impact Number: 22199 **SRP Section Number:** 5.4.7

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

GSI 99 addressed a safety concern that decay heat removal could be lost in PWRs operating in reduced inventory. Generic Letter 88-17 provided the resolution of this issue.

Potential Impact Number: 22200 **SRP Section Number:** 3.9.6

Document: FINAL SER EPRI CH 8
Title: EPRI URD-EPD: PLANT COOLING WATER SYSTEMS

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Summary:

In its staff requirements memorandum (SRM) of June 26, 1990, on SECY-90-016, the Commission approved a staff position on testing of pumps and valves for evolutionary LWRs. The SRM stated that the following provisions should apply to all safety-related pumps and valves and not be limited to ASME Code Class 1, 2, and 3 components:

- Piping design should incorporate provisions for full-flow testing (maximum design flow) of pumps and check valves.

- Designs should incorporate provisions for testing motor-operated valves under design-basis differential pressure.

- Check valve testing should incorporate the use of advanced nonintrusive techniques to address degradation and performance characteristics.

- A program should be established to determine the frequency necessary for the disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced nonintrusive techniques.

These items should be considered applicable to all safety-related power-operated valves. In addition, the staff agreed with the recommendation of the ACRS that the guidelines of GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," be applied to evolutionary LWR designs. The ACRS also recommended that the staff resolve the check-valve testing and surveillance issue and that consideration be given to industry-proposed alternative ways of meeting inservice testing (IST) and surveillance requirements.

Potential Impact Number: 22202 **SRP Section Number:** 3.9.6

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

During its review of Chapter 8 of the Evolutionary Requirements Document, the staff requested that EPRI propose a frequency of disassembly and inspection of safety-related check valves and motor-operated valves. EPRI stated that the frequency of valve inspections was beyond the scope of the Evolutionary Requirements

Document. As discussed in Section 3.2 of the DSER for Chapter 8, the staff concluded that EPRI's response was not acceptable. The staff requested, as a minimum, a commitment to develop a program that will establish the frequency and extent of disassembly and inspection of safety-related valves, including the basis for the frequency and extent of each disassembly. In Chapter 1 of the Evolutionary Requirements Document, EPRI provided a commitment that is consistent with the staff request.

EPRI provided a requirement that where in situ design-basis testing is not practical, MOVs with well-understood performance characteristics will be used in the design of evolutionary plants. Also, EPRI requirements for preinstallation testing include full and partial design-basis conditions. Justification will be required to show that qualification tests apply to all other valves of the same type, size, and conditions. The staff concludes that the above commitments are consistent with staff positions relative to this issue and are, therefore, acceptable.

EPRI stated that nonactive MOVs should be designed for potential mispositioning, but qualification testing was not required. This statement does not completely agree with the current staff guidelines on valve mispositioning. The staff concludes that nonactive MOVs in a safety-related system either should be designed to prevent mispositioning or should be required to be subjected to qualification testing to demonstrate their capability to recover from mispositioning. Mispositioning may occur through actions taken at any time locally (manual or electrical), at a motor control center or in the control room, and includes deliberate changes of valve position for performing surveillance testing. Therefore, the staff will review individual applications for FDA/DC in accordance with the above position.

EPRI should require that provisions be made for measuring both stem thrust and actuator torque because of the importance of information regarding the conversion of torque to thrust (i.e., stem factor). Regarding the capability of isolation valves to close against conditions that may exist during events requiring containment isolation, the isolation valves should be designed and test-qualified to be able to isolate flow resulting from a pipe break at the worst-case differential pressure (e.g., a condition resulting from a failure to scram the reactor in a timely manner), because the potential for a break in a line from the reactor vessel would likely be greatest when the reactor pressure was abnormally high. The staff will review individual applications for FDA/DC in accordance with the above positions.

Potential Impact Number: 22206 **SRP Section Number:** 4.5.1

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Document: C&S: ASTM A262
Title: DETECTING SUSCEPTIBILITY TO INTERGRANULAR ATTACK IN STAINLESS STEEL

Summary:

The Standard Review Plan (1981) in Section 4.5.1 cites ASTM A-262 as endorsed by Regulatory Guide 1.44 for verification of non- sensitization of stainless steel and qualification of welding process employed in production.

Potential Impact Number: 22218 **SRP Section Number:** 3.9.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

This section describes the Piping Preoperational Vibration and Dynamic Effects Testing of (1) ASME Code, Class 1, 2, and 3 piping systems; (2) high-energy piping systems inside seismic Category I structures; (3) high-energy portions of systems whose failure could reduce the functioning of seismic Category I plant features to an unacceptable safety level; and (4) seismic Category I portions of moderate-energy piping systems located outside the containment.

GE's planned detailed test specifications will be prepared in accordance ANSI/OM-1987, Part 3, and ANSI/OM-1986, Part 7. The staff found that this approach conforms to the applicable guidelines of SRP Section 3.9.2 and is acceptable.

Potential Impact Number: 22221 **SRP Section Number:** 4.5.1

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

Cleaning and cleanliness control are in accordance with American National Standards Institute (ANSI)/ASME NQA-2-1983, "Quality Assurance Requirements for Nuclear

Power Plants," and RG 1.37, "Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." The staff has previously reviewed ANSI/ASME NQA-2-1983 and finds it acceptable. The cleaning and cleanliness control specified will adequately control contamination of components during fabrication, shipment, and storage.

NQA-2-1983 is a departure from ANSI N45.2.1-1973 used in the current SRP 4.5.1 (1981)

Potential Impact Number: 22222 **SRP Section Number:** 4.5.1

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

In CESSAR-DC Section 4.5.1.1, ABB-CE states that Inconel 690 is to be used in lieu of Inconel 600 materials for 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 it is more corrosion-resistant than Inconel 600. Thus, the use of Inconel 690 and its equivalent weld metals (Types 52 and 152) will give reasonable assurance of the material integrity of the components and tubing in contact with reactor coolant and most secondary water chemistries because it is more corrosion resistant. On this basis, DSER Open Item 4.5.1-1 is resolved.

Potential Impact Number: 22223 **SRP Section Number:** 4.5.1

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

ABB-CE proposes to use Stellite, a cobalt-based alloy, for pins and latches in the CEDM. Radioactivation of cobalt is a concern in current nuclear plants. Cobalt application should be avoided in the System 80+ design to keep radiation doses as low as reasonably achievable. In CESSAR-DC Section 5.2.3.2.2, "Materials of Construction Compatibility with Reactor Coolant," ABB-CE states that cobalt-based

alloys will be avoided except where no proven alternative exists. Cobalt-free alloys with the wear and corrosion properties of the Stellite (cobalt-base) type alloys are being developed but have not been fully demonstrated to be as durable as the Satellites. Accordingly, the staff acknowledges that the availability of alternative materials for Satellites is limited at this time. Therefore, DSER Open Item 4.5.1-3 is resolved.

The current SRP 4.5.1 (1981) in Section I.1 lists Stellites as materials commonly used in the control rod drive.

Potential Impact Number: 22224 **SRP Section Number:** 4.5.1

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

The applicant is proposing to use Types 304 and 316 austenitic stainless steel. However, these materials are susceptible to intergranular stress corrosion cracking. In the DSER, the staff indicated that the applicant should consider using low-carbon, wrought austenitic stainless steel, which includes Types 304L, 316L, 304NG, 316NG, and modified Type 347.

ABB-CE will use Type 304 and 316 austenitic stainless steel. However, these materials are susceptible to intergranular stress corrosion cracking (IGSCC) when the oxygen content of the reactor coolant exceeds 0.010 ppm at temperatures above 200 F during normal operations. During startup and operation of the System 80+ plant, these temperature and chemical conditions are maintained through specified chemistry control. As allowed in RG 1.44, ABB-CE has taken alternative mitigating approaches, giving reasonable assurance of the integrity of austenitic stainless steel components in contact with reactor coolant. On this basis, DSER Open Item 4.5.1-4 is resolved.

The current SRP 4.5.1 (1981) does not recommend specific grades of austenitic stainless steels or limits on carbon content.

Potential Impact Number: 22225 **SRP Section Number:** 4.5.1

Document: FINAL SER CE80 CH 4

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Title: REACTOR

Summary:

The ferrite content limits for austenitic stainless weld metal in the CESSAR-DC were broader than those in industry guidelines (EPRI, "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, ALWR Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991) and staff guidance (USNRC, "Technical Report on Materials Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," NUREG-0313, Revision 2, January 1988). In CESSAR-DC Amendment L, ABB-CE revised the CESSAR-DC to be consistent with industry guidelines and staff guidance. The revision should assure an acceptable level of structural integrity for stainless steel welds over the life of the plant. On this basis, DSER Open Item 4.5.1-5 is resolved.

Potential Impact Number: 22229 **SRP Section Number:** 3.9.2

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staffs review of Seismic Subsystem Analysis (3.9.2.2) was performed in accordance with SRP Section 3.9.2 and consisted of an evaluation of SSAR Section 3.7.3.

Potential Impact Number: 22230 **SRP Section Number:** 3.9.2

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff clarified that the response analysis, for a piping system supported at points with different dynamic excitations, may use the multiple-support excitation analysis method as an alternative to the enveloped response spectrum method. The staff did stipulate that when the former method is used, that the responses resulting from

motions of supports between two or more different support groups may be combined by the SRSS method if a support group is defined by supports that have the same time history input. This usually means all supports located on the same floor, or portions of a floor in a structure.

The staff stipulated (in Chapter 14) that for this alternative to the enveloped response spectrum method to be acceptable it should be implemented in accordance with the information and recommendations in Sections 2.3 and 2.4 of NUREG-1061, Volume 4.

Potential Impact Number: 22231 **SRP Section Number:** 3.9.2

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff noted that RG 1.61 (already referenced in SRP 3.9.2) and RG 1.84's conditional endorsement of ASME Code Case N-411-1 (not presently referenced in SRP 3.9.2) contain acceptable values of damping to be used in the seismic analysis of structures, systems, and components.

Potential Impact Number: 22232 **SRP Section Number:** 3.9.2

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff reviewed the issue of modal damping for composite structures under this section (3.9.2.2). Specifically, they found some damping values for the piping supports (e.g., snubbers and struts) were higher than the damping values tabulated for the piping. GE indicated that these values were presented because modal damping for composite structures could be used in a response spectrum analysis as an option (NOTE: composite model damping can result in higher damping values than specified in RG 1.61 and RG 1.84's conditional endorsement of ASME Code Case N-411-1.)

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The staff stated that if GE plans to use the modal damping for composite structures as an option for piping analysis, then a description and justification of the approach must be provided in the SSAR for staff review and approval before its use. (NOTE: analysis procedures for damping are discussed in 3.7.2 & 3.7.3, and percentage of critical damping values are provided in 3.7.1. SRP Section 3.9.2's, 'Criteria Used for Damping' closely parallels 3.7.1 as a source of acceptable damping values, but does not provided guidance for analysis procedures (i.e., composite modal damping).)

Potential Impact Number: 22233 **SRP Section Number:** 3.9.2

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

This section describes the Piping Vibrations, Thermal Expansion, and Dynamic Effects Testing.

The staff specifically asked the applicant to commit to conduct the above program in accordance with ANSI/ASME OM-3 and draft OM-7.

Potential Impact Number: 22234 **SRP Section Number:** 3.9.2

Document: DRAFT SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The staff review of seismic subsystem analysis (3.9.2.2) was conducted in accordance with the requirements of SRP Section 3.7.3 which the staff noted are essentially the same as those of SRP Section 3.9.2.

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Potential Impact Number: 22235 **SRP Section Number:** 3.9.2

Document: DRAFT SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

In the CESSAR Section 3.9.2.5, the applicant accounted for the effects of pipes not eliminated by LBB methodologies by increasing the SSE load by 10 percent. The staff observed that the application of the LBB methodology has not been approved by the NRC for the System 80+ design.

The staff found that the applicant should commit to evaluate the faulted condition for the reactor internals and the unbroken loops of the reactor coolant piping in accordance with the criteria of SRP Section 3.9.2, Revision 2, Item II.5. Specifically, the effects of postulated ruptures in accordance with SRP Section 3.6.2, Revision 2, were to be included in the analysis, pending NRC approval of the application of the LBB methodology to the System 80+ design.

The staff noted if LBB is approved, that elimination of pipe rupture effects allowed under LBB was acceptable, but that the applicant should include the effects of pipe ruptures in branch lines in the faulted condition approval, or should provide justification that 10 percent of the SSE effects provides a conservative upper bound for the branch line rupture effects on the reactor internals and the unbroken loops of the reactor coolant piping.

In Section 3.6.3 the staff concluded that the application of the LBB approach to ALWRs seeking design certification under 10 CFR Part 52 is acceptable when appropriate bounding limits are established during the design certification phase using preliminary analyses results and verified during the combined license phase. Bounding analyses must be performed by the applicant for ALWR piping requesting the application of LBB. NUREG-1061, Volume 3, constitutes the NRC-approved LBB methodology and acceptance criteria.

Potential Impact Number: 22236 **SRP Section Number:** 3.9.2

Document: DRAFT SER CE80 CH 3

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Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

In PI 22234 it was noted the staff review of seismic subsystem analysis (3.9.2.2) was conducted in accordance with the requirements of SRP Section 3.7.3. SRP Section 3.7.3, Item II.8 and SRP Section 3.9.2, Item II.2.k. both deal with 'Interaction of other Piping with Category I Piping.' The following information written against 3.7.3 (PI 22097) also applies to the corresponding Item of 3.9.2.

The staff found in sections 3.2.1, 3.7.2, and 3.7.3 that the boundary between interconnected non-seismic and seismic Category I piping systems was adequately design/defined with two exceptions - both dealing with how the piping was anchored or restrained. Specifically,

'The applicant states that, at the boundary between interconnected non-seismic and seismic Category I piping systems, the seismic Category I boundary extends to the first seismic restraint beyond the boundary. The staff interprets the term 'seismic restraint' to mean that the pipe is restrained in six degrees-of-freedom. In CESSAR Section 3.7.3.13(c), the applicant states that pressure boundary valves are protected by restraining or anchoring the non-seismic portion of the system as required. This is not completely acceptable. While the term 'anchoring', as it is interpreted by the staff is acceptable, it should be clarified by the applicant that the term 'restraining as required' means that the non-seismic portion of the piping system will be restrained in such a manner so as not to invalidate the analysis for the seismic portion of the system. The staff has concluded that, except as noted above, these commitments are in conformance with RG 1.29.'

Potential Impact Number: 22237 **SRP Section Number:** 3.9.2

Document: DRAFT SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

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In PI 22234 it was noted the staff review of seismic subsystem analysis (3.9.2.2) was conducted in accordance with the requirements of SRP Section 3.7.3. SRP Section 3.7.3, Item II.9 and SRP Section 3.9.2, Item II.2.g. both deal with 'Multiply-Supported Equipment and Components With Distinct Inputs.' The following information written against 3.7.3 (PI 22099) also applies to the corresponding Item of 3.9.2.

The staff required a commitment to a NUREG not referenced in 3.7.3 in order for the applicants approach for multiple support equipment and components with distinct inputs to be acceptable. Specifically;

'In CESSAR Section 3.7.3.9, the applicant states that the seismic qualification of equipment or components supported at different elevations is based on the envelope of the elevation response spectra or 'multiple support excitation.' The staff asked the applicant to specify that for acceptability, multiple support excitation methods should be in accordance with the staff recommendations on response combinations given in NUREG-1061, Volume 4, Section 2.'

Potential Impact Number: 22238 **SRP Section Number:** 3.9.2

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

In PI 22229 it was noted the staff review of seismic subsystem analysis (3.9.2.2) was performed in accordance with SRP Section 3.7.3. SRP Section 3.7.3, Item II.8 and SRP Section 3.9.2, Item II.2.k. both deal with 'Interaction of other Piping with Category I Piping.' The following information written against 3.7.3 (PI 22096) also applies to the corresponding Item of 3.9.2.

The staff noted: 'SSAR Sections 3.7.2.8 and 3.7.3.13 states that equipment, structures, and piping in the ABWR that are non-seismic Category I but that could damage seismic Category I items if their structural integrity failed, are analyzed and designed to ensure their integrity is maintained under seismic loading from the SSE. 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. The staff has concluded that this commitment is in conformance with RG 1.29.'

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Potential Impact Number: 22242 **SRP Section Number:** 3.9.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

EPRI states that the plant designer may use approved independent support motion response spectrum analyses techniques as a basis for seismic design and identifies this use as an exception to SRP Section 3.9.2.

The staff found this to be acceptable only when 'approved techniques' is in accordance with the information and recommendations in Sections 2.3 and 2.4 of NUREG-1061, 'Report of the U.S. NRC Piping Review Committee,' Volume 4. And a support group is defined by supports that have the same time history input. This usually means all supports located on the same floor (or portions of a floor) of a structure.

Potential Impact Number: 22248 **SRP Section Number:** 3.9.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

ASME Code Case N-411 may only be used as conditioned in RG 1.84. N-411 provides an alternate method of arriving at damping values for specific piping types under specific design conditions.

Potential Impact Number: 22249 **SRP Section Number:** 6.1.1

Document: C&S: ASTM A262
Title: DETECTING SUSCEPTIBILITY TO INTERGRANULAR ATTACK IN STAINLESS STEEL

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Summary:

ASTM A262 is cited in the Review Procedures of SRP Section 6.1.1, subsection III.A in conjunction with Reg. Guide 1.44 (which endorses the 1970 version, Practices A or E, with exceptions). The citation in the SRP is not date specific and is thus assumed to refer to the 1970 version. Reg. Guide 1.37 is also cited in SRP Section 6.1.1 for component/system cleaning. Reg. Guide 1.37 endorses the 1968 version of ASTM A262. The current version of this standard is ASTM A262-1993.

Potential Impact Number: 22250 **SRP Section Number:** 6.1.1

Document: C&S: AWS D1.1
Title: STRUCTURAL WELDING CODE

Summary:

AWS D1.1 is cited in SRP Section 6.1.1, subsection II.A.1.b.2 as acceptance criteria for ferritic steel welding with respect to moisture control on low hydrogen welding materials. The citation is not date specific and it is thus assumed that the version in effect at the time of issuance of SRP Section 6.1.1 (July 1981) is the version of this code that the SRP references.The current version of this standard is AWS D1.1-1992 (13th edition).

Potential Impact Number: 22251 **SRP Section Number:** 6.1.1

Document: NUREG 0313 **Revision:** 2 , 1/1/88
Title: TECHNICAL REPORT ON MATERIAL SELECTION AND PROCESSING GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING. FINAL REPORT

Summary:

This report updates and supersedes the technical recommendations of NUREG-0313, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, published in July 1977, and its subsequent revision published in July 1980. This report provides the technical bases for the NRC staff's revised recommended methods to control the intergranular stress corrosion cracking susceptibility of BWR piping. For piping that does not fully comply with the material

selection, testing, and processing guideline combinations of this document, varying degrees of augmented inservice inspection are recommended. This revision also includes guidance and NRC staff recommendations (not requirements) regarding crack evaluation and weld overlay repair methods for long-term operation or for continuing interim operation of plants until a more permanent solution is implemented. Microfiche No: 44241:127

Potential Impact Number: 22252 **SRP Section Number:** 3.9.2

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

With regards to Leak-Before-Break it was noted that the staff is currently using the methodology and acceptance criteria provided in SRP Section 3.6.3 and NUREG-1061, Volume 3. However, the staff recognized the LBB technology is continually evolving and therefore, will review LBB requests for the ABWR plant on a case-by-case basis using the staff's methodology and acceptance criteria in effect at the time of the submittal.

Potential Impact Number: 22254 **SRP Section Number:** 3.9.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The promulgated rule change to General Design Criterion 4 of 10 CFR Part 50, Appendix A, that allows the application of the LBB method to piping systems could be used to eliminate the need to design for the dynamic effects of postulated pipe breaks, including pipe whip restraints and jet impingement shields, and to design subcompartments for dynamic pressurization loads if these loads are not essential to the containment function. Also, when the LBB method is demonstrated deterministically applicable to a piping system, the combination of SSE loads and LOCA loads effectively becomes the SSE loads alone. Specifically, the application of LBB technology can eliminate the local dynamic effects of postulated pipe ruptures from the design basis.

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The staff concluded that eliminating the combination of pipe rupture loads (global effects) with seismic loads for the containment and other seismic Category I structures is not acceptable. Furthermore, proposals to decouple LOCA and SSE loads for equipment and systems is not acceptable at this time because of the insufficient technical bases to extend the decoupling to structures.

Potential Impact Number: 22262 **SRP Section Number:** 3.9.2

Document: NUREG 1061 VOL 4
Title: REPORT OF THE U.S. NRC PIPING REVIEW COMMITTEE - EVALUATION OF OTHER LOADS AND LOAD COMBINATIONS

Summary:

Microfiche 28224:013 - referenced in the CE80 Chapter 3 DSER, ABWR Chapter 3 and 14 DFSER and EPRI Evolutionary Plant FSER. The staff stipulated multiple support excitation was acceptable as long as the methods would be in accordance with the staff recommendations on response combinations given in NUREG-1061, Volume 4 Section 2.

This NUREG deals with 'Staff Recommendations of Response Combinations.' It provides guidance on Independent Support Motion Method, Sequence of Combinations and High Frequency Modes. This information is relevant to SRP Sections 3.7.3 and 3.9.2.

Potential Impact Number: 22263 **SRP Section Number:** 15.6.3

Document: NUREG 1430
Title: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS

Summary:

SRP Section 15.6.3 references the STS for the three PWR vendors (W, B&W, and CE) to obtain primary-to-secondary leakage and primary coolant iodine concentration limits. The improved STSs (NUREGs -1430, -1431, and -1432) establish primary-to-secondary leakage limits and primary and secondary coolant iodine concentration limits.

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Potential Impact Number: 22265 **SRP Section Number:** 15.6.3

Document: NUREG 1431
Title: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS

Summary:

SRP Section 15.6.3 references the STS for the three PWR vendors (W, B&W, and CE) to obtain primary-to-secondary leakage and primary coolant iodine concentration limits. The improved STSs (NUREGs -1430, -1431, and -1432) establish primary-to-secondary leakage limits and primary and secondary coolant iodine concentration limits.

Potential Impact Number: 22266 **SRP Section Number:** 15.6.3

Document: NUREG 1432
Title: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS

Summary:

SRP Section 15.6.3 references the STS for the three PWR vendors (W, B&W, and CE) to obtain primary-to-secondary leakage and primary coolant iodine concentration limits. The improved STSs (NUREGs -1430, -1431, and -1432) establish primary-to-secondary leakage limits and primary and secondary coolant iodine concentration limits.

Potential Impact Number: 22268 **SRP Section Number:** 3.9.2

Document: NUREG 0800
Title: STANDARD REVIEW PLAN

Summary:

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SRP Section 3.7.2 has been changed since the last revision of 3.9.2. The following major change applies to 3.9.2 via reference to 3.7.3: (1) Guidance for modal combination of high frequency modes are provided and an acceptable approach is outlined in Appendix A.

Potential Impact Number: 22274 **SRP Section Number:** 3.9.2

Document: NUREG 0800
Title: STANDARD REVIEW PLAN

Summary:

SRP Section 3.7.3 has been changed since the last revision to 3.9.2. 3.9.2 parallels 3.7.3, but the latest revision to 3.7.3 provides additional clarification for methods of analysis for Category I buried piping. Additional references were also provided.

Potential Impact Number: 22275 **SRP Section Number:** 3.9.2

Document: NUREG 1061 VOL 3
Title: REPORT OF U.S. NRC PIPING REVIEW COMMITTEE - EVALUATION OF POTENTIAL FOR PIPE BREAKS

Summary:

Microfiche 27849:092 - referenced in the CE80 DSER, ABWR DFSER and EPRI Evolutionary Plant FSER as providing guidance for Leak Before Break analysis.

This report deals with the potential for pipe breaks and recommends modifications to the existing position. Specifically, this report contains the recommendations for application of the leak-before-break (LBB) approach in the NRC licensing process. The LBB approach means the application of fracture mechanics technology to demonstrate that high energy fluid piping is very unlikely to experience doubled-ended ruptures or their equivalent as longitudinal or diagonal splits.

Potential Impact Number: 22276 **SRP Section Number:** 3.9.2

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Potential Impacts

Document: C&S: ANSI OM
Title: OPERATION AND MAINTENANCE OF NUCLEAR POWER PLANTS

Summary:

Part 3, 'Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping Systems.' This provides general requirements for the assessment of piping system vibration for nuclear power plants. It includes steady-state and transient vibration testing and corresponding acceptance criteria, instrumentation and measurement techniques and recommendations for corrective action when required. It is applicable for preoperational and initial start-up testing, as well as being a guide for the assessment of vibration levels of applicable piping systems during plant operation.

This Part (along with Part 7) is referenced in the CE80+ DSER and ABWR DFSER. In the ABWR DFSER the staff found use of this Part (and Part 7) for preparing test specification to be acceptable. In the CE80+ DSER the staff asked the applicant to conduct the testing in accordance with this Part (and Part 7).

Potential Impact Number: 22277 **SRP Section Number:** 3.9.2

Document: C&S: ANSI OM
Title: OPERATION AND MAINTENANCE OF NUCLEAR POWER PLANTS

Summary:

Part 7, 'Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems.' This establishes requirements for development of procedures and for assessment of thermal expansion response in nuclear power plant piping systems. It includes pretest walkdown and test requirements to ensure the system can expand and contracts as required, and expected expansion can be accommodated and is within design tolerances. It includes acceptance criteria, instrumentation and measurement techniques, as well as suggested corrective actions. It is applicable to pre-operational testing, initial start-up testing, and commercial operation testing.

This Part (along with Part 3) is referenced in the CE80+ DSER and ABWR DFSER. In the ABWR DFSER the staff found use of this Part (and Part 3) for preparing test specification to be acceptable. In the CE80+ DSER the staff asked the applicant to conduct the testing in accordance with this Part (and Part 3).

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Potential Impact Number: 22278 **SRP Section Number:** 9.3.5

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff applied the requirements of 10 CFR 50.62(c)(4) as Acceptance Criteria for the review of the ABWR standby liquid control system. The system was reviewed against these requirements and found acceptable in Sections 7.4.1.2 and 9.3.5 of the FSER.

Potential Impact Number: 22279 **SRP Section Number:** 9.3.5

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

Because of the importance of valve reliability for the standby liquid control system storage tank discharge valves, the staff requires GE to confirm that the valves will have adequate testing requirements and the valves be incorporated into the Reliability Assurance Program.

Potential Impact Number: 22280 **SRP Section Number:** 9.3.5

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The staff applied 10 CFR 50.62 as Acceptance Criteria for evolutionary BWR standby liquid control systems during review of the EPRI Utility Requirements Document. (Also see FSER Ch. 5, Section 4.3).

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Potential Impact Number: 22285 **SRP Section Number:** 4.5.1

Document: C&S: ANSI N45.2.1
Title: Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants

Summary:

The ANSI N 45.2.1-1973 Standard is referenced in Section 4.5.1 of the Standard Review Plan (1981) for onsite cleaning and cleanliness control. Continued use of ANSI N 45.2.1-1973 would require consideration of reviewing the updated version, ANSI N 45.1-1-1980, for consistency. ANSI N 45.2.1 is not included in Table 6 for a code comparison by PNL.

Potential Impact Number: 22286 **SRP Section Number:** 4.5.1

Document: C&S: ASTM A708
Title: Recommended Practice for Detection of Susceptibility to Intergranular Corrosion in Severely Sensitized Austenitic Stainless Steel

Summary:

The Standard Review Plan (1981) in Section 4.5.1 cites ASTM A-708 as an alternate test that has been accepted by the branch for verification of non-sensitization of stainless steel and qualification of welding process employed in production. A detailed code comparison to evaluate the differences between the cited and current versions is not planned by PNL. Furthermore, ASTM discontinued ASTM A-708 in 1989.

Potential Impact Number: 22287 **SRP Section Number:** 3.9.2

Document: NRC GENERIC LETTER 84-04
Title: SAFETY EVALUATION OF WESTINGHOUSE TOPICAL REPORTS DEALING WITH ELIMINATION OF POSTULATED PIPE BREAKS IN PWR PRIMARY MAIN LOOPS

APPENDIX II
Potential Impacts

Summary:

After reviewing reports submitted to address asymmetric blowdown loads on the PWR primary systems that result from a limited number of discrete break locations as stipulated in NUREG-0609 (the staff's resolution of Unresolved Safety Issue A-2) the staff concluded an acceptable technical basis has been provided so that the asymmetric blowdown loads resulting from double ended pipe breaks in main coolant loop piping need not be considered as a design basis for the Westinghouse Owner's Group plants.

Potential Impact Number: 22290 **SRP Section Number:** 4.5.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

EPRI proposed requirements that use of Ni-Cr-Fe Alloy 600 in Evolutionary PWRs be justified and be restricted to applications outside the pressurizer and to applications for which low-carbon stainless steel cannot be used. EPRI proposed requirements that use of Ni-Cr-Fe alloys in Evolutionary BWRs be justified. Per EPRI's proposed requirements, pickling of wetted surfaces for all Ni-Cr-Fe alloys is prohibited for both BWRs and PWRs. The staff considers Alloy 600 undesirable for use in the ALWR because of the stress corrosion cracking experienced in existing nuclear plants. It specifically discourages the use of Alloy 600 in steam generators and pressurizers. Therefore, it will require that the applicant for any standard design application identify the use of Alloy 600 and provide information concerning its use. Those applications will be reviewed and approved by the staff on a case-by-case basis. In addition, the use of other Ni-Cr-Fe alloys such as Alloy 690 or 800 should be considered in applications for which primary water stress corrosion cracking is a concern. These applications also will be reviewed on a case-by-case basis. The staff concluded that, with the exception noted above, the implementation of the proposed EPRI requirements will ensure that Ni-Cr-Fe alloys will perform in service as designed.

EPRI proposed requirements that Ni-Cr-Fe Alloy X-750 will be used in the ALWR only in those applications where a lower strength material is impractical and for which a substantial base of successful experience exists. Where Inconel Alloy X-750 is used, the application will be in accordance with EPRI NP-6202, "Material Specification for Alloy X-750 in LWR Internal Components." The staff considers these requirements

acceptable because improperly heat-treated Alloy X-750 is susceptible to cracking and EPRI NP-6202 provides the technical basis for the use of this alloy.

Potential Impact Number: 22296 **SRP Section Number:** 9.3.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The requirements and guidance regarding the design of the postaccident sampling system (PASS) are given in 10 CFR 50.34(f)(2)(viii), RG 1.97, and NUREG-0737 TMI Action Plan Requirement II.B.3. EPRI proposed deviations from the requirements for the PASS in several areas. The staff's evaluation of this issue is provided in a draft Commission paper dated February 27, 1992 (included in Annex C of Ch 1 Appendix B). The staff concluded that ALWRs should be exempt from current requirements for containment hydrogen sampling and BWR reactor coolant dissolved gas analyses. The staff also recommends that required time limits be relaxed for post-accident sampling/analyses of certain parameters. Although the staff has identified its positions, its approach to resolving this issue has not been approved by the Commission and, therefore, does not represent an agency position. This is an open issue that must be satisfactorily resolved before the staff can complete its review of the Evolutionary Requirements Document.

Potential Impact Number: 22302 **SRP Section Number:** 9.3.2

Document: C&S: ANSI N13.1
Title: Guide for Sampling Airborne Radioactive Materials in Nuclear Facilities

Summary:

ANSI N13.1-1969, pertaining to sampling airborne radioactive materials, is cited as guidance for representative sampling from gaseous process streams and tanks in SRP Section 9.3.2, subsections II.3.b and III.2. The current version of this standard is ANSI N13.1-1969 (reaffirmed 1993).

APPENDIX II
Potential Impacts

Potential Impact Number: 22306 **SRP Section Number:** 6.1.1

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

The ferrite content limits for austenitic stainless steel castings and weld metal in the System 80+ design are broader than those in industry guidelines Electric Power Research Institute (EPRI), "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, Advanced Light Water Reactor (ALWR) Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991, and staff guidance U.S. Nuclear Regulatory Commission, "Technical Report on Material Selection and Processing Guidelines for (BWR) Coolant Pressure Boundary Piping," NUREG-0313, Rev. 2, January 1988. In Amendment Q to CESSAR-DC, ABB-CE committed to limit the ferrite content of austenitic stainless steel castings to a maximum of 20 percent for components with normal operating temperature exposure above 260 C (500 F). In Amendment L to CESSAR-DC, ABB-CE modified the upper ferrite limit to 15 percent for austenitic stainless steel weld metal. The staff believes that these lower limits at ferrite content 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.

Potential Impact Number: 22312 **SRP Section Number:** 6.3

Document: NUREG 0660
Title: NRC ACTION PLAN DEVELOPED AS A RESULT OF THE TMI-2 ACCIDENT

Summary:

A review of NUREG 0660 determined that Action Plan Item II.E.2.1 is an item which addresses reliance on ECCS. The objective of this item was stated as follows: "NRR will instruct all operating reactor licensees to provide a report that details experience with ECCS actuation (conditions, cause, frequency, results, etc.), compares cumulative experience with design bases for ECCS, and assesses the reliability of the system to perform its intended function under these conditions."

APPENDIX II
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Potential Impact Number: 22313 **SRP Section Number:** 6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

TMI Action Plan Item II.E.2.1 called for the collection of ECCS operating experience. The purpose was to ultimately decrease the reliance on ECCS for other than LOCAs.

This issue is covered under item II.K.3(17) which was implemented as part of NUREG-0737.

Item II.K.3(17) required a detailed report from operating licensees on ECCS outages and the 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 II.K.3(17).

Potential Impact Number: 22317 **SRP Section Number:** 6.3

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

The ADS valves, accumulators, and associated equipment and instrumentation must be capable of performing their functions during and following exposure to hostile environments and taking no credit for nonsafety-related equipment or instrumentation. Additionally, air (or nitrogen) leakage through valves must be accounted for in order to assure that enough inventory of compressed air is available to cycle the ADS valves.

Potential Impact Number: 22323 **SRP Section Number:** 3.7.1

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

APPENDIX II
Potential Impacts

Summary:

The staff discussed its proposal to decouple the operating basis earthquake (OBE) from the safe-shutdown earthquake(SSE). The regulations in Appendix A to 10 CFR Part 100 establish the OBE at 1/2 the intensity of the SSE. The staff stated that the OBE should not control the design of safety systems and was evaluating the possible changes to the regulations that would reduce the magnitude of the OBE relative to the 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 its SRM of June 26, 1990, the Commission approved the staff's recommendation.

Potential Impact Number: 22325 **SRP Section Number:** 3.7.1

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

SECY-93-087 discusses proposed positions related to elimination of OBE from design consideration and states that the staff is continuing to evaluate this issue in further detail.

Potential Impact Number: 22326 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: Appendix C

Summary:

10 CFR 73 Appendix C describes the scope, purpose and content requirements for Safeguards Contingency Plans.

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Potential Impacts

This impact supersedes Potential Impacts 11955, 11958, 11963, 11970, 11974, 11977, and 11979.

Potential Impact Number: 22327 **SRP Section Number:** 13.6

Document: RG 5.54 **Revision:** 0
Title: STANDARD FORMAT AND CONTENT OF SAFEGUARDS CONTINGENCY PLANS FOR NUCLEAR POWER PLANTS

Summary:

Regulatory Guide 5.54 provides detailed guidance on the purpose, format, content and objectives of Safeguards Contingency Plans.

This impact supersedes Potential Impacts 12331, 12332, 12333, 12334, 12335, 12336, 12337, 12338, 12339, 12340, 12341, 12342, 12343, 12344, 12345,12346 and 12347.

Potential Impact Number: 22328 **SRP Section Number:** 13.6

Document: NRC POLICY STATEMENT 53 FR 7534
Title: NUCLEAR POWER PLANT ACCESS AUTHORIZATION PROGRAM; POLICY STATEMENT

Summary:

Proposed policy statement defined NRC Policy (circa 1988) regarding unescorted access to nuclear power plant protected and vital areas. Published (for comment) the Industry Guidelines for Nuclear Power Plant Access Authorization Programs.

This potential impact supersedes Potential Impacts 12039, 12040, 12041, 12042, 12043, 12044, 12045, 12046, 12053, 12056, 12059, 12063, 12064, ,12065, 12066, 12068, 12070, 12081, 12083, 12092, 12094, 12095, 12099, 12106, 12111, 12712, 12713, 12714, 12735, 12733, 12732, 12731, 12729, 12728, 12721, 12720, 12734, 12719, 12718, 12717, 12716, and 12715.

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Potential Impact Number: 22330 **SRP Section Number:** 13.6

Document: FINAL SER ABWR CH 13
Title: CONDUCT OF OPERATIONS

Summary:

The staff requires that at least 60 days before loading fuel, the COL applicant will 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.

The SRP currently requires 'implementation... 1 to 2 months before fuel loading.'

Potential Impact Number: 22331 **SRP Section Number:** 5.2.1.1

Document: DRAFT SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The staff requested that the applicant revise CESSAR Table 5.2-1 and Section 3.2.2 or 3.9.3 to identify the ASME Code edition or addenda applicable to all Quality Group A, B, and C (Safety Class 1, 2, and 3) components, equipment, and their supports.

Potential Impact Number: 22335 **SRP Section Number:** 5.2.1.1

Document: C&S: ASME III
Title: CLASS 1 AND 2 SYSTEMS

Summary:

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Potential Impacts

ASME BPVC , Section III - 1980 is cited in SRP section 5.2.1.1 for Class 1 and Class 2 reactor coolant boundary components. Designation as Class 2 is determined in accordance with 10 CFR 50.55a.

Potential Impact Number: 22338 **SRP Section Number:** 5.2.1.1

Document: DRAFT FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The introduction to this section recognizes that according to 10 CFR 50.55a, components important to safety are subject to the appropriate ASME Code Section III classification requirements. This includes Class 1 (RG 1.26 Quality Group A) components, as well as Class 2 and 3 (RG 1.26 Quality Groups B and C, respectively) components.

The staff noted that SRP Section 5.2.1.1 recommends that SARs for both construction permits and operating license contain a table identifying the ASME component code, code edition and applicable addenda for all ASME Code, Section III, Class 1 and 2 pressure vessel components, piping, pumps and valves in the RCPB. While the staff noted this narrow scope of SRP Section 5.2.1.1 they also appear to have reviewed the code edition and addenda information from appropriate tables for all ASME Code Class 1, 2, and 3 components, not just the RCPB Class 1 and 2 components. Specifically, the staff stated, 'All ASME Code Class 1, 2, and 3 pressure retaining components and their supports shall be designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, using the specific edition and addenda provided in the SSAR. The COL applicant shall ensure that the design is consistent with the construction practices of the ASME BPV Code edition and addenda in effect, as endorsed in 10 CFR 50.55a.' In addition, the staffs final conclusion for this section covered all ASME Class 1, 2, and 3 components and their supports and their conformance to the appropriate ASME Code editions and addenda.

Potential Impact Number: 22341 **SRP Section Number:** 6.4

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

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Potential Impacts

Summary:

In the CE System 80+ DSER, the staff indicated that during its plant-specific reviews (of proposed resolution of GSI 83, Control Room Habitability), the staff will verify conformance with the applicable RGs 1.52, 1.78, and 1.95; industry standards [ANSI/ANS 59.2, ANSI/ASME N509, ANSI/ASME N510, ASME Code AG-1, 'Code of Nuclear Air and Gas Treatment,' and ASTM D3803, 'Standard Test Methods for Radiological Testing of Nuclear-Grade Gas-Phase Absorbents']; and evaluation methodology, and will verify that technical specifications and surveillance requirements for the control room habitability system, as a whole, to meet GDC 4, 5, and 19 have been proposed. This will demonstrate that the control room will adequately protect the control room operators and will remain habitable in accordance with Three Mile Island (TMI) Task Action Plan Item III.D.3.4 (NUREG-0737, 'Clarification of TMI Action Plan Requirements').

In the FSER, the staff did not discuss the above approach as necessary to resolve issue 83 for the CE System 80+ design. Instead, the staff concluded that the control room habitability systems and the control complex ventilation system comply with GDC 4, 5 and 19 by conforming to the acceptance criteria of SRP Sections 6.4 and 9.4.1. On the basis of this discussion, the staff concluded that ABB-CE has adequately addressed the safety concerns of Issue 83 in the CESSAR-DC and therefore, Issue 83 is resolved for the System 80+ design.

Potential Impact Number: 22344 **SRP Section Number:** 9.4.1

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

During its plant-specific reviews (of proposed resolution of GSI 83, Control Room Habitability), the staff will verify conformance with the applicable RGs 1.52, 1.78, and 1.95; industry standards [ANSI/ANS 59.2, ANSI/ASME N509, ANSI/ASME N510, ASME Code AG-1, 'Code of Nuclear Air and Gas Treatment,' and ASTM D3803, 'Standard Test Methods for Radiological Testing of Nuclear-Grade Gas-Phase Absorbents']; and evaluation methodology, and will verify that technical specifications and surveillance requirements for the control room habitability system, as a whole, to meet GDC 4, 5, and 19 have been proposed. This will demonstrate that the control room will adequately protect the control room operators and will remain habitable in accordance with Three

Mile Island (TMI) Task Action Plan Item III.D.3.4 (NUREG-0737, 'Clarification of TMI Action Plan Requirements').

In the FSER, the staff did not discuss the above approach as necessary to resolve issue 83 for the CE System 80+ design. Instead, the staff concluded that the control room habitability systems and the control complex ventilation system comply with GDC 4, 5 and 19 by conforming to the acceptance criteria of SRP Sections 6.4 and 9.4.1. On the basis of this discussion, the staff concluded that ABB-CE has adequately addressed the safety concerns of Issue 83 in the CESSAR-DC and therefore, Issue 83 is resolved for the System 80+ design.

Potential Impact Number: 22345 **SRP Section Number:** 15.6.3

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

In RAI Q440.109, the staff asked the applicant to address the staff's concern about preventing SG overfilling during an SGTR. In its response of May 8, 1992, the applicant stated that in the event of an SGTR, the affected SG water level is controlled by the following actions: (1) control the affected SG water by initiating or terminating the emergency feedwater, (2) minimize primary-to-secondary-side pressure difference by using the pressurizer gas vent valves and throttling the HPSI flow, and (3) control the SG water level by steaming the affected SG via the SG atmospheric dump valve in the event of loss of ac power. Neither acceptance criteria for the operator actions nor analyses were submitted to show that the operator actions are effective for controlling the affected SG water level, and that the radiological consequences are within the limit imposed by 10 CFR Part 100. The applicant is required to provide the technical basis that demonstrates the effectiveness of the operating strategy and compliance with the 10 CFR Part 100 requirements.

Potential Impact Number: 22346 **SRP Section Number:** 15.6.3

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

APPENDIX II
Potential Impacts

Summary:

Recently, the staff raised an issue concerning the potential for containment bypass due to a rupture of one or more SG tubes. During a tube rupture event, the potential exists for lifting of SG safety or relief valves and discharging primary system radioactive inventory outside the containment. Such a containment bypass is undesirable for either a design- basis event or a postulated severe accident. Consequently, the staff believes that possible mitigation of this containment challenge should be considered.

Consistent with the staff recommendations in SECY-90-016 and SECY-93-087, the staff, in Section 15.3.9 of this chapter, states that ABB-CE should consider potential design features that would reduce the amount of containment bypass leakage from an SGTR event. The staff also recommended three potential design features that could mitigate the release from the tube ruptures. Instead of these three design features, however, CESSAR-DC Appendix 5F provides ABB-CE's study to evaluate certain automatic design features that can be used to enable the plant to mitigate SGTR consequences. Sections 3 and 4 of CESSAR-DC Appendix 5F describe these analyses and provide an evaluation of the attendant benefits and limitations of each of these automatic design features. As noted in Section 19.2.3.3.5.2 of this report, ABB-CE assessed the three design alternatives identified in SECY-93-087 in a report dated September 23, 1993 and titled, "Design Alternatives for the System 80+ Nuclear Power Plant," and found these alternatives to be cost prohibitive (see Section 19.2.3.3.5.2 for more information).

As a result of these analyses, some features have been added to the System 80+ design to reduce the potential containment bypass leakage from the SGTR events. These features include: (1) a design modification to the component cooling water system (CCWS) to ensure continued cooling of the instrument air compressors after a safety injection actuation signal (SIAS), (2) addition of two nitrogen-16 (N-16) radiation monitors (one per SG) in the steamlines, (3) implementation of technical specifications and ITAACs related to N-16 monitors, and (4) emergency operations guidelines (EOGs) improvements. The staff has determined that this open item has been properly addressed with these enhancements. In arriving at this conclusion, the staff considered whether the System 80+ design provides sufficient time, diagnostic information, mitigation capability, and proper EOGs for operator coping actions following an SGTR event to mitigate the consequence.

Potential Impact Number: 22347 **SRP Section Number:** 15.6.3

APPENDIX II
Potential Impacts

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

The applicant analyzed three SGTR events and presented the analysis in CESSAR Sections 15.6.3.1 through 15.6.3.3, respectively:

- SGTR without a LOOP,
- SGTR with a LOOP,
- SGTR with a LOOP and a single failure.

The most limiting failure, the failure of an atmospheric dump valve to close in the affected SG, was assumed for calculating radiological releases resulting from an SGTR event. The staff found this analysis to be acceptable for evaluating SGTR.

Potential Impact Number: 22349 **SRP Section Number:** 3.7.3

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff's review of Seismic Subsystem Analysis (3.9.2.2) was performed in accordance with SRP Section 3.9.2 and consisted of an evaluation of SSAR Section 3.7.3.

Potential Impact Number: 22350 **SRP Section Number:** 3.7.3

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

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Potential Impacts

The staff's review of seismic subsystem analysis (3.9.2.2) was conducted in accordance with the requirements of SRP Section 3.7.3 which the staff noted are essentially the same as those of SRP Section 3.9.2.

Potential Impact Number: 22351 **SRP Section Number:** 3.7.3

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

SECY-93-087 discusses proposed positions related to elimination of OBE from design consideration. Included in this discussion is the effect of eliminating the OBE on determining the number of earthquake cycles.

At present, SRP Section 3.7.3 Item II.2 and 3.9.2 Item II.2.b both set the number of earthquake cycles during the plant life such that at least one SSE and five OBE are assumed. The number of cycles per earthquake is obtained from the synthetic time history or is set at a minimum of 10 maximum stress cycles per earthquake. NOTE: In SECY-93-087 the staff only identified the relationship to 3.9.2, the identical item in 3.7.3 (which 3.9.2 references) was not mentioned.

In SECY-93-087, the staff proposed two alternatives:

1) Uses an equivalent cyclic load which 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) Or the staff proposed that number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Standard 344-1987.

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Potential Impact Number: 22352 **SRP Section Number:** 3.9.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

SECY-93-087 discusses proposed positions related to elimination of OBE from design consideration. Included in this discussion is the effect of eliminating the OBE on determining the number of earthquake cycles.

At present, SRP Section 3.7.3 Item II.2 and 3.9.2 Item II.2.b both set the number of earthquake cycles during the plant life such that at least one SSE and five OBE are assumed. The number of cycles per earthquake is obtained from the synthetic time history or is set at a minimum of 10 maximum stress cycles per earthquake. NOTE: In SECY-93-087 the staff only identified the relationship to 3.9.2, the identical item in 3.7.3 (which 3.9.2 references) was not mentioned.

In SECY-93-087, the staff proposed two alternatives:

1) Uses an equivalent cyclic load which 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) Or the staff proposed that number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Standard 344-1987.

Potential Impact Number: 22353 **SRP Section Number:** 3.7.3

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

APPENDIX II
Potential Impacts

Summary:

The staff discussed its proposal to decouple the operating basis earthquake (OBE) from the safe-shutdown earthquake(SSE). The regulations in Appendix A to 10 CFR Part 100 establish the OBE at 1/2 the intensity of the SSE. The staff stated that the OBE should not control the design of safety systems and was evaluating the possible changes to the regulations that would reduce the magnitude of the OBE relative to the 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 its SRM of June 26, 1990, the Commission approved the staff's recommendation.

Potential Impact Number: 22354 **SRP Section Number:** 3.9.2

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

The staff discussed its proposal to decouple the operating basis earthquake (OBE) from the safe-shutdown earthquake(SSE). The regulations in Appendix A to 10 CFR Part 100 establish the OBE at 1/2 the intensity of the SSE. The staff stated that the OBE should not control the design of safety systems and was evaluating the possible changes to the regulations that would reduce the magnitude of the OBE relative to the 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 its SRM of June 26, 1990, the Commission approved the staff's recommendation.

Potential Impact Number: 22355 **SRP Section Number:** 3.7.3

Document: C&S: IEEE 344-87
Title: Recommended Practice for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations

Summary:

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Potential Impacts

IEEE Standard 344-1987, Appendix D is identified in SECY-93-087 as one of two proposed alternate methods of determining the number of earthquake cycles when the OBE is eliminated from the design basis.

Potential Impact Number: 22356 **SRP Section Number:** 15.6.3

Document: NRC POLICY STATEMENT 52 FR 3788
Title: PROPOSED POLICY STATEMENT ON TECHNICAL SPECIFICATION IMPROVEMENTS FOR NUCLEAR POWER REACTORS

Summary:

This statement presents the policy of the Nuclear Regulatory Commission with respect to the scope and purpose of Technical Specifications for nuclear power plants as required by 10 CFR 50.36. The Policy Statement is expected to produce an improvement in the safety of nuclear power plants through the development of more operator oriented Technical Specifications, improved Technical Specification Bases, reduced action statement induced plant transients, and more efficient use of NRC and industry resources.

Potential Impact Number: 22357 **SRP Section Number:** 15.6.3

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The staff agrees with the statement in the Evolutionary Requirements Document regarding the not yet defined and assessed elements of this issue. However, it is not clear why EPRI states that total resolution of this issue is not applicable to the evolutionary design and what is EPRI's position regarding the actions required by the MPA letters. Consequently, the staff will evaluate the resolution of Issue 135 during its review of individual applications for FDA/DC using the appropriate requirements and guidance current at that time.

APPENDIX II
Potential Impacts

Potential Impact Number: 22358 **SRP Section Number:** 15.6.3

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

EPRI states that containment bypass sequences can result in significant releases from the containment and have the potential to be important risk contributors. Bypass sequences that have been identified as important in past PRAs include

- steam generator tube rupture (PWR only)

In the NUREG-1150 studies for Surry and Sequoyah, bypass sequences dominate early fatality risk. The staff agrees that containment bypass sequences are important and, therefore, agrees with the EPRI requirement to assess such sequences.

Potential Impact Number: 22359 **SRP Section Number:** 15.6.3

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

SECY-90-016 is referenced in the CE80+ DSER relative to containment bypass due to SGTR. In SECY-90-016, Issue III.D, "Containment Performance", the staff requires the applicant to reduce the potential for conditional containment failure. This would include the potential for containment bypass due to SGTR.

Potential Impact Number: 22368 **SRP Section Number:** 6.3

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

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Potential Impacts

The capability of all systems and components necessary to provide core cooling and decay heat removal following a station blackout should be determined, including station battery capacity, condensate storage tank capacity, compressed air capacity, and instrumentation and control requirements.

If a system is required for primary coolant charging and makeup, reactor coolant pump seal cooling or injection, decay heat removal, or maintaining appropriate containment integrity specifically to meet the station blackout duration recommended in Regulatory Position 3.1, specific criteria listed in the regulatory guide should be met:

Potential Impact Number: 22371 **SRP Section Number:** 6.3

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Section 12.2 of the EPRI FSER contains guidance and positions on the inservice testing of pumps and valves.

Potential Impact Number: 22373 **SRP Section Number:** 6.3

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016 the staff concludes that the ALWR designs should include a depressurization system. In the SRM for SECY 90-016 the Commission approved the staff's position that the ELWR designs include a depressurization system

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Potential Impact Number: 22374 **SRP Section Number:** 6.3

Document: FINAL SER EPRI CH 5

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Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

The EPRI final SER reviewed the safety depressurization and venting system in regards to direct containment heating under section 5.4 safety injection system. The EPRI final SER states, "In revision 3 of chapter 5, EPRI requires that the safety depressurization and venting system be capable of reducing reactor coolant system pressure to 250 psig or less prior to reactor vessel melt-through, as a means to preclude containment challenges through direct containment heating (DCH). The staff concludes that this is an acceptable design objective. However, the designer should justify that the automatic depressurization system will depressurize low enough to preclude DCH. The designer should also demonstrate that the depressurization system is adequate to provide sufficient capacity to handle primary feed-and-bleed operations during a total loss-of-feedwater event and to prevent creep rupture of steam generator tubes from a postulated high-pressure core-damage event.

Potential Impact Number: 22380 **SRP Section Number:** 9.5.6

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

9.5.6 Diesel Generator Starting Air System. GE identified several design parameters as future interface requirements to be determined once a diesel generator vendor has been selected. The interface requirements to be specified would dictate design parameters such as the volume and design pressure of the air receivers and compressor size (sufficient discharge flow to recharge the system in under 30 minutes). GE has committed to provide additional requirements: one addressing the number of starts from the air receiver, and a second on the time needed to recharge the air receivers.

Potential Impact Number: 22383 **SRP Section Number:** 15.6.3

Document: SECY 93-087

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Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

The staff recommended 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 SG tube rupture.

Potential Impact Number: 22384 **SRP Section Number:** 5.2.1.2

Document: C&S: ASME III
Title: CLASS 1 AND 2 SYSTEMS

Summary:

The ASME Code is referenced in this SRP Section for:

Section III, Division 1, Subsection NB - Class 1 Components,
Section III, Division 1, Subsection NC - Class 2 Components,
Section III, Division 1, Subsection ND - Class 3 Components,
Section III, Division 1, Subsection NE - Class MC Components,
Section III, Division 1, Subsection NF - Component Supports,
Section III, Division 1, Subsection NG - Core Support Structures,
Section III, Division 2, Concrete Containments.

Potential Impact Number: 22387 **SRP Section Number:** 5.2.1.2

Document: C&S: ASME III
Title: CLASS 1 AND 2 SYSTEMS

Summary:

Under the rules of ASME, Section III, Subsection NCA-1140, 'Use of Code Editions, Addenda, and case,' the owner or it's designee establish the Code Edition and

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Addenda to be included in the Design Specifications. This subsection also provides guidance for the relationships between Code edition and addenda to be used in Design Specifications and construction permit applications. The subsection also provides information regarding code cases. Specifically, code cases are permissible and may be used beginning with the date of approval by the ASME Council (and the American Concrete Institute for Division 2 design and construction). Only Code cases that are specifically identified as being applicable to this Section may be used.

Potential Impact Number: 22388 **SRP Section Number:** 5.2.1.2

Document: DRAFT FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The staff requested that the COL applicant ensure that the design is consistent with the construction practices (including inspection and examination methods) of the ASME Boiler and Pressure Vessel Code edition and addenda in effect, as endorsed in 10 CFR 50.55a. The portions of the later Code editions and addenda shall be identified to the Nuclear Regulatory Commission (NRC) staff for review and approval with the COL application.

The staff noted that this position is not totally consistent with ASME, Section III, Subsection NCA-1140, 'Use of Code Editions, Addenda, and Cases,' and a comparison was provided. Included in the comparison was a discussion of Code Cases:

1) In NCA-1140(c) code cases are permissible and may be used beginning with the date of approval by the ASME Council (and the American Concrete Institute for Division 2 design and construction). Only Code cases that are specifically identified as being applicable to this Section may be used. However, the staff stated that only those Code cases identified in RG 1.84 Rev. 24 or 1.85 Rev. 24, as specified in the SSAR may be used. The COL applicant may submit for staff review and approval future Code cases that are endorsed in future revisions of RG 1.84 and 1.85 with its COL application provided these cases do not involve a change to the certified design or to the supporting SSAR design information.

2) In NCA-1140(d) code cases may be used by mutual consent of the owner or it's designee, and the certificate holder on or after the date permitted by (c) above. For Division 2 design and construction, the consent of the designer shall also be obtained.

However, the staff stated that code cases to be used in the design of the ABWR must be identified by GE in the SSAR during design certification.

Potential Impact Number: 22389 **SRP Section Number:** 5.2.1.2

Document: DRAFT FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The COL applicant may submit, with its COL application, future code cases that are endorsed in RG 1.84 and 1.85 at the time of the applicant COL provided they do not alter the staff's safety findings on the ABWR certified design. In addition, the COL applicant should submit those code cases that are applicable to RG 1.147, 'Inservice Inspection Code Case Acceptability - ASME Section XI, Division 1.'

Potential Impact Number: 22390 **SRP Section Number:** 5.2.1.2

Document: DRAFT SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

It is the staff's position that the applicant should provide a complete list of all ASME Code Case interpretations that are presently referenced in the CESSAR. This list may be augmented by a COL applicant in the future through the use of a 50.59-like process.

Potential Impact Number: 22394 **SRP Section Number:** 9.5.6

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

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Section 9.5.6 of the Final SER identified several design parameters as future requirements to be determined once a DG vendor has been selected. Requirements to be specified include the devices to crank the engine, air start requirements with regard to the duration of the cranking cycle, and the number of engine revolutions per start attempt. These requirements will, in turn, dictate design parameters, such as the volume and design pressures of the air receivers (sufficient for five diesel engine start cycles) and compressor size (capacity to recharge the system within 30 minutes). This information should be provided as part of each plant license application.

Potential Impact Number: 22395 **SRP Section Number:** 9.5.6

Document: RG 1.9
Title: SELECTION, DESIGN, AND QUALIFICATION OF DIESEL-GENERATOR UNITS USED AS STANDBY (ONSITE) ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

Summary:

Diesel generator shall meet the requirements of IEEE Standard 387-1984.

Regulatory Position C-1.5 through C-1.8 includes positions on design features to be provided by the diesel generator unit and its support systems. These sections supplement the IEEE requirements for surveillance systems and diesel generator protective trips.

Potential Impact Number: 22400 **SRP Section Number:** 9.5.6

Document: C&S: DEMA Standard
Title: Diesel Manufacturers Association (DEMA) Standard

Summary:

The SRPs rely on the DEMA standard as part of the Acceptance Criteria. The Standard is not readily available. The DEMA organization is listed as defunct in the 1993 Encyclopedia of organizations

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Potential Impact Number: 22401 **SRP Section Number:** 9.5.6

Document: C&S: IEEE 387
Title: IEEE STANDARD CRITERIA FOR DIESEL-GENERATOR UNITS APPLIED AS STANDBY POWER SUPPLIES FOR NUCLEAR POWER GENERATING STATIONS

Summary:

The Standard describes the criteria for the application of diesel-generator units as individual units of the Class 1E standby power supplies in stationary nuclear power generating stations. The starting system is within the scope of the standard. Note: The current Standard is dated November 21, 1984.

Potential Impact Number: 22402 **SRP Section Number:** 4.4

Document: FINAL SER EPRI CH 4
Title: EPRI URD-EPD: REACTOR SYSTEMS

Summary:

In the DSER for Chapter 4, the staff concluded that the requirements of Section 2.2.4 of Chapter 4 of the Evolutionary Requirements Document pertaining to freedom from power oscillations were inadequate. Revisions to Chapter 4 have significantly enhanced the original requirements. Section 2.2.4 now requires that fuel safety limits not be exceeded for any power oscillation, without operator actions, through inherent characteristics or prevention by the reactor protection system.

In response to concerns expressed by the NRC staff, the EPRI URD now requires that operation beyond T-H stability limits be prevented by an operationally-proven, reliable I&C system. The I&C system must be capable of detecting all expected modes of instability, and automatically reducing reactor power to return the reactor to a stable condition before fuel safety limits are exceeded.

Potential Impact Number: 22403 **SRP Section Number:** 4.4

Document: FINAL SER EPRI CH 1

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Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Section 3.5.2 of Chapter 1 requires the PWR plant to be capable of operating at reduced power with a secured coolant pump to enhance the availability of the plant and to reduce reactor trips. Appendix B to Chapter 1 indicates EPRI's commitment to comply with Generic Letter (GL) 86-09, "Technical Resolution of Generic Issue No. B-59 - (N-1) Loop Operation in BWRs and PWRs." GL 86-09 states that (N-1) loop operation is acceptable provided acceptable evaluation results are shown for certain plant-specific design characteristics, such as the impact of the down loop on instrumentation and control systems, human factors, operational systems, safety systems, status of valves, core-flow distribution, and potential for cold water reactivity insertion. Since these characteristics are highly dependent on the specific design of the plant, acceptability of operation with one secured reactor coolant pump is subject to plant-specific evaluation to address the concerns delineated in GL 86-09. The staff will evaluate that analysis during its review of an individual application for FDA/DC.

Potential Impact Number: 22404 **SRP Section Number:** 4.4

Document: FINAL SER EPRI CH 4
Title: EPRI URD-EPD: REACTOR SYSTEMS

Summary:

As part of its continuing study of T-H stability, the NRC has raised concerns about T-H stability during an ATWS. Since the EPRI URD does not address the issue, the NRC will review FDA/DC applications to ensure that designers demonstrate that the T-H stability performance of the core is acceptable during an ATWS event.

Potential Impact Number: 22405 **SRP Section Number:** 4.4

Document: FINAL SER ABWR CH 4
Title: REACTOR

Summary:

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In the DFSER, the staff stated that the information presented by GE and the staff review of the subject of instability during ATWS transients was not complete. This was DFSER Open Item 4.4-1. GE subsequently submitted a report describing their analyses of oscillations during limiting ATWS transients susceptible to thermal-hydraulic instability and possibly large oscillations. GE performed calculations using the TRACG code, which has been used for stability studies for operating BWRs and has been audited by the staff. The TRACG modelling and input for these calculations were found to be acceptable. The staff concludes that the issue of instability induced large power oscillations during ATWS has been properly addressed for the ABWR. GE has also included this information in the SSAR and the staff find it to be acceptable. On the basis of this evaluation, this item is resolved.

Potential Impact Number: 22407 **SRP Section Number:** 9.2.2

Document: DRAFT SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

Probabilistic risk assessment (PRA) analyses have indicated that the overall probability of core damage due to a small break could be dominated by RCP seal failures. The objective for the resolution of GSI-23 includes improving the reliability of RCP seals by reducing the probability of seal failure during normal operations and under normal conditions. Specifically, acceptable resolutions to GSI-23 vulnerabilities include an RCP seal design that ensures the RCP seal integrity for an event of loss of all seal injection cooling systems resulting from station blackout (SBO) for an extended period.

...

The staff requires that the applicant submit adequate test data to demonstrate the ABB-CE KSB seal integrity during an SBO during an extended period or provide a diverse seal injection system, which should be a safety-grade system and independent of the CVCS (a non-safety-grade system) and associated support systems to the extent practicable. The staff will review the further RCP seal submittals when they become available.

Potential Impact Number: 22408 **SRP Section Number:** 9.2.2

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Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Issue 23 has not yet been generically resolved by the staff and has a high safety priority. Unless the final agency resolution of this issue dictates other actions for ALWR evolutionary plant designs, the staff 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.

Potential Impact Number: 22409 **SRP Section Number:** 9.2.1

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

GI-130 identified a concern with multi-unit sites that have only two service water pumps per plant with crosstie capability between plants. The identified section of the SER describes the applicants proposed resolution of GI-130 and describes the staff review of this issue. The staff indicated that the Design Certification applicant should provide adequate guidance such that the Combined License applicant proposing a shared service water system at a multi- unit site will be able to provide supporting documentation and analysis sufficient to respond to the concerns of GI-130 and demonstrate compliance with GDC-5.

Potential Impact Number: 22410 **SRP Section Number:** 9.2.1

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

GI-130 identified a concerns with multi-unit sites that have only two service water pumps per plant with crosstie capability between plants. The identified section of the

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SER describes the applicants proposed resolution of GI-130 and describes the staff review of this issue.

Potential Impact Number: 22411 **SRP Section Number:** 9.2.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

GI-130 identified a concerns with multi-unit sites that have only two service water pumps per plant with crosstie capability between plants. The identified section of the SER describes the EPRI proposed resolution of GI-130 and describes the staff review of this issue.

Potential Impact Number: 22412 **SRP Section Number:** 9.2.1

Document: FINAL SER EPRI CH 8
Title: EPRI URD-EPD: PLANT COOLING WATER SYSTEMS

Summary:

GI-130 identified a concerns with multi-unit sites that have only two service water pumps per plant with crosstie capability between plants. The identified section of the SER describes the EPRI proposed resolution of GI-130 and describes the staff review of this issue.

Potential Impact Number: 22413 **SRP Section Number:** 9.2.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.34(f)(2)(xxi)

Summary:

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Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (Applicable to BWR's only). (II.K.1.22)

Potential Impact Number: 22414 **SRP Section Number:** 9.2.1

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

This section of the DFSER describes the staff's review of the applicants compliance with 10 CFR 50.34(f)(2)(xxi) and II.K.1.22 related to automatic and manual actions in the event the main feedwater system is inoperable. The staff found that the applicant's description of automatic ECCS functions and operator actions in response to a loss of feedwater were acceptable in addressing 10 CFR 50.34(f)(2)(xxi) and II.K.1.22.

Potential Impact Number: 22416 **SRP Section Number:** 10.4.9

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

The staff noted that the Evolutionary Requirements Document states that the EFWS must be provided with a means to detect potential EFWS pump steam binding as a result of steam and hot water leakage through check valves in the pump discharge lines. This will consist of temperature monitoring of the portion of discharge piping upstream of the check valves, with indication and alarm in the control room. Appropriate vents and drains will be provided for removing steam in the event steam is detected. The Evolutionary Requirements Document also states that the EFWS must be provided with means to permit periodic surveillance testing of EFW pumps and valves and functional testing of the integrated operation of the system.

Potential Impact Number: 22417 **SRP Section Number:** 10.4.9

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Document: FINAL SER CE80 CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

Steam Binding of EFW Pumps: Concerning the steam binding of emergency feedwater system pumps, the applicant stated that (a) the temperature sensor located between the flow control valve and the isolation valve on each subtrain is continuously monitored and audibly alarmed in the control room; the sensor will be monitored locally upon loss of control room indication; and the referencing applicant will provide an operating procedure for local monitoring and (b) the system vents through EFW storage tanks continuously and by use of normally closed isolation valves (upstream of the interface with the main feedwater system); and audible indication will signal the control room operator to vent the EFW pumps; and this provision will be in the referencing applicant's plant operating procedures. These provisions are site specific and would provide guidance for recognizing the effects of steam binding of the EFW pumps and for restoring the system to operable status following a steam binding incident.

Potential Impact Number: 22418 **SRP Section Number:** 10.4.9

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

The EFW system is designed to avoid steam binding of the EFW pumps by continuous system venting through the EFW storage tanks and by the use of normally closed isolation valves upstream of where the EFW system joins the main feedwater system; however, in the event that steam binding of the EFW pumps does occur, the control room alarm associated with the temperature sensor discussed above will signal the plant operator to vent the EFW pumps. Plant operating procedures developed by the owner-operator will prescribe this action. The applicant has adequately addressed GSI-93 for the System 80+.

Potential Impact Number: 22421 **SRP Section Number:** 9.2.1

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Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

The staff will review site-specific aspects of the resolution of GI 51 as part of its review of the site-specific application. Such a review will include, among other things, site-specific surveillance and control program and whether the referencing applicant commits to meet all the GE identified interface requirements.

Potential Impact Number: 22422 **SRP Section Number:** 9.2.1

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

The staff will review site-specific aspects of the resolution of GSI-51 as when it reviews the site-specific application. Site-specific reviews will include a maintenance and inspection program(s) and the determination of whether or not the referencing applicant commits to meet all the applicant- identified interface requirements.

Potential Impact Number: 22423 **SRP Section Number:** 13.2.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR Part 50.120

Summary:

Each nuclear power plant applicant, by [October 25, 1993, [publication] or 18 months prior to fuel load, whichever is later, and each nuclear power plant licensee, by [October 25, 1993, publication] shall establish, implement, and maintain a training program derived from a systems approach to training as defined in 10 CFR 55.4. The training program must provide for the training and qualification of specific categories of nuclear power plant personnel listed in the regulation.

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Potential Impact Number: 22424 **SRP Section Number:** 13.2.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR Part 50.120

Summary:

Each nuclear power plant applicant, by [October 25, 1993, [publication] or 18 months prior to fuel load, whichever is later, and each nuclear power plant licensee, by [October 25, 1993, publication] shall establish, implement, and maintain a training program derived from a systems approach to training as defined in 10 CFR 55.4. The training program must provide for the training and qualification of specific categories of nuclear power plant personnel listed in the regulation.

Potential Impact Number: 22428 **SRP Section Number:** 13.2.2

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 10 CFR Part 52.78

Summary:

52.78 Contents of applications; training and qualification of nuclear power plant personnel

(a) Applicability. The requirements of this section apply only to the personnel associated with the operating phase of the combined licenses.

(b) The application must demonstrate compliance with the requirements for training programs established in 50.120 of this chapter.

Potential Impact Number: 22431 **SRP Section Number:** 3.9.6

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Document: NRC GENERIC LETTER 89-10 SUP 5
Title: INACCURACY OF MOTOR-OPERATED VALVE DIAGNOSTIC EQUIPMENT

Summary:

On the basis of the new information on MOV diagnostic equipment inaccuracy discussed in this generic letter, licensees were requested to reexamine their MOV programs and to identify measures taken or planned to account for uncertainties in properly setting valve operating thrust to ensure operability.

Potential Impact Number: 22441 **SRP Section Number:** 13.5.1.1

Document: NRC GENERIC LETTER 91-16
Title: LICENSED OPERATORS' AND OTHER NUCLEAR FACILITY PERSONNEL FITNESS FOR DUTY

Summary:

The Commission believes all facility licensees should manage licensed activities so as to provide reasonable assurance that nuclear reactor personnel will perform their tasks in a reliable and trustworthy manner, and not be under the influence of alcohol or any prescription, over-the-counter, or illegal substance that in any way affects their ability to safely and competently perform their duties. Individuals should also know that, even if a conviction for possession or distribution of a controlled substance does not result in a declaration of ineligibility for Federal benefits by the sentencing court, the NRC may, in accordance with its independent responsibility to protect the public health and safety and the common defense and security, act on its own to deny, revoke, suspend, or modify a license or remove an individual from licensed activities, upon such conviction. NRC sanctions for licensed operators in response to positive drug tests are described in 10 CFR Parts 2 and 55.

On June 7, 1989 (54 FR 24468), the NRC issued a new 10 CFR Part 26, "Fitness-for-Duty Programs," to require each licensee authorized to construct or operate a nuclear power reactor to implement a fitness-for-duty program.

Potential Impact Number: 22442 **SRP Section Number:** 13.6

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Document: NRC GENERIC LETTER 91-16
Title: LICENSED OPERATORS' AND OTHER NUCLEAR FACILITY PERSONNEL FITNESS FOR DUTY

Summary:

The Commission believes all facility licensees should manage licensed activities so as to provide reasonable assurance that nuclear reactor personnel will perform their tasks in a reliable and trustworthy manner, and not be under the influence of alcohol or any prescription, over-the-counter, or illegal substance that in any way affects their ability to safely and competently perform their duties. Individuals should also know that, even if a conviction for possession or distribution of a controlled substance does not result in a declaration of ineligibility for Federal benefits by the sentencing court, the NRC may, in accordance with its independent responsibility to protect the public health and safety and the common defense and security, act on its own to deny, revoke, suspend, or modify a license or remove an individual from licensed activities, upon such conviction. NRC sanctions for licensed operators in response to positive drug tests are described in 10 CFR Parts 2 and 55.

On June 7, 1989 (54 FR 24468), the NRC issued a new 10 CFR Part 26, "Fitness-for-Duty Programs," to require each licensee authorized to construct or operate a nuclear power reactor to implement a fitness-for-duty program.

Potential Impact Number: 22444 **SRP Section Number:** 9.2.1

Document: NRC GENERIC LETTER 91-13
Title: REQUEST FOR INFORMATION RELATED TO THE RESOLUTION OF GENERIC ISSUE 130, 'ESSENTIAL SERVICE WATER SYSTEM FAILURES AT MULTI-UNIT SITES,' PURSUANT TO 10 CFR 50.54(f)

Summary:

To implement actions recommended by this generic letter each licensee would modify technical specifications in accordance with Enclosure 1 to this generic letter, as well as implement procedures for operating and maintaining HPI pump integrity and testing and manipulating the ESWS crosstie between units during a LOSW event.

The purpose of this letter is to inform affected licensees and applicants of the technical findings resulting from the NRC resolution of Generic Issue 130, "Essential Service

Water System Failures at Multi-Unit Sites," and to request information from licensees and applicants at affected multi-unit sites relating to the applicability of certain findings regarding their facilities. Affected licensees and applicants are required to respond to the request for information contained in this letter, but no new requirements or staff positions are imposed on the affected licensees and applicants by this letter.

Potential Impact Number: 22465 **SRP Section Number:** 10.4.7

Document: NRC GENERIC LETTER 80-95
Title: FINAL EDITION OF NUREG-0619, 'BWR FEEDWATER NOZZLE AND CONTROL ROD DRIVE RETURN LINE NOZZLE CRACKING'

Summary:

This Generic Letter transmitted NUREG-0619 for licensee implementation. NUREG-0619 provides the staff's resolution of the NRC's Generic Technical Activity A-10, which was an "Unresolved Safety Issue". NUREG-0619 describes the technical issues, the technical studies and 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.

Potential Impact Number: 22471 **SRP Section Number:** 6.7

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

'The staff concludes that the above described approach {performing dynamic seismic analysis of the condenser, a portion of the main steam bypass line, and the piping between the turbine stop valve and the turbine inlet} to resolve the BWR main steam line issue for both evolutionary and passive ALWRs provides reasonable assurance that the main steam piping from the outermost isolation valve up to the turbine stop valve, the main steam drain and by-pass lines up to the condenser, and the main condenser will retain their pressure and structural integrity during and following a SSE.

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The staff recommends that the Commission approve the above-described approach to resolve the main steamline classification for both evolutionary and passive ALWRs.'

Potential Impact Number: 22473 **SRP Section Number:** 6.7

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

"GE has eliminated the main steam isolation valve leakage control system for the ABWR plant design. Instead, it proposes to rely on the use of an alternate leakage path that takes advantage of the large volume and surface area in the main steam piping, bypass line, and condenser to hold up and plate out the release of fission products following core damage. In this manner, the main steam piping, bypass line, and condenser are used to mitigate the consequences of an accident and are required to remain functional during and after an SSE. ...

The resolution of this item will be discussed in Section 3.7.2 of this report. This is Open Item 3.2.1-1."

Potential Impact Number: 22476 **SRP Section Number:** 6.7

Document: DRAFT FINAL SER ABWR CH 14
Title: VERIFICATION PROGRAMS

Summary:

"For the ABWR plant design, GE proposes to eliminate the main steam isolation valve leakage control system and to rely on the use of an alternate leakage path that takes advantage of the large volume and surface area in the main steam piping, bypass line, and condenser to hold up and plate out the release of fission products following core damage. In this manner, the main steam piping, bypass line, and condenser will be used to mitigate the consequences of an accident and will be required to remain functional during and after a safe shutdown earthquake (SSE)."

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Potential Impact Number: 22482 **SRP Section Number:** 5.4.7

Document: NRC GENERIC LETTER 80-53
Title: GENERIC LETTER CONCERNING DECAY HEAT REMOVAL CAPABILITY

Summary:

This generic letter requested licensees to propose TS changes for facilities that provide for redundancy in decay heat removal capability for their plant(s) in all modes of operation. To assist licensees in preparing their submittal the NRC enclosed a copy of Model TSs which would provide an acceptable resolution of the NRC's concern.

Potential Impact Number: 22488 **SRP Section Number:** 10.4.9

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.63(a)(2)

Summary:

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.

Potential Impact Number: 22494 **SRP Section Number:** 9.3.4

Document: NRC GENERIC LETTER 80-21
Title: VACUUM CONDITION RESULTING IN DAMAGE TO CHEMICAL VOLUME CONTROL SYSTEM (CVCS) HOLDUP TANKS (SOMETIMES CALLED 'CLEAN WASTE RECEIVER TANKS')

Summary:

This Generic Letter transmitted I&E Bulletin 80-05 to licensees and requested they take action in regard to protecting their tanks from collapsing due to a vacuum condition.

Licensees were to ensure that adequate measures were taken to protect against vacuum conditions that could result in tank damage with the potential for release of radioactive material or detrimental effects with regard to overall safety of plant operations.

Potential Impact Number: 22498 **SRP Section Number:** 9.3.1

Document: NRC GENERIC LETTER 80-04
Title: OPERABILITY OF ADS VALVE PNEUMATIC SUPPLY

Summary:

This Generic Letter transmitted I&E Bulletin 80-01 to BWR licensees and requested that they take action in regard to the operability of the ADS system pneumatic air supply. The Generic Letter specified certain actions for GE BWR facilities that use a pneumatic operator for the ADS function.

Potential Impact Number: 22531 **SRP Section Number:** 6.6

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

'The ASME has published in ASME Section XI, Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." 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. Therefore, the PSI program for the ABWR should include provisions that ultrasonic testing be performed in accordance with Appendices VII and VIII pursuant to 10 CFR 50.55a(g)(3).'

Potential Impact Number: 22543 **SRP Section Number:** 6.6

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Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

'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." 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. Therefore, the applicant should revise the CESSAR to indicate that the Section XI requirements are to be augmented with the requirements in Appendices VII and VIII.'

Potential Impact Number: 22544 **SRP Section Number:** 6.6

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

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. Subsection IWH of ASME Section XI has not been incorporated in the code at the time of this review. However, ABB-CE in a submittal dated December 18, 1992, committed to review Subsection IWH and ensure that the System 80+ design PSI and ISI programs comply accordingly. Where the design will not practically accommodate full adherence to IWH, alternatives will be proposed in accordance with 10 CFR 50.55a(a)(3).

Potential Impact Number: 22547 **SRP Section Number:** 9.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.34 (f)(2)(viii)

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Summary:

(2) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage.....

(viii) Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID-14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 75 rems to the extremities, or, for licensees implementing the provisions of 20.1001-20.2401 of this chapter, 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)

This regulation was subsequently amended resulting in the following exposure limits: "... without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities." Reference PIs 4435, 22547 and 23692.

Potential Impact Number: 22548 **SRP Section Number:** 9.3.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

Provides recommendation for Commission approval of the following staff positions on ALWR post-accident sampling systems:

The staff recommends that the Commission approve the position that post-accident sampling systems for evolutionary and passive ALWRs of the pressurized water reactor type be required to have the capability to analyze dissolved gases and chloride, in accordance with the requirements of 10 CFR 50.34(f)(2)(viii) and Item II.B.3 of NUREG-0737. The time for taking these samples can be extended to 24 hours following the accident. For evolutionary and passive ALWRs of the boiling water reactor

type, there would be no need for the post-accident sampling system to analyze dissolved gases.

The staff recommends that the Commission approve the deviation from the requirements of Item II.B.3 of NUREG-0737 with regard to requirements for sampling reactor coolant for boron concentration and activity measurements using the post-accident sampling system in evolutionary and passive ALWRs.

The modified requirement would require the capability to take boron concentration samples and activity measurements 8 hours and 24 hours, respectively, following the accident.

Potential Impact Number: 22549 **SRP Section Number:** 9.3.2

Document: DRAFT FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

Provides the following staff position on the ABWR post-accident sampling system:

GE needs to design PASS to be capable of obtaining reactor water samples 24 hours after the end of power generation in order to evaluate concentrations of dissolved gases and chlorides in the reactor coolant. The information on the amounts of dissolved hydrogen, oxygen and chlorides in the reactor coolant is an important factor in evaluating post-accident conditions existing in the reactor vessel. This is Open Item 9.3.2.2-2.

Potential Impact Number: 22550 **SRP Section Number:** 9.3.2

Document: DRAFT SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

Provides the following staff position on the CE 80+ post-accident sampling system:

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The staff is currently seeking Commission approval of alternate requirements for 4 of the 11 clarifications in NUREG-0737, Issue II.B.3. The information is documented in Section II.I of an Enclosure in a letter, from D. M. Crutchfield of NRC to E. E. Kintner of GPU Nuclear Corporation, Parsippany, New Jersey, dated February 27, 1992. In response to Q281.43, the applicant stated that the postaccident sampling system is consistent with Issue II.B.3 in NUREG-0737. If the applicant meets all the requirements of Issue II.B.3 in NUREG-0737, the applicant will also meet the alternative requirements for the postaccident sampling system.

If the applicant chooses to adopt the alternative requirements, then the applicant should provide additional information on how to meet the alternative requirements and the remaining clarifications in Issue II.B.3 of NUREG-0737. In addition, the applicant should show how the degree of reactor core damage can be determined considering short-lived radioisotopes may have decayed before the sample was taken and analyzed (Open Item 9.3.2-2).

Potential Impact Number: 22557 **SRP Section Number:** 4.4

Document: FINAL SER EPRI CH 4
Title: EPRI URD-EPD: REACTOR SYSTEMS

Summary:

With regard to nuclear and T-H design, Section 4.2.1.6.2 of Chapter 4 specifies that the core design will 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 benefits of extending cycle operation must be weighed against the undesirability of decreasing core T-H stability.

The staff will address this matter during its review of an application for FDA/DC.

Potential Impact Number: 22558 **SRP Section Number:** 4.4

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

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The results of calculations to evaluate the thermal hydraulic stability of ABWR under the recirculation runback and feedwater runback conditions associated with ATWS events have been provided for staff review. ABWR design response to ATWS enables avoidance of large oscillations and the staff audit of stability calculations is complete. This is discussed in detail in Section 4.4 of this report (DFSER Open Item 4.4-1).

Potential Impact Number: 22559 **SRP Section Number:** 4.4

Document: FINAL SER ABWR CH 4
Title: REACTOR

Summary:

Core flow patterns are expected to be uniform at the core inlet during normal operations as a result of flow distributions from the downcomer through the RIPs, into the lower plenum, up through the orifices of the lower core plate, and into the fuel assemblies. TS will require for normal operation that at least 9 of the 10 RIPs are operating. Operation with fewer than 9 RIPs operating will require supporting analyses and justification by the COL applicant. This was DFSER TS Item 4.4-1. GE asserts that, with the allowed number of RIPs inoperable, pump operation will be close to normal and bounded by one recirculation loop operation in current jet pump BWRs (with, in effect, half the pumps out) for which there are no restrictions other than similar-type LOCA power-density restrictions. The restriction to no fewer than 9 RIPs operating is a recent modification of SSAR Section 4.4 and of the TS, and therefore, DFSER TS Item 4.4-1 is resolved. The restriction and specification are acceptable.

Potential Impact Number: 22560 **SRP Section Number:** 4.4

Document: FINAL SER EPRI CH 4
Title: EPRI URD-EPD: REACTOR SYSTEMS

Summary:

The staff made the following comments which EPRI was to consider in preparing future revisions of the Evolutionary Requirements Document: (1) Although the staff has found that thermal-hydraulic oscillations are not a problem for current PWRs, this finding may have to be revalidated if the thermal-hydraulic characteristics of the evolutionary PWRs

are sufficiently different from those of current PWRs; (2) It is current staff practice to allow PWRs to have axially unstable, but not radially unstable, xenon oscillations; and (3) The staff should make certain that the thermal-hydraulic stability and radial xenon stability characteristics of new designs have been established by testing during its reviews of individual applications for FDA/DC. This was identified as an open issue in the DSER for Chapter 4. In response to this DSER open issue, EPRI revised the Evolutionary Requirements Document to require that the thermal-hydraulic stability and xenon stability characteristics of new designs be verified by testing.

Potential Impact Number: 22562 **SRP Section Number:** 3.9.5

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Section 3.3 of Chapter 1 states that the plant design life for the ALWR will be 60 years. This proposed design life raises questions relative to the margins available in the current ASME fatigue design curves. These margins were established almost 30 years ago and were obtained from best-fit curves of fatigue test data by applying a factor of either 2 on stress or 20 on cycles, whichever was more conservative at each point. These factors were originally intended to cover such effects as environment, size effect, and scatter of data. However, on the basis of limited data currently available, the staff concludes that these margins may not be sufficient to account for variations in the original fatigue test data as a result of various environmental effects. In its letter dated May 17, 1991, the staff requested a commitment in Section 4.7.3 of Chapter 1 to consider such effects in the designs of applicable ASME Code Class 1 systems, components, and equipment. In its letter dated August 1, 1991, EPRI stated that if additional data or research results yield findings requiring changes to the current ASME fatigue design curves, the code consensus process will provide the proper vehicle to address such findings. The staff does not agree with all of the discussions in this response and concludes that the above commitment is not completely acceptable. The ASME Code curves may not be revised for many years. Therefore, the staff's position is that until these curves are revised, all ALWR applicants and all licensees applying for license renewal should propose appropriate fatigue design curves that will be reviewed by the staff.

The staff is assessing all available data relative to environmental effects on fatigue for austenitic stainless steel and ferritic steel. The objective of this effort is to propose

interim fatigue curves that may be used in lieu of the current ASME curves until ASME has acceptably addressed this concern. Pending staff implementation of these interim curves, the position stated above will be in effect.

Potential Impact Number: 22563 **SRP Section Number:** 4.4

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

In steady-state, two-phase, heated flow in parallel channels, the potential for hydrodynamic instability exists. ABB-CE gave the following information in the CESSAR-DC to support the contention that the System 80+ core is thermal hydraulically stable. According to the literature cited in CESSAR-DC, flow instabilities that have been observed occur almost exclusively in closed channel systems operating at low pressures relative to pressurized-water reactor (PWR) pressures. For PWR operating pressures, experimental results have shown that, even with closed channel systems, operating limits caused by critical heat flux are encountered before the flow stability threshold is reached. Kao, Morgan, and Parker (H.S. Kao, C.D. Morgan, and M.B. Parker, "Prediction of Flow Oscillation in Reactor Core Channel," American Nuclear Society Transactions, Vol. 16, p. 212) conducted flow-stability experiments at pressures up to 15.2 x 103 kPa (2200 psia) with closed, parallel, heated channels. They found that, at flow and power levels encountered in power reactors, no flow oscillations could be induced at pressures above 8.3 x 103 kPa (1200 psia). The low resistance to coolant cross-flow among subchannels of fuel assemblies would be expected to have a stabilizing effect, and this expectation is confirmed by experimental results by Veziroglu and Lee (T.N. Veziroglu and S.C. Lee, "Boiling-Flow Instabilities in a Cross-Connected Parallel-Channel or Flow System," American Society of Mechanical Engineers (ASME) Paper 71-HT-12, August 1971), who found that the cross-flow between parallel, heated channels improves flow stability. Experimental evidence that flow instabilities will not occur is given by data from rod bundle DNB tests conducted by ABB-CE (CENPD-162-P-A). Analytical support for the conclusion that flow instabilities will not reduce the thermal margin of ABB-CE's PWRs is contained in a letter to J.F. Stolz, (NRC) from D.H. Williams, Arkansas Power and Light Company, January 16, 1978, and enclosure, "Assessment of Core Flow Stability for Applicant PWRs," CEN-64(A)-P (proprietary) and CEN-64(A) (non- proprietary), July 1977. This document assesses core flow stability for a typical ABB-CE PWR using the CE-HYDNA code (H.B. Currin, et al., "HYDNA- Digital Computer Program for Hydrodynamic

Transients in a Pressure Tube Reactor or a Closed Channel Core," report CVNA-77, 1961). It was found that, for nominal coolant conditions, the flow is stable throughout the range of reactor power levels examined (100-percent to 250-percent rated power). On the basis of past operating experience, flow stability experiments, and inherent thermal-hydraulic characteristics of ABB-CE PWRs, the System 80+ design stability evaluation is adequate.

Potential Impact Number: 22564 **SRP Section Number:** 10.4.9

Document: FINAL SER CE80 CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

As part of the probabilistic risk assessment (PRA) submitted for the System 80+, the applicant performed a reliability analysis of the EFWS in accordance with the requirements of Item II.E.1.1 of NUREG-0737. The applicant evaluated the EFWS reliability for seven postulated transient and accident scenarios. Overall system unavailability for each of these cases was determined using the methodology described in NUREG/CR-2815 and NUREG/CR-2300, and using component failure rates from several generally accepted generic data bases. The staff primarily reviewed the applicant's reliability evaluation on the loss of main feedwater, loss-of-offsite power, and station blackout scenarios. The EFWS unavailability, as estimated in the applicant's PRA, meets the SRP acceptance criteria of 10-4 to 10-5 per demand. Only in the station blackout scenario is the EFWS unavailability greater than 10-4, at 3 x 10-3. Such a relatively small number for this scenario is due to the existence of two turbine-driven pumps in the System 80+ design. It should be noted that the applicant did not take credit for the startup feedwater pump. The only identified weakness of the analysis is that common-cause failures of the injection valves was not handled in a comprehensive and consistent manner for all of the cases analyzed. However, the incorporation of a more comprehensive and consistent valve common-cause model would not affect the conclusions of the analysis.

Potential Impact Number: 22565 **SRP Section Number:** 9.5.5

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

APPENDIX II
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Summary:

The diesel generator cooling water system conforms with Position C.7 of RG 1.9 as it relates to engine cooling water protective interlocks. All trips are bypassed during LOCA conditions except low cooling water pressure and low differential pressure of secondary cooling water. Both of these trips are 2-out-of-2 logic trips.

Potential Impact Number: 22568 **SRP Section Number:** 5.2.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

In its DSER for Chapter 5 (February 1990), the staff concluded that EPRI's proposed resolution for Issue 70 was consistent with the staff's positions. However, the staff concluded that the Evolutionary Requirements Document should specify a criterion for the depressurization rate of the SDVS. In Section 6.6.4.3 of Chapter 5, EPRI specifies that the SDVS will be capable of depressurizing the reactor coolant system below 250 psig before reactor vessel melt-through as a means of precluding containment challenges through direct containment heating.

The staff has resolved Issue 70 and has issued GL 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" (requiring technical specifications revisions at PWRs with PORVs and block valves). It has also drafted revisions of the Standard Review Plan (Sections 3.2.2, "System Quality Group Classifications"; 5.2.2, "Overpressure Protection"; and 5.4.7, "Residual Heat Removal System").

The low-temperature overpressure protection (LTOP) function for the ALWR PWRs is to be provided by the residual heat removal pressure-relief system. In a letter to the NRC dated March 28, 1988, EPRI committed to revise appropriate chapters of the Evolutionary Requirements Document to provide information on LTOP. The staff has confirmed the incorporation of this information in Section 3.3.2 of Chapter 3 of the Evolutionary Requirements Document, and the staff evaluation is documented in Section 3.3 of Chapter 3 of this report. Therefore, this confirmatory issue is resolved.

APPENDIX II
Potential Impacts

Potential Impact Number: 22570 **SRP Section Number:** 5.2.2

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

Issue 94, in NUREG-0933, addressed low-pressure overpressurization events since the resolution of Issue A-26. Therefore, Issue 94 was to address the additional guidance for RCS low-temperature overpressure protection (LTOP) to ensure reactor vessel integrity beyond the requirements specified for Issue A-26 in SRP Section 5.2.2, "Overpressure Protection," and BTP RSB 5-2, "Overpressure Protection of Pressurized Water Reactors While Operating at Low Temperature." Issue 94 was resolved with the additional requirements to have the TSs for overpressure protection consistent with those specified in Enclosure B to GL 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)."

In the CESSAR-DC, ABB-CE describes the LTOP design and its compliance with the requirements of SRP Section 5.2.2 and BTP RSB 5-2. The staff has reviewed and approved, in Section 5.2.2 of the FSER, the LTOP design for the System 80+ plant. The TSs for LTOP are in CESSAR-DC Sections 16.7.11 and 16A.7-11. The LTOP function is performed by the relief valves in the shutdown cooling system (SCS). The analysis to support the adequacy of the LTOP is approved as indicated in Section 5.2.2 of the FSER.

In the DSER, ABB-CE was requested by the staff to revise the TSs for LTOP by reducing allowable outage time for a single channel from 7 days to 24 hours when the plant is operating in Modes 5 or 6, in accordance with Enclosure B to GL 90-06. ABB-CE revised surveillance requirements specified in the TSs in CESSAR-DC Section 16.7.11, to be consistent with Enclosure B to GL 90-06.

ABB-CE has provided an acceptable LTOP design and the associated TSs are consistent with the requirements of BTP RSB 5-2 and Enclosure B to GL 90-06; Therefore, Issue 94 is resolved for the System 80+ design.

Potential Impact Number: 22573 **SRP Section Number:** 13.1.1

APPENDIX II
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Document: C&S: ANSI N18.1
Title: Selection, Qualification and Training of Personnel for Nuclear Power Plants

Summary:

Section 4.6.1 of ANSI N18.1 (as endorsed by Reg. Guide 1.8) is cited as Acceptance Criteria for the qualifications of the applicant's corporate management and technical support organization 'Engineer in Charge', in SRP Section 13.1.1, subsection II.B.6.

The version of the standard endorsed by Reg. Guide 1.8 is ANSI N18.1-1971. The current version of the standard is ANSI/ANS 3.1-1987.

Potential Impact Number: 22574 **SRP Section Number:** 13.1.1

Document: C&S: ANS 3.1
Title: Selection, Qualification and Training of Personnel for Nuclear Power Plants

Summary:

Section 4.6.1 of ANSI N18.1 (as endorsed by Reg. Guide 1.8) is cited as Acceptance Criteria for the qualifications of the applicant's corporate management and technical support organization 'Engineer in Charge', in SRP Section 13.1.1, subsection II.B.6.

The version of the standard endorsed by Reg. Guide 1.8 is ANSI N18.1-1971. The current version of the standard is ANSI/ANS 3.1-1987.

Potential Impact Number: 22600 **SRP Section Number:** 11.2

Document: NRC GENERIC LETTER 80-21
Title: VACUUM CONDITION RESULTING IN DAMAGE TO CHEMICAL VOLUME CONTROL SYSTEM (CVCS) HOLDUP TANKS (SOMETIMES CALLED 'CLEAN WASTE RECEIVER TANKS')

Summary:

This Generic Letter transmitted I&E Bulletin 80-05 to licensees and requested they take action in regard to protecting their tanks from collapsing due to a vacuum condition.

Licensees were to ensure that adequate measures were taken to protect against vacuum conditions that could result in tank damage with the potential for release of radioactive material or detrimental effects with regard to overall safety of plant operations.

Potential Impact Number: 22608 **SRP Section Number:** 10.4.9

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

The basic requirements for the pressurized water reactor manufactured by Combustion Engineering are specified in paragraphs (c)(1) and (c)(2) of 10 CFR 50.62 (ATWS Rule) which state, in part: "Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater and initiate a turbine trip under conditions indicative of an ATWS. [paragraph (c)(1)] and must have a diverse scram system from the sensor output to interruption of power to the control rods. [paragraph (c)(2)]. "

Potential Impact Number: 22613 **SRP Section Number:** 9.5.5

Document: C&S: IEEE 387
Title: IEEE STANDARD CRITERIA FOR DIESEL-GENERATOR UNITS APPLIED AS STANDBY POWER SUPPLIES FOR NUCLEAR POWER GENERATING STATIONS

Summary:

This Standard describes the criteria for the application of diesel-generator units as individual units of the Class 1E standby power supplies in stationary nuclear power generating stations. The cooling water system is within the scope of the standard. Note: The current Standard is dated November 21, 1984.

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Potential Impact Number: 22614 **SRP Section Number:** 9.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue addresses concerns about reactor coolant pump seal failures that could cause a SBLOCA. This issue has a high priority.

Potential Impact Number: 22615 **SRP Section Number:** 9.2.2

Document: NRC GENERIC LETTER 91-13
Title: REQUEST FOR INFORMATION RELATED TO THE RESOLUTION OF GENERIC ISSUE 130, 'ESSENTIAL SERVICE WATER SYSTEM FAILURES AT MULTI-UNIT SITES,' PURSUANT TO 10 CFR 50.54(f)

Summary:

Enclosure 3 provides information about the staff's view of backup RCP seal cooling systems.

"The specific features of such a backup, dedicated RCP seal cooling system would be as follows:
- Single high pressure pump, 50-100 gpm capacity
- Dedicated water storage tank with capacity to last at least 8-10 hours
- AC-independent (non-seismic) pump
- No support system cooling required
- Once-through RCP seal heat removal
...
Generic Issue 23, 'Reactor Coolant Pump Seal Failures,' deals with this recommendation also, and specific guidance for resolving that generic issue is given in proposed Regulatory Guide DG-1008. While awaiting completion of public review and comment on draft Regulatory Guide DG-1008, resolution of this GI-130 item has been deferred until GI-23 is resolved."

Potential Impact Number: 22616 **SRP Section Number:** 9.2.2

APPENDIX II
Potential Impacts

Document: DRAFT FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

"The staff required GE to confirm that the failure of the following systems will not cause a LOCA.

- recirculation motor cooling system
- recirculation motor seal purge system
- recirculation motor inflatable shaft seal subsystem"

Potential Impact Number: 22617 **SRP Section Number:** 9.2.2

Document: DRAFT SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

"The supply and return headers for the RCP support systems do not isolate on an safety injection actuation signal (SIAS) and will be supplied with cooling water following a small-break LOCA with a LOOP in accordance with the requirements of 10 CFR 50.34(f)(1)(iii). Low- and high-flow alarms are provided for the various RCP heat exchangers which use CCWS for cooling. These alarms alert the control room of flow anomalies associated with the heat exchangers and ensure that timely notification of the cooling problems is provided to protect the RCP pumps."

Potential Impact Number: 22620 **SRP Section Number:** 3.5.1.3

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The probability of unacceptable damage from turbine missiles is expressed (in SRP Section 3.5.1.3 Review Procedure III.2) as the product of (1) the probability of turbine missile generation resulting in the ejection of turbine disk (or internal structure)

fragments through the turbine casing (P1); (2) the probability of ejected missiles perforating intervening barriers and striking safety-related structures, systems, or components (P2); and (3) the probability of struck structures, systems, or components failing to perform their safety functions (P3).

In previous reviews of the probability calculation, the staff found that the mathematical models that were used to calculate P2 and P3 require numerous approximations and simplifying assumptions in order to incorporate available data in the analysis. As a result, the calculations of P2 and P3 were not accurate; however, the calculation of P1 is considered more accurate and precise. Therefore, in recent years, the staff placed emphasis on reviewing the probability of turbine missile generation (P1). The staff believes that if P1 is controlled to a minimum, the final failure probability of unacceptable damage can be met. Consistent with the staff position taken in recently licensed nuclear plants, the probability of turbine missile generation should be kept to no greater than 10-5 per reactor-year for an unfavorably oriented turbine and 10-4 for a favorably oriented turbine. The staff also recommended certain actions for situations when the probability does not meet the minimum values required in SER Table 3.1. The staff recommended in its DSER that GE include the minimum requirement for the probability of turbine missile generation in SSAR Table 3.1.

Potential Impact Number: 22621 **SRP Section Number:** 3.5.1.3

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

GE prepared a proprietary report in January 1984 entitled, 'Probability of Missile Generation in General Electric Nuclear Turbines.' The staff reviewed the report and found the methodology acceptable for calculating the probability of turbine missile generation (NUREG-1048, Supplement 6). Various parameters are considered in the methodology that influence the outcome of the turbine missile generation probability, such as turbine disk design, disk material properties, turbine speed control systems, postulated crack growth rate, and inspection intervals. On the basis of parametric variations, the methodology can optimize turbine disk inspection intervals so that the turbine missile generation probability will be kept to the allowable limit. The staff believes that by placing emphasis on the turbine disk inspection program, which includes a specified inspection interval, and on an effective turbine control system maintenance program, the turbine will be operated in a safe manner.

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Consistent with the staff position taken in recently licensed nuclear plants, the COL applicant must submit for NRC approval, within 3 years of obtaining a COL, a turbine system maintenance program including probability calculations of turbine missile generation based on the methodology approved by the NRC, or commit to volumetrically inspect all low-pressure turbine rotors at the second refueling outage and every other (alternate) refueling outage thereafter until a maintenance program is approved by the staff.

Potential Impact Number: 22622 **SRP Section Number:** 3.5.1.3

Document: FINAL SER EPRI CH 13
Title: EPRI URD-EPD: MAIN TURBINE-GENERATOR SYSTEMS

Summary:

In the DSER for Chapter 13, the staff stated that the ALWR low-pressure turbine(s) should meet the specific failure probability of turbine-missile generation related to specific turbine orientation. 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. A turbine is favorably oriented if any plane perpendicular to the turbine-generator axis is not intersected with the primary containment structure. These probabilities originated from the staff's review of Westinghouse and General Electric topical reports on analytical procedures for turbine-missile analyses and were used for recently licensed operating plants. The staff approved the Westinghouse methodology in its SER, 'Extension of Turbine Disc Inspection Schedule for Indian Point 3 Nuclear Generating Station,' February 2, 1987, and the General Electric methodology in NUREG-1048, 'Safety Evaluation Report Related to the Operation of Hope Creek Generating Station,' Supplement No. 6, July 1986.

EPRI's proposed probability of unacceptable damage to safety-related systems and components of 1.0E-7 per year satisfies RG 1.115, 'Protection Against Low-Trajectory Turbine Missiles,' and is acceptable. However, in the DSER for Chapter 13, the staff concluded that the ALWR turbine(s) should also satisfy the failure probability of missile generation of either 1.0E-4 or 1.0E-5 per reactor year, depending on the turbine orientation, and identified this as an open issue.

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Potential Impact Number: 22623 **SRP Section Number:** 3.5.1.3

Document: FINAL SER EPRI CH 13
Title: EPRI URD-EPD: MAIN TURBINE-GENERATOR SYSTEMS

Summary:

Section 3.2.1 of Chapter 13 also specifies variation in duty cycle that can affect the long-term integrity of the turbine. Examples include operation without reheat, changes from full to partial arc admission, load following, and isolation of feedwater heater strings. These can create fatigue effects, change the location of the point where moisture forms, increase the amount of condensation, or modify stage flows. The staff will consider these issues when it reviews an individual application for FDA/DC. In addition, applicants should consider the fracture toughness properties of the turbine casing material in specifying the turbine design. This consideration should reflect the potential effects of the environment in which the turbine will operate. This fracture toughness consideration is a measure to mitigate the effects of turbine-missile-generating events. The staff will review individual applications for FDA/DC to ensure that appropriate materials have been selected for the turbine casing.

Potential Impact Number: 22624 **SRP Section Number:** 10.4.9

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

The Auxiliary Feedwater Actuation Signal (AFAS) will start emergency feedwater (EFW) to a steam generator when the water in that steam generator decreases below a predetermined level. Its circuitry is diverse from that of the RPS. The EFW (pumps and valves) is actuated by sending isolated AFAS signals to the ESFAS. The level of diversity for this feature is not acceptable because the APS uses the ESF component control logic to control the EFWS. The AFAS must be completely diverse from the Plant Protection System. The staff concluded in Section 7.7 of this DSER that the ATWS implementation proposed by the applicant is not acceptable. The applicant must demonstrate, and the staff must confirm, the diversity of this system design.

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Potential Impact Number: 22625 **SRP Section Number:** 3.5.1.3

Document: DRAFT SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The applicant indicates that the combined probability of turbine missile generation and adverse impact effects on seismic Category I systems and components is less than 10-4 per year. However, the staff's position requires that the turbine maintenance and inspection program as discussed in Section 10.2 of this DSER be implemented to ensure that the failure and missile generation probability be less than 10-4 per year for a favorably oriented turbine system (Refs. 1 and 2). The applicant should revise CESSAR to be consistent with this staff position (Open Item 3.5.1.3-1). The applicant has sufficiently demonstrated to the staff, in accordance with RG 1.115, that the probability of turbine missile damage to structures, systems, and components important to safety is acceptably low. Therefore, the staff concludes that the turbine missile risk for the proposed plant design is acceptable and meets the requirements of GDC 4, subject to the resolution of the above open item.

The COL applicant should verify that its turbine maintenance and inspection program will ensure that the failure and missile generation probability will be less than 10-4 per year. The COL applicant should submit a summary of the turbine maintenance and inspection program and the results of the probabilistic evaluation for staff review (COL Action Item 3.5.1.3-1).

References

(1) 'Safety Evaluation Report Related to the Operation of Hope Creek Generating Station,' NUREG-1048, Supplement No. 6, July 1986.

(2) Letter from C. E. Rossi of NRC to J. A. Martin of Westinghouse, Orlando, Florida, dated February 2, 1987.

Potential Impact Number: 22626 **SRP Section Number:** 10.4.9

Document: FINAL SER CE80 CH 10

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Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

The EFWS can be actuated either manually or automatically by the emergency feedwater actuation system or the alternate protection system. The initiation signal starts all auxiliary feedwater pumps and supporting systems, aligns the auxiliary feedwater sources, and opens flow paths from the auxiliary feedwater pumps to the steam generator.

Potential Impact Number: 22627 **SRP Section Number:** 3.5.1.3

Document: NUREG 1048 SUP 6
Title: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF HOPE CREEK GENERATING STATION

Summary:

Supplement No. 6 to the Safety Evaluation Report on the application filed by Public Service Electric and Gas Company on its own behalf as co-owner and as agent for the other co-owner, the Atlantic City Electric Company, for a license to operate Hope Creek Generating Station has been prepared by the Office of Nuclear Regulation of the U.S. Nuclear Regulatory Commission. This supplement reports the status of certain items that had not been resolved at the time of the publication of the Safety Evaluation Report. This supplement supports the issuance of a full-power license to operate the Hope Creek Generating Station.

The advanced reactor program SERs cite this NUREG supplement as the source regulatory document for staff positions related to achieving acceptable probability of generation of turbine missiles through a prescribed turbine maintenance program.

Microfiche No. 37530:309

Potential Impact Number: 22631 **SRP Section Number:** 6.4

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

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Summary:

The staff considers EPRI's proposed resolution of Issue 83 to be complete, subject to the staff's plant-specific reviews, as applicable, of individual applications for FDA/DC. During its reviews, the staff will ensure that the existing minimum requirements have been met, by conforming to the guidance of SRP Sections 6.4 and 9.4.1 (to meet GDC 19). Appropriate requirements are identified in the Evolutionary Requirements Document that include conformance with the applicable regulatory guides (RGs 1.52, 1.78, and 1.95) and industry standards (ANSI/ANS-59.2, ANSI/ASME N509, and ANSI/ASME N510). However, during its plant-specific reviews, the staff will also verify conformance with industry standards (ASME Code AG-1, 'Code on Nuclear Air and Gas Treatment,' and American Society for Testing and Materials (ASTM) D3803, 'Standard Test Methods for Radiological Testing of Nuclear-Grade Gas-Phase Absorbents') and evaluation methodology and verify that technical specifications and surveillance requirements for the control room habitability system as a whole to meet GDC 4, 5, and 19 have been proposed. This 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 (NUREG-0737). Additionally, the FDA/DC applicant will have to ensure that the control room habitability design meets GDC 4, 5, and 19 and that operators are protected in accordance with TMI Action Plan Item III.D.3.4.

EPRI states that the control room HVAC system and other specific areas will be designed in accordance with National Fire Protection Association (NFPA) 90A, 'Installation of Air Conditioning and Ventilating Systems.' As stated in Section 3.3.1 of Chapter 9, by committing to NFPA 90A, EPRI has committed to use the HVAC systems for removing smoke from specific areas as a means of satisfying the smoke control provisions of NRC fire protection guidelines. The staff will evaluate the design, installation, and operation of the HVAC systems functioning in the smoke removal mode during its review of individual applications for FDA/DC.

Potential Impact Number: 22632 **SRP Section Number:** 9.4.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The staff considers EPRI's proposed resolution of Issue 83 to be complete, subject to the staff's plant-specific reviews, as applicable, of individual applications for FDA/DC.

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During its reviews, the staff will ensure that the existing minimum requirements have been met, by conforming to the guidance of SRP Sections 6.4 and 9.4.1 (to meet GDC 19). Appropriate requirements are identified in the Evolutionary Requirements Document that include conformance with the applicable regulatory guides (RGs 1.52, 1.78, and 1.95) and industry standards (ANSI/ANS-59.2, ANSI/ASME N509, and ANSI/ASME N510). However, during its plant-specific reviews, the staff will also verify conformance with industry standards (ASME Code AG-1, 'Code on Nuclear Air and Gas Treatment,' and American Society for Testing and Materials (ASTM) D3803, 'Standard Test Methods for Radiological Testing of Nuclear-Grade Gas-Phase Absorbents') and evaluation methodology and verify that technical specifications and surveillance requirements for the control room habitability system as a whole to meet GDC 4, 5, and 19 have been proposed. This 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 (NUREG-0737). Additionally, the FDA/DC applicant will have to ensure that the control room habitability design meets GDC 4, 5, and 19 and that operators are protected in accordance with TMI Action Plan Item III.D.3.4.

EPRI states that the control room HVAC system and other specific areas will be designed in accordance with National Fire Protection Association (NFPA) 90A, 'Installation of Air Conditioning and Ventilating Systems.' As stated in Section 3.3.1 of Chapter 9, by committing to NFPA 90A, EPRI has committed to use the HVAC systems for removing smoke from specific areas as a means of satisfying the smoke control provisions of NRC fire protection guidelines. The staff will evaluate the design, installation, and operation of the HVAC systems functioning in the smoke removal mode during its review of individual applications for FDA/DC.

Potential Impact Number: 22633 **SRP Section Number:** 3.5.1.4

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The regulatory positions of Reg. Guide 1.76 have been reevaluated using more recent tornado data. The reevaluation is documented in NUREG/CR-4461, 'Tornado Climatology of the Contiguous United States,' dated May 1, 1986. The reevaluation found that the tornado strike probabilities range from near 10E-7 per year for much of the Western United States to about 10E-3 per year in the central United States. The wind speeds associated with a tornado having a strike probability of 10E-7 range from

less than 246 km/hr (153 mph) to 534 km/hr (332 mph). These wind speed estimates are 48 to 161 km/hr (30 to 100 mph) lower than the speed estimates presented in WASH-1300 and RG 1.76 for most of the United States. From the reevaluation the staff concluded that it would be reasonable to reduce DBT wind speeds to 322 km/hr (200 mph) for the United States west of the Rocky Mountains and to 483 km/hr (300 mph) for the United States east of the Rocky Mountains. The revised tornado parameters are discussed in Section 2.3.1 of the ABWR FSER. GE used the modified RG 1.76 guidelines in identifying the DBT as 483 km/hr (300 mph) with a strike probability of 10E-7 per year.

Also in the DFSER, the staff noted that the COL applicant should identify missiles generated by other site-specific natural phenomena that might be more limiting than those considered in the ABWR design and should provide protection for the SSCs against such missiles. GE addressed this item by stating that the COL applicant shall identify missiles generated by other site-specific natural phenomena that may be more limiting than those considered in the ABWR design and shall provide protection for the SSCs against such missiles.

Potential Impact Number: 22635 **SRP Section Number:** 3.5.1.4

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

Based on the updated tornado strike data and analysis provided in NUREG/CR-4461, the staff concludes that it is acceptable to reduce the design-basis tornado wind speed to 322 km/hr (200 mph) for states west of the Rocky Mountains and to 482 km/hr (300 mph) for states east of the Rocky Mountains.

The staff recommends that the Commission approve the position that a maximum tornado wind speed of 482 km/hr (300 mph) be used in the design basis tornado employed in the design of evolutionary and passive ALWRs. The COL applicant will have to demonstrate that a design capable of withstanding a 482 km/hr (300 mph) tornado will also be sufficient to withstand other site-specific hazards.

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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.

Potential Impact Number: 22636 **SRP Section Number:** 3.5.1.4

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The regulatory positions in RG 1.76 were reevaluated by an NRC contractor using the considerable quantity of tornado data that are now available but were not when the regulatory guide was developed. The contractor's reevaluation is provided in NUREG/CR-4461, 'Tornado Climatology of the Contiguous United States,' dated May 1986.

The contractor found that the tornado strike probabilities range from nearly 1.0E-7 per year for much of the Western United States to about 1.0E-3 per year in the Central United States. On the basis of discussions between the contractor and the staff, wind speed values associated with a tornado having a mean recurrence interval of 10E-7 per year were estimated to be about 200 mph for the United States west of the Rocky Mountains and 300 mph for the United States east of the Rocky Mountains.

EPRI proposed that a maximum tornado wind speed of 300 mph and a tornado recurrence interval of 1.0E-7 per year tornado strike probability be used for the design-basis tornado in the design of the evolutionary ALWRs.

In the draft Commission paper on evolutionary and passive plant policy issues dated February 27, 1992, the staff stated that it will accept the tornado design basis of 300 mph recently proposed by EPRI. Until the staff resolves this issue with the Commission, it considers this an open issue that must be resolved before it can complete its review of Chapter 1.

Potential Impact Number: 22644 **SRP Section Number:** 3.3.1

Document: DRAFT SER CE80 CH 3

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Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The plant design with respect to capability of the structures to withstand design wind loadings is acceptable and meets the requirements of GDC 2....

These requirements are being met by the use of ANSI A58.1-1982 and ASCE Papers 3269 and 4933 to transform the wind velocity into an effective pressure on structures and for selecting pressure coefficients corresponding to the structural geometry and physical configuration...

The procedures used to determine the design wind loadings on structures for the plant are acceptable.

Potential Impact Number: 22648 **SRP Section Number:** 3.3.1

Document: C&S: ANSI A58.1
Title: Minimum Design Loads in Buildings and Other Structures

Summary:

Standard ANSI A58.1 is cited (version not specified) in SRP Section 3.3.1 as guidance for transforming design basis wind velocities into effective structural pressure loads. ANSI A58.1-1972 was the version in effect at the time of issuance of SRP Section 3.3.1.

The CE System 80+ DSER and the EPRI Evolutionary Plant FSER cite ANSI A58.1-1982 in conjunction with staff reviews of design basis wind velocities and their associated effective structural loads.

Potential Impact Number: 22649 **SRP Section Number:** 3.3.2

Document: C&S: ANSI A58.1
Title: Minimum Design Loads in Buildings and Other Structures

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Summary:

Standard ANSI A58.1 is cited (version not specified) in SRP Section 3.3.2 as guidance for transforming design basis tornado wind velocities into effective structural pressure loads. ANSI A58.1-1972 was the version in effect at the time of issuance of SRP Section 3.3.2.

The CE System 80+ DSER and the EPRI Evolutionary Plant FSER cite ANSI A58.1-1982 in conjunction with staff reviews of design basis tornado wind velocities and their associated effective structural loads.

Potential Impact Number: 22650 **SRP Section Number:** 3.3.2

Document: C&S: ASCE 7-88
Title: Minimum Design Loads for Buildings and Other Structures

Summary:

Standard ANSI A58.1 is cited (version not specified) in SRP Section 3.3.2 as guidance for transforming design basis tornado wind velocities into effective structural pressure loads. ANSI A58.1-1972 was the version in effect at the time of issuance of SRP Section 3.3.2.

The CE System 80+ DSER and the EPRI Evolutionary Plant FSER cite ANSI A58.1-1982 in conjunction with staff reviews of design basis tornado wind velocities and their associated effective structural loads.

ANSI A58.1 has been revised and redesignated as ANSI/ASCE-7-1988.

Potential Impact Number: 22651 **SRP Section Number:** 3.3.1

Document: C&S: ASCE 7-88
Title: Minimum Design Loads for Buildings and Other Structures

Summary:

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Standard ANSI A58.1 is cited (version not specified) in SRP Section 3.3.1 as guidance for transforming design basis wind velocities into effective structural pressure loads. ANSI A58.1-1972 was the version in effect at the time of issuance of SRP Section 3.3.1.

The CE System 80+ DSER and the EPRI Evolutionary Plant FSER cite ANSI A58.1-1982 in conjunction with staff reviews of design basis wind velocities and their associated effective structural loads.

ANSI A58.1 has been revised and redesignated as ANSI/ASCE-7-1988.

Potential Impact Number: 22656 **SRP Section Number:** 3.3.2

Document: DRAFT SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The procedures used to transform the tornado wind velocity into pressure loadings are the same as for the winds discussed in Section 3.3.1 of this DSER (ANSI A58.1-1982 and ASCE Papers 3269 and 4933)....

The applicant meets the guidelines of SRP Section 3.3.2 with respect to the structural capability to withstand design tornado wind loading and tornado missiles.

The individual COL applicant must ensure that tornado parameters... presented in CESSAR Table 2.0-1 shall not be exceeded by those of the site-specific design basis tornado (COL Action Item 3.3.2-1).

Potential Impact Number: 22660 **SRP Section Number:** 3.9.1

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

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The list of transients in SSAR Table 3.9.1 appear to be based on a 40-year life. For a design life of 60 years, the number of cycles for each transient shall be increased by a factor of 1.5 and be applicable to all safety-related systems and components.

Potential Impact Number: 22662 **SRP Section Number:** 3.9.1

Document: DRAFT SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The staff observed that the transients and the number of cycles for the transients specified for the 60-year design life System 80+ plant are the same as those identified for the 40-year design life System 80 plant. The applicant was asked to justify the use of the same number of cycles for the System 80+ design as was used for the System 80 design. The applicant responded by committing to revise the CESSAR to include the events and frequency of occurrence expected during the 60-year design life.

Potential Impact Number: 22664 **SRP Section Number:** 9.2.1

Document: FINAL SER EPRI CH 8
Title: EPRI URD-EPD: PLANT COOLING WATER SYSTEMS

Summary:

Many operating plants do not have the capability to test the heat load of the heat exchangers. It has been determined that adding this heat load testing capability after the plant is already in operation might not be economically feasible. In the DSER for Chapter 8, the staff concluded that vendors should consider including the capability for performance testing of key heat exchangers during the planning and design stage of the ALWR plants. Enclosure 2 ("Program for Testing Heat Transfer Capability") of GL 89-13 should be consulted and incorporated in the Evolutionary Requirements Document as appropriate.

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Potential Impact Number: 22665 **SRP Section Number:** 9.2.1

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The RSW system picks up heat from the RCW heat exchangers and rejects the heat to the UHS, which is to be designed by individual applicants referencing the ABWR design. The GE scope of the RSW includes all the piping, valves, heat exchangers, instrumentation, and controls within the control building. All other equipment outside the control building, including the RSW pumps, are designated as interface items and are the responsibility of the COL applicant.

Potential Impact Number: 22668 **SRP Section Number:** 6.5.1

Document: FINAL SER EPRI CH 9
Title: EPRI URD-EPD: SITE SUPPORT SYSTEMS

Summary:

The EPRI evolutionary FSER contains a summary of key requirements for safety-related HVAC systems. Two of these requirements are stated as follows: "(1) All HVAC units and components and supports/hangers will be designed, constructed, and installed in accordance with ANSI/ASME Standard AG-1, 'Code on Nuclear Air and Gas Treatment,' (1989), and ANSI/ASME standard N509, 'Nuclear Power Plant Air Cleaning Units and Components,' (1989), and (2) HVAC systems will be designed to permit appropriate inservice inspections and functional testing in accordance with ANSI/ASME Standard N510, 'Testing of Nuclear Air Cleaning Systems' (1989)."

In the DSER for Chapter 9, the staff concluded that this requirement should be imposed on not only the filters, but also the filtration systems, and recommended that the requirement be revised to read "all nuclear air treatment filtration systems shall be designed....," instead of "all nuclear air treatment filters shall be designed...." This was identified as an open issue in the DSER. EPRI has revised Section 8.2.1.1.8 of Chapter 9 to require application of the regulatory guides and the referenced standards to all nuclear air treatment filtration systems, not just the filters. The staff concludes that these revisions are acceptable. Therefore, this DSER open issue was closed.

However, because portions of ANSI/ASME AG-1-1988, including rules for the design of HVAC ductwork, are still being prepared, the staff has not yet fully endorsed this standard.

In the interim, the staff concludes that 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 that the staff deems are acceptable. In Revision 2 of Regulatory Guide 1.52 (March 1978), the staff recommends that ductwork be designed, constructed, and tested in accordance with Section 5.10 of ANSI/ASME N509.

Potential Impact Number: 22676 **SRP Section Number:** 6.2.5

Document: DRAFT SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

In response to the TMI-2 accident, the Commission promulgated regulatory requirements on hydrogen control in 10 CFR 50.34 and 50.44. A hydrogen control system is required by 10 CFR 50.34(f) 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 postaccident atmosphere that will not support hydrogen combustion.

In Section 3.5.3 of Appendix B to Chapter 1 of the URD for Evolutionary Plant Designs, EPRI proposes that the evolutionary plant design have the capability to ensure that necessary accident prevention and mitigation functions can be performed during and after events in which hydrogen is produced and recommended that the design be based on 75 percent cladding oxidation. In Appendix B of its DSER for Chapter 5 (February 1990), the staff concluded that EPRI's proposed approach to resolving this issue was unacceptable.

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). This regulation specifically requires a hydrogen control system that can safely accommodate an amount of hydrogen equivalent to that generated by the reaction of 100 percent of the fuel-clad metal and that can ensure that uniformly distributed hydrogen gas concentrations in the containment do not exceed 10 percent by volume.

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The Commission approved the staff's recommendation in a staff requirements memorandum of June 26, 1990.

The CESSAR is in compliance with 10 CFR 50.34(f). However, since the staff position in SECY-90-016 is that the plant-specific design must comply with 10 CFR 50.34(f) for combustible gas control, the staff will review relevant design features for conformance to 10 CFR 50.34(f) during its review of individual applications for the System 80+ plant (COL Action Item 20.2-7).

Potential Impact Number: 22678 **SRP Section Number:** 6.2.5

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

Because hydrogen control is necessary, given present analytical capabilities, to preclude local concentrations of hydrogen to detonable limits, the staff concludes ALWRs should provide containment-wide hydrogen control (e.g., igniters, inerting) for severe accidents. Additional advantages of providing hydrogen control mitigation features (rather than relying on random ignition of richer mixtures) include the lessening of pressure and temperature loadings on the containment and essential equipment.

Because of the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, the staff stated that, as a minimum, ALWRs should be designed to (1) accommodate hydrogen equivalent to 100-percent metal-water reaction of the fuel cladding and (2) limit containment hydrogen concentration to no greater than 10 percent. The staff's position is consistent with the requirements of 10 CFR 50.34(f) as referenced in 10 CFR Part 52.

The staff concluded that the Evolutionary Requirements Document should provide appropriate guidance regarding the timing of igniter activation in the event of an accident.

In Section 6.5.3.1.6 of Revision 1 to Chapter 5, EPRI states that the designer will define criteria (including timing) for manual initiation and shutdown of the hydrogen igniter system. Initiation will precede possible hydrogen generation and shutdown should follow verification of a safe, stable plant condition. This position appears to be

consistent with the position in SECY-90-016 that a deliberate system should be provided that will control the combustible gas.

Potential Impact Number: 22679 **SRP Section Number:** 6.2.5

Document: DRAFT FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

10 CFR 52.47, Item (ii) requires an application for a standard design certification to include a demonstration of compliance with any technically relevant portions of the TMI requirements identified in 10 CFR 50.34(f). 10CFR 50.34(f)(2)(ix) requires provision for a hydrogen control system that can safely provide assurance that uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent during and following an accident that releases an amount of hydrogen equivalent to a 100-percent fuel-clad metal- water reaction or that the post-accident atmosphere will not support hydrogen combustion. In addition, equipment 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 equivalent to that generated by a 100-percent fuel-clad metal-water reaction, including the environmental conditions created by the actuation of the hydrogen control system. In SSAR Section 19B.2.6, GE states that the ABWR containment will have an inert atmosphere and will be able to withstand the pressure and energy addition from 100-percent fuel-clad metal-water reaction. The inerted containment and the provision for permanently installed hydrogen recombiners are acceptable as hydrogen control measures and adequately address this TMI issue.

Potential Impact Number: 22682 **SRP Section Number:** 6.5.1

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

Section 8.2.1.2.7 of Chapter 9 in Volume II of the Requirements Document specifies that the HVAC components and supports will be designed, constructed, and installed in

accordance with ANSI/ASME AG-1-1988, and ANSI/ASME N509. Portions of ANSI/ASME AG-1-1988, including rules for the design of HVAC ductwork are still under preparation. Therefore, the staff has not yet fully endorsed this standard. Until this standard has been fully endorsed, the staff's interim position is that Article AA-4000 in the 1988 revision of the AG-1 Standard provides acceptable minimum design requirements for the structural design of HVAC equipment and supports. For HVAC ductwork, in Revision 2 to RG 1.52, (March 1978), the staff recommends that ductwork should be designed, constructed, and tested in accordance with the provisions of Section 5.10 of ANSI N509.

The staff concludes that the design criteria required in Volume II, Chapter 9, Section 8.2.1, of the Requirements Document are consistent with current staff positions and provide an acceptable minimum design basis for ensuring that HVAC components and supports will withstand the most adverse combination of loading events without loss of structural integrity. However, the staff will continue to review the acceptability of the detailed HVAC design criteria that are being developed for the AG-1 Standard.

Potential Impact Number: 22684 **SRP Section Number:** 3.5.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

In the DSER (SECY 91-153), the staff identified the need for an interface requirement to protect safety-related equipment from failures of non-safety-related SSCs not housed in tornado-resistant buildings or structures (Outstanding Issue 9). Upon further evaluation, the staff determined that this action did not meet the definition of an interface requirement. This issue was reclassified as DFSER Open Item 3.5.2-1 related to ITAAC. Verification of adequate separation of redundant divisions of safety-related systems and location of these systems in structures designed to withstand externally generated missiles (including those that result from the failure of non-safety-related SSCs not housed in tornado-resistant buildings) ensure that these systems will not be rendered inoperable as a result of the missiles. In the DFSER, this verification was identified as Open Item 3.5.2-1. Subsequently, GE submitted design descriptions and ITAAC for the buildings within the ABWR design scope. These include verifications related to missile protection for safety-related SSCs and location of these systems in structures designed to withstand externally generated missiles.

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In the DFSER, the staff noted that the COL applicant should design SSCs outside the ABWR design scope. Any failure of these SSCs which may result in external missile generation should not prevent safety-related SSCs from performing their intended safety function. This was DFSER COL Action Item 3.5.2-1. GE has included this action item in the SSAR. This is acceptable.

Potential Impact Number: 22685 **SRP Section Number:** 6.5.1

Document: C&S: ASME N509-89
Title: Nuclear Power Plant Air Cleaning Units and Components

Summary:

SRP section 6.5.1 Acceptance Criteria specifies ANSI 509-1980. This code has been revised and is now designated as ASME N509-1989 "Nuclear Power Plant Air-Cleaning Units and Components."

Potential Impact Number: 22687 **SRP Section Number:** 6.5.1

Document: C&S: ASME N510-89
Title: Testing of Nuclear Air Treatment Systems

Summary:

SRP section 6.5.1 Acceptance Criteria specifies ANSI 510-1980. This code has been revised and is now designated as ASME N510-1989 "Nuclear Power Plant Air-Cleaning Units and Components."

Potential Impact Number: 22688 **SRP Section Number:** 6.5.1

Document: FINAL SER EPRI CH 9
Title: EPRI URD-EPD: SITE SUPPORT SYSTEMS

Summary:

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The EPRI FSER contains a summary of key requirements for Safety-Related HVAC Systems in section 8.2.1, two of these that are not properly addressed by SRP 6.5.1 are as follows:

All HVAC units and components and supports/hangers will be designed, constructed, and installed in accordance with ANSI/ASME Standard AG-1, "Code on Nuclear Air and Gas Treatment," (1989), and ANSI/ASME Standard N509, "Nuclear Power Plant Air Cleaning Units and Components," (1989).

HVAC systems will be designed to permit appropriate inservice inspections and functional testing in accordance with ANSI/ASME Standard N510, "Testing of Nuclear Air Cleaning Systems" (1989).

In the summary for this section of the FSER the staff concluded that the general design requirements in section 8.2.1 do not conflict with the guidance in several listed SRP sections and are acceptable.

Potential Impact Number: 22690 **SRP Section Number:** 9.5.7

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The diesel generator lubrication system conforms to position C.7 of Reg. Guide 1.9 as it relates to the protective interlocks for the diesel engine lubrication system. All trips associated with the lubrication system are bypassed during LOCA conditions. The diesel generator system protective interlocks are discussed in Section 8.3 of this report.

Potential Impact Number: 22691 **SRP Section Number:** 9.5.7

Document: C&S: IEEE 387
Title: IEEE STANDARD CRITERIA FOR DIESEL-GENERATOR UNITS APPLIED AS STANDBY POWER SUPPLIES FOR NUCLEAR POWER GENERATING STATIONS

Summary:

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The Standard describes the criteria for the application of diesel-generator units as individual units of the Class 1E standby power supplies in stationary nuclear power generating stations. The lubricating system is within the scope of the standard. Note: The current Standard is dated November 21, 1984.

Potential Impact Number: 22693 **SRP Section Number:** 9.5.8

Document: C&S: IEEE 387
Title: IEEE STANDARD CRITERIA FOR DIESEL-GENERATOR UNITS APPLIED AS STANDBY POWER SUPPLIES FOR NUCLEAR POWER GENERATING STATIONS

Summary:

The Standard describes the criteria for the application of diesel-generator units as individual units of the Class 1E standby power supplies in stationary nuclear power generating stations. The combustion air intake and exhaust system is within the scope of the standard. Note: The current Standard is dated November 21, 1984.

Potential Impact Number: 22694 **SRP Section Number:** 9.5.8

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The diesel generator combustion air intake and exhaust system meets the guidelines of RG 1.9, Revision 3 (July 1993) as it relates to system protective interlocks.

Potential Impact Number: 22696 **SRP Section Number:** 9.5.4

Document: C&S: IEEE 387
Title: IEEE STANDARD CRITERIA FOR DIESEL-GENERATOR UNITS APPLIED AS STANDBY POWER SUPPLIES FOR NUCLEAR POWER GENERATING STATIONS

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Summary:

The Standard describes the criteria for the application of diesel-generator units as individual units of the Class 1E standby power supplies in stationary nuclear power generating stations. The diesel engine fuel oil system is within the scope of the standard. Note: The current Standard is dated November 21, 1984.

Potential Impact Number: 22698 **SRP Section Number:** 9.5.4

Document: C&S: ANSI N195
Title: Fuel Oil Systems for Emergency Diesel Generators

Summary:

The standard provides the design requirements for the fuel oil system for diesel-generators that provide standby power for nuclear power generating stations. Note: the current version of the standard is ANSI ANS 59.51 dated October 27, 1989.

Potential Impact Number: 22714 **SRP Section Number:** 9.2.2

Document: FINAL SER EPRI CH 3
Title: EPRI URD-EPD: REACTOR COOLANT SYSTEM AND REACTOR NON-SAFETY AUXILIARY SYSTEMS

Summary:

Section 3.4.2.2.1 of Chapter 3 of the original Evolutionary Requirements Document required that CCW to the reactor coolant pumps and motors not be isolated on an automatic containment isolation signal. Plant operators will manually isolate the cooling water flow if it becomes a release path from the containment during a LOCA. By not requiring automatic isolation of pump cooling water on a containment isolation signal, 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. The staff concluded that there was merit to not requiring that the CCW to the reactor coolant pumps and motors be automatically isolated on the receipt of a containment isolation signal. However, the staff recommended that EPRI extend the requirement to 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.

Section 3.4.2.3.1 of Chapter 3 has been revised to state that CCW to the reactor coolant pumps and motors will not be isolated on an automatic containment isolation signal. However, the main control room operators will be provided with the necessary information (e.g., from radiation monitoring instrumentation in the CCW return lines) and bases to determine when it is necessary to isolate the affected line from the main control room. EPRI states that main control room operators will manually isolate cooling water flow if it becomes a release path from the containment during a LOCA. The staff concludes that the provisions included by EPRI to require the control room operators to have the necessary information and bases to determine when it is necessary to isolate the affected line are acceptable.

Potential Impact Number: 22715 **SRP Section Number:** 6.4

Document: DRAFT FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

GE identified the interface requirement described below for the applicants referencing the ABWR design. However, as a result of further review of all interface items, GE has committed to revise the SSAR to reclassify this item as a COL ITAAC item.

In accordance with TMI Action Plant Item III.D.3.4, utility applicants referencing the ABWR design will be required to demonstrate that control room operators are adequately protected against the effects of the release of toxic substances, either on or off the site, and that the plant can be safely operated or shut down under conditions created by design-basis accidents. The amounts and locations of any possible sources of toxic substances in each plant vicinity will have to be identified following the methods outlined in RGs 1.78 (Rev. 0) and 1.95 (Rev. 1). Specific detectors to permit automatic control room isolation will have to be provided, where necessary. The need for site-specific toxic gas protection will be reviewed to ensure that the control room operators are protected against releases of hazardous material.

Potential Impact Number: 22716 **SRP Section Number:** 6.4

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Document: DRAFT SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In accordance with TMI Action Plan Item III.D.3.4, utility applicants referencing the System 80+ design must demonstrate that control room operators are adequately protected against the effects of the release of toxic substances, either on or off the site, and that the plant can be safely operated or shut down under conditions created by any design-basis accident. The COL applicant must determine the amounts and locations of any possible sources of toxic substances near the plant using the methods in RGs 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release." The COL applicant must include specific detectors where necessary to permit automatic isolation of the control room. The COL applicant will review the need for site-specific protection against toxic gases to ensure that control room operators are protected against releases of hazardous material (COL Action Item 6.2-4).

Potential Impact Number: 22719 **SRP Section Number:** 9.4.1

Document: DRAFT FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

DFSER Section 6.4: GE identified the interface requirement described below for the applicants referencing the ABWR design. However, as a result of further review of all interface items, GE has committed to revise the SSAR to reclassify this item as a COL ITAAC item.

In accordance with TMI Action Plant Item III.D.3.4, utility applicants referencing the ABWR design will be required to demonstrate that control room operators are adequately protected against the effects of the release of toxic substances, either on or off the site, and that the plant can be safely operated or shut down under conditions created by design-basis accidents. The amounts and locations of any possible sources of toxic substances in each plant vicinity will have to be identified following the methods outlined in RGs 1.78 (Rev. 0) and 1.95 (Rev. 1). Specific detectors to permit automatic control room isolation will have to be provided, where necessary. The need for

site-specific toxic gas protection will be reviewed to ensure that the control room operators are protected against releases of hazardous material.

DFSER Section 9.4.1: The evaluation of requirements for protection against hazardous chemical releases will be site specific. COL applicants referencing the ABWR design will need to provide further information regarding compliance with RG 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release." In its review, the COL applicants will be required to identify plant design features that will automatically isolate the control room from the outside when necessary, and thus protect the control room operators from toxic substances in the plant vicinity. This is a COL Action Item 9.4.1.1-1.

Potential Impact Number: 22720 **SRP Section Number:** 9.4.1

Document: DRAFT SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

Section 6.4 of the DSER: In accordance with TMI Action Plan Item III.D.3.4, utility applicants referencing the System 80+ design must demonstrate that control room operators are adequately protected against the effects of the release of toxic substances, either on or off the site, and that the plant can be safely operated or shut down under conditions created by any design-basis accident. The COL applicant must determine the amounts and locations of any possible sources of toxic substances near the plant using the methods in RGs 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release." The COL applicant must include specific detectors where necessary to permit automatic isolation of the control room. The COL applicant will review the need for site-specific protection against toxic gases to ensure that control room operators are protected against releases of hazardous material (COL Action Item 6.2-4).

Section 9.4.1 of the DSER: Regarding specifics relating to compliance with GDC 19 as it pertains to protection of the control room against intrusion of toxic gasses into the control room, refer to Section 6.4 of this DSER. As stated in that section, the

compliance with GDC 19 in this regard is demonstrated by meeting the guidance of RGs 1.78 and 1.95 which, in turn, is within the scope of the referencing applicant since it is site dependent.

Potential Impact Number: 22721 **SRP Section Number:** 9.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue addressed concerns with the potential for core-melt due to component cooling water (CCW) system failures. The principle concern identified in NUREG 0933 is the potential for a common CCW failure resulting in a SBLOCA due to RCP seal failure concurrent with the unavailability of most of the ECCS.

This issue was given a high priority, however because of relationship between CCW failure and RCP seal failure, this issue was latter integrated into GI-23.

Potential Impact Number: 22722 **SRP Section Number:** 9.2.2

Document: FINAL SER EPRI CH 3
Title: EPRI URD-EPD: REACTOR COOLANT SYSTEM AND REACTOR NON-SAFETY AUXILIARY SYSTEMS

Summary:

Sections 3.4.2.2, 3.4.2.6, 3.4.2.11, and 6.3.1 of Chapter 3 of the original Evolutionary Requirements Document included utility requirements that resolve GSI-23, "Reactor Coolant Pump Seal Failures" for the ALWR, assuming they could be met in specific plant designs. However, it appeared that some of these requirements may not be practical without the development of adequate pump seals and/or the provisions of independent seal cooling. In the DSER for Chapter 3, the staff stated that, unless otherwise determined at the time this issue is resolved, new plant designs should provide independent seal cooling during station blackout.

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Potential Impact Number: 22723 **SRP Section Number:** 9.2.2

Document: FINAL SER EPRI CH 8
Title: EPRI URD-EPD: PLANT COOLING WATER SYSTEMS

Summary:

The staff concluded that, specifically for a PWR, EPRI should evaluate the possibility of providing an RCP seal cooling system that is independent of the service water system to address the issue of RCP seal failure GI-23.

In response to this issue, EPRI stated that Chapter 3 had been revised to ensure seal leakage will be minimal following loss of seal cooling. Specifically, Section 3.4.2.3.4 of Chapter 3 now requires that the degradation of the shaft seal system be negligible following loss of both seal injection and pump cooling water for up to 1 hour. In addition, Section 3.4.2.3.5 of Chapter 3 now specifies that in the event of the loss of all ac power, reactor coolant system (RCS) leakage through the RCP shaft seals will be limited so that the reactor core will remain covered and natural circulation cooling of the core will be maintained for at least 8 hours. EPRI stated that an additional cooling water system was not specified because it would complicate the design and could possibly result in an additional service water system. The staff concludes that EPRI has provided information to address the intent of GI 23.

Potential Impact Number: 22727 **SRP Section Number:** 14.2

Document: FINAL SER ABWR CH 14
Title: INITIAL TEST PROGRAM

Summary:

The SSAR lists the NRC regulatory guides that will be used in the development of the initial test program and states that the applicable tests will comply with these guides. The applicable revisions to these regulatory guides are listed in an SSAR Table. GE corrected the reference to RG 1.68.3 contained in the SSAR, to Revision 0, issue date of April 1982. This is acceptable.

Potential Impact Number: 22734 **SRP Section Number:** 14.2

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Document: FINAL SER CE80 CH 14
Title: VERIFICATION PROGRAMS

Summary:

The Initial Test Program (ITP) is consistent with the recommendations of the following regulatory guides associated with startup (with exceptions as noted and revisions as specified in CESSAR-DC Table 1.8-1): 1.9, 1.18, 1.20, 1.30, 1.41, 1.52, 1.68, 1.68.2, 1.68.3, 1.79, 1.108, 1.116, 1.118, and 1.140.

Potential Impact Number: 22737 **SRP Section Number:** 14.2

Document: FINAL SER CE80 CH 14
Title: VERIFICATION PROGRAMS

Summary:

Regulatory Guide 1.68.2, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water Cooled Nuclear Power Plants"

In the DSER, the staff asked ABB-CE to revise CESSAR-DC Section 14.2.12.4.7, "Shutdown From Outside the Control Room Test," to address the following items in accordance with RG 1.68.2:

(1) Specify that plant systems are in the normal configuration with the turbine generator in operation.

(2) Specify that the test is accomplished with the minimum shift crew complement.

ABB-CE revised CESSAR-DC Section 14.2.12.4.7 to specify that plant systems are in the normal configuration with the turbine generator in operation and that the test is accomplished with the minimum shift crew compliment in accordance with RG 1.68.2. The staff reviewed the changes that were incorporated into CESSAR-DC in Amendment J; they are acceptable.

Potential Impact Number: 22738 **SRP Section Number:** 14.2

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Document: FINAL SER CE80 CH 14
Title: VERIFICATION PROGRAMS

Summary:

Regulatory Guide 1.68.3, "Preoperational Testing of Instrument and Control Air Systems"

In the DSER, the staff asked ABB-CE to revise CESSAR-DC Section 14.2.12.1.88, "Compressed Air System Test," to address the following items in accordance with RG 1.68.3:

(1) Determine that the total air demand at normal steady-state conditions, including leakage from the system, is in accordance with design requirements (Position C.5).

(2) Demonstrate that the plant equipment designated to be supplied by the instrument air system is not degraded when supplied by the station air system which may have less-restrictive air quality requirements (Position C.9).

For Item 1 above, ABB-CE revised CESSAR-DC Section 14.2.12.1.88, "Compressed "Air System Test," to reflect the guidance of RG 1.68.3, Regulatory Position C.5. The staff reviewed the changes to the compressed air system test in CESSAR-DC to verify test abstract conformance to RG 1.68.3, Regulatory Position C.5 (Amendment Q); they are acceptable.

For Item 2 above, ABB-CE revised CESSAR-DC Section 14.2.7 to state that "the CE System 80+ Instrument Air System has no interconnection to any other air compressed system; therefore, ingress of less quality air is not possible. Consequently, Position C.9 does not apply." The applicant submitted this information in CESSAR-DC (Amendment L). The staff accepted the applicant's resolution of the associated open item.

Potential Impact Number: 22739 **SRP Section Number:** 14.2

Document: FINAL SER CE80 CH 14
Title: VERIFICATION PROGRAMS

Summary:

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As is evident from the subsection headings and associated content in Section 14.2 of the CE System 80+ FSER, in addition to the eight topical Areas of Review identified in SRP Section 14.2, the staff reviewed four topics related to the applicant's initial test program description which are not explicitly covered in SRP Section 14.2. The four additional topics appear to be Organization and Staffing; Conduct of the Test Program; Review, Evaluation, and Approval of Test Results; and Test Records.

Potential Impact Number: 22740 **SRP Section Number:** 14.2

Document: FINAL SER CE80 CH 14
Title: VERIFICATION PROGRAMS

Summary:

The startup test program described in CESSAR-DC Section 14.2 includes security lighting system and security radio system tests. In CESSAR-DC Section 9.5.2, ABB-CE describes a security radio system for offsite communications. Security "lock-down" of the protected area and startup testing of the rest of the security system (i.e., intrusion detection system, alarm assessment system, access control system, etc.) were not addressed.

In response to the applicant's response to defer portions for COL action, the staff finds it acceptable that detailed description and NRC review of the rest of the security system and its test and acceptance criteria be deferred to the COL applicant's security plan submittal required by 10 CFR 50.34. This is COL Action Item 14.2.13-1. By letter dated January 26, 1993, ABB-CE noted in CESSAR-DC Section 14.2.7.5 that the COL applicant will provide site-specific security, contingency, and guard training plans. This is acceptable.

In RAI Q500.31, the staff noted that the security lighting system test described in CESSAR-DC Section 14.2.12.1.85 was incomplete in that it does not address testing on loss of normal power nor testing of its adequacy for support of closed-circuit television (CCTV) security functions. The applicant's initial proposed change addressed demonstrating lighting adequacy on loss of power "for each affected room" but not where "required to monitor isolation zones and all outdoor areas within the plant protected perimeter." ABB-CE eventually modified CESSAR-DC Section 14.2.12.1.85 to require demonstration of the adequacy of illumination for CCTV security functions on loss of normal lighting power for monitored isolation zones and outdoor areas within the

plant protected perimeter. ABB-CE incorporated this test abstract information into CESSAR-DC Amendment O. On this basis, DSER Open Item 14.2.13-1 was resolved.

Also in RAI Q500.31, the staff noted that the communications test described in CESSAR-DC Section 14.2.12.1.87 did not address verifying communication capability and noninterference with equipment from normal security patrol, alarm response, and interdiction paths within the plant. The applicant committed to amend subsections of the communications system test. This was designated as DSER Confirmatory Item 14.2.13-1. By letter dated April 4, 1993, the applicant added the following statement to CESSAR-DC Section 14.2.12.1.87:

3.4 Verify the Security Radio System functions properly at all locations throughout the plant.

This is acceptable. On this basis, DSER Confirmatory Item 14.2.13-1 is resolved.

Potential Impact Number: 22743 **SRP Section Number:** 14.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

EPRI requires a plant security plan covering the construction phase. Although this is a utility requirement outside the scope of 10 CFR Part 73, the requirement is compatible with NRC requirements.

EPRI requires the utility to establish security boundaries as part of the startup testing program. EPRI agreed that the detailed construction and startup schedule will have to address NRC review and approval of the installed security system for the operating phase before the first fuel loading, but that this milestone was beyond the scope of the Evolutionary Requirements Document. The staff, therefore, will address this matter during its review of an individual application for a combined license.

The staff expects that at least 60 days before loading fuel, a combined license licensee will have confirmed 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 their entirety as

they would when the reactor is operating and will continue to do so. The licensee's determination that operational status has been achieved must be based on tests conducted under realistic operating conditions of sufficient duration that demonstrate that the equipment is properly operating and capable of long-term, reliable operation; that procedures have been developed, approved, and implemented; and that personnel responsible for security operations and maintenance have been appropriately trained and have demonstrated their capability to perform their assigned duties and responsibilities.

Potential Impact Number: 22747 **SRP Section Number:** 14.2

Document: NUREG 0694
Title: TMI - RELATED REQUIREMENTS FOR NEW OPERATING LICENSES

Summary:

Part 1 - Fuel Loading and Low-Power Testing Requirements....

p. 14- I.C.7 NSSS Vendor Review of Procedures

Obtain nuclear steam supply system (NSSS) vendor review of low-power testing procedures to further verify their adequacy. This requirement must be met before fuel loading.

p.15-I.G.1 Training During Low-Power Testing

Define and commit to a special low power testing program approved by NRC 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 and to provide supplemental training. This requirement shall be met before fuel loading.

Part 2 - Full-Power Requirements....

p. 21-I.C.7 NSSS Vendor Review of Procedures

Obtain NSSS vendor review of power-ascension test and emergency procedures to further verify their adequacy. This requirement must be met before issuance of a full-power license.

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p. 21-I.G.1 Training During Low-Power Testing

Supplement operator training by completing the special low power test program. Tests may be observed by other shifts or repeated on other shifts to provide training to the operators. This requirement shall be met before issuance of a full-power license.

Potential Impact Number: 22758 **SRP Section Number:** 14.2

Document: RG 1.9 **Revision:** 3 , 7/1/93
Title: SELECTION, DESIGN, AND QUALIFICATION OF DIESEL-GENERATOR UNITS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

Summary:

This regulatory guide 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.

This revision of Regulatory Guide 1.9 integrates into a single regulatory guide pertinent guidance previously addressed in Revision 2 of Regulatory Guide 1.9, Revision 1 of Regulatory Guide 1.108, and Generic Letter 84-15, and it endorses, as appropriate, guidelines set forth in IEEE Std 387-1984.

Conformance with the guidelines in IEEE Std 387-1984, 'IEEE Standard Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations, is acceptable to the NRC staff for satisfying the Commission's regulations with respect to design, qualification, and periodic testing of diesel generator units used as onsite electric power systems for nuclear power plants subject to the exceptions identified in the regulatory positions.

Regulatory positions C.2 through C.4 provide a considerable amount of new guidance related to preoperational testing, reliability assurance, recordkeeping, and reporting of failures for diesel generators.

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Potential Impact Number: 22769 **SRP Section Number:** 11.3

Document: FINAL SER EPRI CH 12
Title: EPRI URD-EPD: RADIOACTIVE WASTE PROCESSING SYSTEMS

Summary:

The staff stated that it is appropriate to design ALWR radwaste systems so that they have the capability to process liquid and gaseous radioactive wastes corresponding to 1-percent failed fuel for an evolutionary PWR and 100 Ci/sec-MWt noble gas release rate at 30 minutes' decay for the off-gas system for an evolutionary BWR in accordance with the criteria in SRP Sections 11.2 (Item III.2.C) and 11.3 (Item III.2.b). Further, the above basis should be used to (1) demonstrate compliance of concentrations of radionuclides in gaseous effluents and liquid effluents discharged to unrestricted areas with applicable 10 CFR Part 20 regulatory limits and (2) analyze the failure of gaseous radioactive waste processing systems to show that the system design meets the guidelines of Branch Technical Position ETSB 11-5, 'Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure' (SRP Section 11.3).

Potential Impact Number: 22770 **SRP Section Number:** 11.3

Document: DRAFT FINAL SER ABWR CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

The staff also reviewed the capability of the offgas system to limit the wholebody dose to less than 10 percent of the 10 CFR Part 100 limits for an individual exposure of 2 hours at the nearest exclusion area boundary due to radioactive releases from a postulated offgas system leak or failure as assumed in BTP ETSB 11-5, Revision 0, July 1981.

Potential Impact Number: 22771 **SRP Section Number:** 11.3

Document: DRAFT SER CE80 CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

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Summary:

The staff evaluated the waste gas processing system's compliance with Branch Technical Position Effluent Treatment Systems Branch (BTP ETSB) 11-5, 'Postulated Radioactive Releases Due to Waste Gas System Leak or Failure.'

Potential Impact Number: 22772 **SRP Section Number:** 6.6

Document: NRC BULLETIN 87-01
Title: THINNING OF PIPE WALLS IN NUCLEAR POWER PLANTS

Summary:

All licensees have either explicitly or implicitly committed to maintain the functional capability of high-energy piping systems that are a part of the licensing basis for the facility. An important part of this commitment is that piping will be maintained within allowable thickness values. The purpose of this bulletin is to request that licensees submit information concerning their programs for monitoring the thickness of pipe walls in high-energy single-phase and two-phase carbon steel piping systems.

Potential Impact Number: 22773 **SRP Section Number:** 14.2

Document: FINAL SER ABWR CH 14
Title: INITIAL TEST PROGRAM

Summary:

Appendix A to SSAR states that testing described in Chapter 14 is consistent with the BWR Owner's Group response to Action Item I.G.1 of NUREG-0737 as documented in a letter dated February 4, 1981, from D.B. Waters to D.G. Eisenhut. The staff determined that the test abstracts in SSAR Section 14.2.12 that describe testing outlined in Appendix E of this letter should be identified or modified accordingly. This was DSER Open Item 124. GE stated in its response of May 20, 1991, that testing outlined in Appendix E of the referenced document would be specified in the applicable test abstracts.

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GE revised test abstracts 14.2.12.1.1(3)(a), 14.2.12.1.9(3)(j), and 14.2.12.1.44(3)(a) to include a reference to 1A.2.4 of Appendix A to SSAR Chapter 1, and GE revised 1A.2.4 to discuss the requirements of Action Item I.G.1 Appendix E applicable to the initial test program. The staff found this item acceptable subject to incorporation into a future SSAR revision. This resolved DSER Open Item 124, and was tracked in the DFSER as DFSER Confirmatory Item 14.2.12.5-1.

GE revised the above test abstracts in SSAR Amendment 23. The staff finds the changes acceptable. This resolved DFSER Confirmatory Item 14.2.12.5-1.

Potential Impact Number: 22774 **SRP Section Number:** 6.6

Document: NRC GENERIC LETTER 89-08
Title: EROSION/CORROSION-INDUCED PIPE WALL THINNING

Summary:

Generic Letter 89-08 requested a response from licensees that was to include information on whether or not they have implemented or intend to implement a long term erosion/corrosion monitoring program that provides assurances that procedures or administrative controls are in place to assure that the NUMARC program or another equally effective program is implemented and the structural integrity of all high- energy (two phase as well as single phase) carbon steel systems is maintained.

Potential Impact Number: 22775 **SRP Section Number:** 14.2

Document: FINAL SER ABWR CH 14
Title: INITIAL TEST PROGRAM

Summary:

In the DSER, the staff determined that Section 14.2.12.2.14, "Feedwater Control," should address the following items in accordance with RG 1.68, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants," Revision 1:

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(1) Modify the test description for demonstration of the required functionality of the feedwater system at low reactor power (less than or equal to 15-percent reactor power) (RG 1.68. Position 1.C.2.a). This was DSER Open Item 126.

GE stated in its response of May 20, 1991, that this testing is already specified in the current description, and supplemented by information to be included in a test matrix.

The staff verified that GE's proposed test matrix, to be included in the SSAR, identified feedwater system performance and feedwater control system adjustment/confirmation tests to be performed at the nuclear heat-up and low-power testing plateaus. In the DFSER, the staff found this acceptable subject to incorporation into a future SSAR revision. This resolved DSER Open Item 126, and this issue became DFSER Confirmatory Item 14.2.12.6-2.

GE added the test matrix in SSAR Amendment 23. The test matrix contained feedwater system performance and feedwater control system adjustment/confirmation tests which are performed at the nuclear heat-up and low-power testing plateaus. The staff finds this to be acceptable. This resolved DFSER Confirmatory Item 14.2.12.6-2.

(2) Modify or clarify the test acceptance criteria to provide assurance that vibration levels for system components and piping are within predetermined limits (RG 1.68, Position 1C.2.f); piping movement during heat up and steady state and transient operation are within predetermined limits (RG 1.68, Position 1.C.2.g); and adequate margins exist between system variables and set points of instruments monitoring these variables to prevent spurious actuation or loss of system pumps and motor-operated valves (RG 1.68, Position 1.C.2.h). This was DSER Open Item 127.

GE stated in its response of May 20, 1991, that the testing called for by Positions C.2.f and C.2.g is included in the test abstracts of SSAR Sections 14.2.12.1.51, 14.2.12.1.53(3)(b) and (k), 14.2.12.2.10, 14.2.12.2.11, and 14.2.12.2.18. GE further stated that it revised Section 14.2.12.2.18 in SSAR Amendment 18 to more specifically address Position C.2.h. In the DFSER, the staff verified the above, and found it acceptable. This resolved DSER Open Item 127.

Analyst Note-The above discussion relates to conformance with Reg. Guide 1.68.1.

Potential Impact Number: 22776 **SRP Section Number:** 14.2

Document: FINAL SER ABWR CH 14

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Title: INITIAL TEST PROGRAM

Summary:

In the DSER, the staff determined that, for SSAR Section 14.2.12.1.27, GE should address the following items in accordance with RG 1.68:

(1) Determine that the total air demand at normal steady-state conditions, including leakage from the system, is in accordance with design (RG 1.68, Position 3.C.5). This was DSER Open Item 127.

GE revised Section 14.2.12.1.27(3)(f) in SSAR Amendment 18 to include testing of the total air demand at normal steady-state conditions, including leakage from the system, in accordance with the design as specified in Position 3.C.5. The staff finds this acceptable. This resolved DSER Open Item 127.

(2) Demonstrate that the plant equipment designated by design to be supplied by the instrument and control air system is not being supplied by other compressed air supplies (such as service air (SA)) that may have less restrictive air quality requirements (RG 1.68, Position 3.C.9). This was DSER Open Item 128.

For Position 3.C.9, GE stated in its response of May 20, 1991, that the SA system acts as a backup to instrument air upstream of the instrument air filters. Furthermore, although totally separate (except for the manual backup cross-tie), the design of the two systems is essentially identical. Thus, the air supplied to the inlet of the instrument air filters is of the same quality, whether it is sourced from the instrument or SA system; therefore, the outlet air will be of the same quality. Because the design precludes occurrence of the conditions hypothesized, no specific test demonstration is needed beyond the construction verification and preoperational testing already planned. The staff finds this acceptable. This resolved DSER Open Item 128.

(3) Demonstrate that functional testing of instrument and control air systems important to safety is performed to ensure that credible failures resulting in an increase in the supply system pressure will not cause loss of operability (RG 1.68, Position 3.C.11). This was DSER Open Item 129.

For Position 3.C.11, GE revised Section 14.2.12.1.27 in SSAR Amendment 18 to include testing that will demonstrate continued operability of supplied loads in response to credible failures that result in an increase in the supply system pressure. The staff finds this item acceptable. This resolved DSER Open Item 129.

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Analyst Note-The above discussion from the ABWR FSER relates to conformance with Reg. Guide 1.68.3. Reg. Guide 1.68.3 is not explicitly identified as guidance in SRP Section 14.2, subsection II. Reg. Guide 1.68.3 supersedes Reg. Guide 1.80 which is cited in SRP Section 14.2, subsection II.

Potential Impact Number: 22778 **SRP Section Number:** 6.6

Document: NUREG 0313 **Revision:** 2 , 1/1/88
Title: TECHNICAL REPORT ON MATERIAL SELECTION AND PROCESSING GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING. FINAL REPORT

Summary:

Substantial efforts in research and development have been sponsored by the BWR Owners Group for IGSCC Research. The results of this program, along with other related work by vendors, consulting firms, and confirmatory research sponsored by the NRC, have permitted the development of revised Staff Positions regarding the IGSCC problems. The technical bases for these positions are detailed in NUREG-0313, Rev. 2 "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." NUREG-0313, Rev. 2 was promulgated by Generic Letter 88-01.

Potential Impact Number: 22798 **SRP Section Number:** 9.5.4

Document: RG 1.9 **Revision:** 3 , 7/1/93
Title: SELECTION, DESIGN, AND QUALIFICATION OF DIESEL-GENERATOR UNITS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

Summary:

Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel-Generator Units used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 3 was issued in July of 1993. Revisions to this Regulatory Guide should be reviewed for impact on the SRP.

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Potential Impact Number: 22815 **SRP Section Number:** 3.5.1.1

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

"The COL applicant shall provide design details of all pressurized gas bottles as well as details of missile protection features for structures, systems, and components that are outside of the ABWR scope. The staff will review these designs and features on a plant-specific basis."

SRP Section 3.5.1.1 does not currently specifically describe a review of design details associated with pressurized gas bottles.

Potential Impact Number: 22816 **SRP Section Number:** 3.5.1.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The COL applicant shall provide procedures ensuring that equipment undergoing maintenance will be removed from containment prior to operation, moved to a location where it is not a potential hazard to safety-related equipment or will be seismically restrained to protect it from becoming a missile.

SRP Section 3.5.1.2 does not currently describe a review of the applicants procedures for addressing this issue.

Potential Impact Number: 22817 **SRP Section Number:** 2.4.10

Document: C&S: ANSI N170
Title: Standards for Determining Design Basis Flooding at Power Reactor Sites

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Summary:

Standard ANSI N170, relating to determination of design basis flooding at reactor sites, is cited, without specific identification of version/date, as Reference 8 in SRP Section 2.4.10, subsection VI. Reg. Guide 1.59, which is cited as guidance for SRP Section 2.4.10 reviews in subsections II and III, endorses ANSI N170-1976 (See Appendix A of the Reg. Guide) with exceptions.

This standard was retitled and redesignated in 1981 as ANSI/ANS 2.8-1981, "Determining Design Basis Flooding at Power Reactor Sites." The current version of this standard is ANSI/ANS 2.8-1992.

Potential Impact Number: 22818 **SRP Section Number:** 2.4.10

Document: C&S: ANS 2.8
Title: DETERMINING DESIGN BASIS FLOODING AT POWER REACTOR SITES

Summary:

Standard ANSI N170, relating to determination of design basis flooding at reactor sites, is cited, without specific identification of version/date, as Reference 8 in SRP Section 2.4.10, subsection VI. Reg. Guide 1.59, which is cited as guidance for SRP Section 2.4.10 reviews in subsections II and III, endorses ANSI N170-1976 (See Appendix A of the Reg. Guide) with exceptions.

ANSI N170 was retitled and redesignated in 1981 as ANSI/ANS 2.8-1981, "Determining Design Basis Flooding at Power Reactor Sites." The current version of this standard is ANSI/ANS 2.8-1992.

Potential Impact Number: 22820 **SRP Section Number:** 3.9.1

Document: DRAFT SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

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The applicant should submit additional information for the staff to review regarding the applicability of the St. Lucie and System 80 analyses to System 80+. The applicant should also verify that the number of natural convection cooldown events assumed in the analyses are applicable to System 80+ which has a 60-year plant design life.

Potential Impact Number: 22822 **SRP Section Number:** 3.9.5

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

For the design of critical reactor vessel internals, in the early amendments of CESSAR-DC Section 3.9.5.4, ABB-CE stated that for components subject to fatigue, the stress analysis will be performed using the design fatigue curve of I-9-2 of Section III of the ASME Code. The staff raised concerns relating to possible detrimental environmental effects not presently reflected in current ASME Code design fatigue curves. Subsequently, in CESSAR-DC Section 3.9.1.1 (Amendment R), ABB-CE noted that observations of significant environmental degradation in LWR environments are primarily related to high strain ranges, slow strain rates, high oxygen contents of primary water, high sulfur contents of carbon and low alloy steels, and low-flow-rate conditions. The CESSAR-DC further stated that, since System 80+ components are not exposed to high oxygen content environments at elevated temperatures, and no carbon or low alloy steel is directly exposed to the primary coolant, significant environmental degradation would not likely occur to the System 80+ components which include reactor vessel internals. This clarification is acceptable and resolves DSER Open Item 3.9.5-4.

Potential Impact Number: 22824 **SRP Section Number:** 3.9.5

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

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Issue A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant Systems," was resolved with the publication of NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems." To resolve A-2, the asymmetric loads on the reactor vessel, internals, primary coolant loop, and components should not exceed the limits imposed by the applicable codes and standards. The staff also issued Generic Letter (GL) 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," on February 1, 1984, to permit the application of leak-before-break (LBB) technology to eliminate the postulated pipe rupture from the design basis. Subsequently, the staff revised General Design Criterion (GDC) 4 to permit the application of LBB.

In CESSAR-DC Section 3.6.3, ABB-CE proposes to use LBB methodology to eliminate the postulated pipe rupture from the design basis. The staff evaluated the LBB methodology in Section 3.6.3 of this report and approved its application to the System 80+ design. Where the LBB approach cannot be applied, ABB-CE states that the pipe break locations and resulting dynamic effects are determined. Each postulated pipe rupture is considered separately as a single postulated initiating event.

In the DSER, the staff stated if ABB-CE could not obtain staff approval for the LBB approach, it would have to submit details of its analysis on assessing the effects of the asymmetric blowdown loads. This was designated as DSER Open Item 20.1-2. On the basis that the staff has approved the application of LBB for the System 80+ design, this DSER Open Item 20.1-2 is resolved.

On the basis of the above, ABB-CE's proposal for resolving Issue A-2 is adequate to address asymmetric blowdown loads and, thus, is acceptable in resolving Issue A-2 for the System 80+ design.

Potential Impact Number: 22825 **SRP Section Number:** 3.9.5

Document: NRC GENERIC LETTER 84-04
Title: SAFETY EVALUATION OF WESTINGHOUSE TOPICAL REPORTS DEALING WITH ELIMINATION OF POSTULATED PIPE BREAKS IN PWR PRIMARY MAIN LOOPS

Summary:

After reviewing reports submitted to address asymmetric blowdown loads on the PWR primary systems that result from a limited number of discrete break locations as

stipulated in NUREG-0609 (the staff's resolution of Unresolved Safety Issue A-2) the staff concluded an acceptable technical basis has been provided so that the asymmetric blowdown loads resulting from double ended pipe breaks in main coolant loop piping need not be considered as a design basis for the Westinghouse Owner's Group plants.

Potential Impact Number: 22826 **SRP Section Number:** 3.9.5

Document: NUREG 0609
Title: ASYMMETRIC BLOWDOWN LOADS ON PWR PRIMARY SYSTEMS - RESOLUTION OF GENERIC TASK ACTION PLAN A-2

Summary:

The NRC staff, after being informed of newly identified asymmetric loadings resulting from postulated ruptures of primary piping, initiated a generic investigation, Task Action Plan A-2. The intent of the investigation was to develop acceptable criteria and guidelines for evaluating plant analyses. The staff concludes that an acceptable basis is provided in this report for performing and reviewing plant analyses. Criteria were developed for evaluating loading transients, structural components and the fuel assembly. The staff approved computer programs and modeling techniques submitted by each PWR vendor for development of the subcooled blowdown and cavity-pressure loading transients.

Potential Impact Number: 22831 **SRP Section Number:** 3.9.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue addresses the potential relating to an unanalyzed reactor vessel thermal stress that could occur during natural convection cooldown of PWR reactors. The concern emerged from a preliminary B&W evaluation of the voiding event that had occurred in the upper head of the St. Lucie Unit 1 reactor on June 11, 1980. On the basis of several conservative assumptions, Babcock & Wilcox tentatively concluded that during natural convection cooling, axial temperature gradients could develop in the

vessel flange area which could produce thermal stresses in the flange area or in the studs that might exceed values allowed by the code when added to the stresses already considered (such as boltup loads or pressure loads). This issue was resolved and no new requirements were established because (1) natural convection cooldown events that result in the plant being brought to a cold shutdown condition occur infrequently and (2) the actual severity of a specific natural convection cooldown event will determine the need for actions (if any) and the extent of actions that may be required of any licensee following certain natural convection cooldown events that may place a reactor vessel in an unanalyzed condition or outside its documented design basis.

Potential Impact Number: 22832 **SRP Section Number:** 3.9.1

Document: NRC GENERIC LETTER 92-02 Revision Date: 3/6/92
Title: RESOLUTION OF GENERIC ISSUE 79, UNANALYZED REACTOR VESSEL (PWR) THERMAL STRESS DURING NATURAL CONVECTION COOLDOWN

Summary:

This letter informs licensees of the resolution of Generic Issue 79, "Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown" and the conclusions reached by the staff as the result of the evaluations performed to resolve this generic issue. No new requirements are being established and no specific action or written response is required.

Potential Impact Number: 22833 **SRP Section Number:** 2.4.11

Document: NUREG 0733 Revision Date: 8/1/81
Title: ANALYSIS OF ULTIMATE HEAT-SINK SPRAY PONDS

Summary:

This report 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 required by U.S. Nuclear Regulatory Commission Regulatory Guide 1.27. The models of spray-pond performance are based on heat and mass transfer characteristics of drops in an

environment whose humidity and velocity have been modified by the presence of the sprays. Drift loss from the sprays is estimated by a ballistics model. The pond performance model is used first to scan a long-term weather record from a representative meteorological station in order to determine the periods of most adverse meteorology for cooling or evaporation. The identified periods are used in subsequent calculations to actually estimate the design-basis pond temperature. Additionally, methods are presented to correlate limited quantities of onsite data to the longer offsite record, and to estimate the recurrence interval of the design-basis meteorology chosen.

This document is cited , without identification of its number, in SRP Section 2.4.11, subsection VI, Reference 25.

Microfiche No 09881:001

Potential Impact Number: 22834 **SRP Section Number:** 2.4.2

Document: C&S: ANS 2.8
Title: DETERMINING DESIGN BASIS FLOODING AT POWER REACTOR SITES

Summary:

ANSI N170, which is referenced as a guidance document, was revised to ANSI/ANS-2.8 in 1981 and further revised in 1992. The standard provides methods for estimating probable maximum floods at power reactor sites.

Potential Impact Number: 22835 **SRP Section Number:** 2.4.2

Document: C&S: ANSI N170
Title: Standards for Determining Design Basis Flooding at Power Reactor Sites

Summary:

ANSI N170, which is referenced as a guidance document, was revised to ANSI/ANS-2.8 in 1981 and further revised in 1992. The standard provides methods for estimating probable maximum floods at power reactor sites.

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Potential Impact Number: 22836 **SRP Section Number:** 2.4.3

Document: C&S: ANS 2.8
Title: DETERMINING DESIGN BASIS FLOODING AT POWER REACTOR SITES

Summary:

ANSI N170, which is referenced as a guidance document, was revised to ANSI/ANS-2.8 in 1981 and further revised in 1992. The standard provides methods for estimating probable maximum floods at power reactor sites.

Potential Impact Number: 22837 **SRP Section Number:** 2.4.3

Document: C&S: ANSI N170
Title: Standards for Determining Design Basis Flooding at Power Reactor Sites

Summary:

ANSI N170, which is referenced as a guidance document, was revised to ANSI/ANS-2.8 in 1981 and further revised in 1992. The standard provides methods for estimating probable maximum floods at power reactor sites.

Potential Impact Number: 22841 **SRP Section Number:** 9.1.1

Document: C&S: ANS 57.1
Title: Design Requirements for Light Water Reactor Fuel Handling Systems

Summary:

This standard describes the design requirements for nuclear fuel handling systems. The fuel handling system is a integral part of the fuel storage facility.

Potential Impact Number: 22842 **SRP Section Number:** 9.1.1

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Document: C&S: ANS 57.3
Title: Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants

Summary:

This standard defines the required functions of wet and dry storage facilities for new fuel at LWR power plants. It provides minimum design requirements for safe storage of new nuclear fuel and control components.

Potential Impact Number: 22844 **SRP Section Number:** 2.3.5

Document: NUREG CR-2919
Title: USER GUIDE FOR XOGDOQ: EVALUATING ROUTINE EFFLUENT RELEASES AT COMMERCIAL NUCLEAR POWER STATIONS

Summary:

Provides a users guide for the NRC computer program XOQDOQ which implements Regulatory Guide 1.111. This NUREG supersedes NUREG-0324 which was published as a draft in September 1977. NUREG-0324 is currently cited in SRP Section 2.3.5 as describing the staff's standard long-term diffusion model for independent evaluation of routine or anticipated intermittent releases at nuclear power stations.

Microfiche No./Location: 15795/167

Potential Impact Number: 22845 **SRP Section Number:** 2.4.12

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: 50.55(e)(1)(iii) (old), 50.55(e)(1)(iii) (old), 50.55(e) (new)

Summary:

Regulation [10 CFR 50.55(e)(1)(iii)] currently cited in SRP Section 2.4.12 Branch Technical Position HGEB-1 subsection V(b) (Available from Recall ALLNOSRP Index):

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Each construction permit shall be subject to the following terms and conditions:...(e)(1) If the permit is for construction of a nuclear power plant, the holder of the permit shall notify the Commission of each deficiency found in design and construction, which, were it to have remained uncorrected, could have affected adversely the safety of operations of the nuclear power plant at any time throughout the expected lifetime of the plant, and which represents:.. (iii) A significant deficiency in construction of or significant damage to a structure, system, or component which will require extensive evaluation, extensive redesign, or extensive repair to meet the criteria and bases stated in the safety analysis report or construction permit or to otherwise establish the adequacy of the structure, system, or component to perform its intended safety function...

Current version of the regulation (10 CFR 50.55(e) available from Recall 9303all Index):

Each construction permit shall be subject to the following terms and conditions:...(e)(3) The holder of a facility construction permit subject to this part 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 must notify the Commission of the defect through a director or responsible officer or designated person as discussed in this section....(5) The notification requirements of this section 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 construction permit. Evaluation of potential defects and failures to comply and reporting of defects and failures to comply under this section satisfies the construction permit holder's evaluation and notification obligations under part 21 of this chapter and, satisfies the responsibility of individual directors or responsible officers of holders of construction permits issued under 50.23 of this chapter to report defects, and failures to comply associated with substantial safety hazards under section 206 of the Energy Reorganization Act of 1974.

Potential Impact Number: 22848 **SRP Section Number:** 12.2

Document: C&S: ANSI N237-76
Title: Radioactive Source Term for Normal Operation of Light Water Reactors

Summary:

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ANSI N237-1976 is cited in the Acceptance Criteria subsection as it relates to the establishment of typical long-term concentrations of principal radionuclides in fluid streams of light-water cooled nuclear power plants. The subject standard has been superseded by ANS 18.1, the latest version of which is dated 1984.

Potential Impact Number: 22851 **SRP Section Number:** 6.2.1.2

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

GE has committed to revise SSAR Table 6.2-2 to include the values of design differential pressure for each subcompartment in order to confirm that the margin between calculated and design peak differential pressure of 40 percent required by SRP Section 6.2.1.2 (II.B.5) is achieved for the ABWR. This 40-percent design margin is acceptable subject to the SSAR revision to reflect this commitment.

Potential Impact Number: 22853 **SRP Section Number:** 2.4.4

Document: C&S: ANSI N170
Title: Standards for Determining Design Basis Flooding at Power Reactor Sites

Summary:

Standard ANSI N170, relating to determination of design basis flooding at reactor sites including guidance for consideration of river flow conditions coincident with potential dam failures at sites located on streams or rivers, is cited, without specific identification of version/date, in subsection II and as Reference 22 in SRP Section 2.4.4. Reg. Guide 1.59, which is cited as guidance for SRP Section 2.4.4 reviews in subsection II, endorses ANSI N170-1976 (See Appendix A of the Reg. Guide) with exceptions.

This standard was retitled and redesignated in 1981 as ANSI/ANS 2.8-1981, "Determining Design Basis Flooding at Power Reactor Sites." The current version of this standard is ANSI/ANS 2.8-1992.

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Potential Impact Number: 22857 **SRP Section Number:** 5.4.1.1

Document: C&S: ASTM E208
Title: Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels

Summary:

SRP section 5.4.1.1 acceptance criteria specifies ASTM E-208-69 (1969) for determining the nil-ductility transition temperature. This standard has been revised and is now referenced as ASTM E-208-91 (1991) "Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels."

Potential Impact Number: 22858 **SRP Section Number:** 5.4.1.1

Document: C&S: ASTM A370
Title: Standard Test Methods and Definitions for Mechanical Testing of Steel Products

Summary:

SRP section 5.4.1.1 acceptance criteria specifies ASTM A-370-72 (1972) for determining the Charpy V-notch upper-shelf energy level in the "weak" direction. This standard has been revised and is now referenced as ASTM A-370-92 (1992) "Standard Test Methods and Definitions for Mechanical Testing of Steel Products."

Potential Impact Number: 22859 **SRP Section Number:** 5.4.1.1

Document: C&S: ASME III
Title: CLASS 1 AND 2 SYSTEMS

Summary:

SRP section 5.4.1.1 acceptance criteria for fracture toughness specifies that 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) is now designated as the critical or reference stress intensity factor K(Ir).

Potential Impact Number: 22860 **SRP Section Number:** 6.2.7

Document: C&S: ASME III
Title: CLASS 1 AND 2 SYSTEMS

Summary:

The ferritic containment pressure boundary materials are evaluated against Section III of the ASME Boiler and Pressure Vessel Code, Summer 1977 Addenda, Subsection NC, which is cited in Sections I, II and VI (Reference 4), of SRP Section 6.2.7. The latest version of this standard is dated 1992.

Subsection NE, " Class MC Components," establishes ASME Code Section III requirements for metal containment vessels.

Potential Impact Number: 22862 **SRP Section Number:** 6.2.3

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

As discussed in the DFSER, RAI Q430.32 and Q430.34 requested information on the ability of Standby Gas Treatment System (SGTS) to draw a negative pressure assuming that all lines that do not receive an isolation signal are open and assuming that the worst-case containment isolation valve (CIV) fails to close. In addition, the staff asked GE to identify the instrumentation used to ensure that the CIVs shut. GE's initial response did not specify the alarm capacity for these indicators and did not address alarms in the control room. The staff did not identify this as an open item.

In response to RAI Q430.32, GE revised SSAR Section 6.5.1.3.1 to state that a secondary containment draw-down analysis will be performed to demonstrate the capability of the SGTS to maintain the designed negative pressure following a LOCA, including inleakage from the open, nonisolated penetration lines identified during

construction engineering and in the event of the worst single failure of a secondary CIV to close. GE also added SSAR Section 6.5.5.1 to state that the COL applicant will perform an SGTS dose, functional damage, and draw-down analysis in accordance with SSAR Sections 6.5.1.2.3.7 and 6.5.1.3.1(5). The staff found GE's response acceptable. This was identified as COL Action Item 6.2.3-1 in the DFSER.

To further address this issue, GE submitted a list of secondary containment penetrations in SSAR Table 6.2-9. All piping and cable-tray penetrations will be sealed with a sealing compound for leakage and fire protection and are all provided with containment isolation features. To ensure that the SGTS is capable of maintaining the design negative pressure, GE stated that the draw-down analysis would assume 50-percent containment inleakage regardless of valve positions. During construction engineering, the COL applicant will identify the piping that could cause inleakage. SSAR Section 6.5.1.3.2 states that each SGTS fan is sized to individually establish a continuously negative differential pressure (considering the effect of wind) within 10 minutes after SGTS initiation. The dose analysis assumes direct leakage from the containment to the environs for twice the required draw-down period. The staff finds that the SGTS design includes some conservatism when considering any malfunction of secondary containment isolation features. The staff concludes that it does not need to evaluate the results of the draw-down analysis for its FSER safety finding. Therefore, COL Action Item 6.2.3-1 is acceptable as included in the SSAR.

Potential Impact Number: 22863 **SRP Section Number:** 6.2.3

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

Section 6.2.3 of the FSER references GDC 19 as it relates to the design of the secondary containment for fission product removal but refers to section 6.4 of the SER for discussion of the staff's evaluation of compliance with the radiation protection requirements of GDC 19.

Potential Impact Number: 22864 **SRP Section Number:** 6.2.1.1.B

Document: FINAL SER EPRI CH 5

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Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

In Section 2.4.1.3 of Revision 0 of Chapter 5 of the Evolutionary Requirements Document, EPRI stated that at the preliminary design stage, a margin will be provided between calculated peak pressure and design pressure of 10 percent for dry containments and 15 percent for pressure-suppression containments. This requirement has been deleted from Section 2.4.1.3. The deletion of this requirement is acceptable because the margin between calculated peak pressure and containment design pressure is only applicable at the construction permit stage of the licensing review. The evolutionary ALWR design will be licensed under 10 CFR Part 52; therefore, the information required will be that required in a final safety analysis report. The requirements of Section 8.2.4.2 of Chapter 5 conform to the NRC criteria in SRP Section 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments," and are, therefore, acceptable.

Potential Impact Number: 22865 **SRP Section Number:** 6.2.1.1.B

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In Item II.a of SRP Section 6.2.1.1.A, the staff states that the applicant should maintain a 10-percent margin during the construction permit stage of review and that at the operating license (OL) stage of review, the calculated peak pressure "should be approximately the same as at the construction permit stage." In Item II.2 of SRP Section 6.2.1.1.A, the staff cites GDC 50 as requiring "sufficient margin" for calculated peak containment pressure. The 1 percent margin between calculated peak primary containment pressure and its design value does not meet the requirements given in GDC 50 or SRP Section 6.2.1.1.A.

The applicant stated that the single-step licensing process for System 80+ precludes considering this review to be at the CP stage and that relaxing the 10 CFR Part 100 requirement for considering the Level B operating basis earthquake would provide additional margin in containment pressure. The System 80+ design is not final until construction and operation. Experience has shown that numerous factors such as design changes, as-built deviations, and analytical developments have affected other

plants, prompting applicants to reanalyze the containment response to LOCAs and MSLBs and increase the calculated peak containment pressure. Furthermore, a "final design" should incorporate a sufficiently large margin in peak calculated containment pressure to accommodate such future changes that might increase the calculated peak containment pressure.

Potential Impact Number: 22867 **SRP Section Number:** 6.1.2

Document: FINAL SER EPRI CH 6
Title: EPRI URD-EPD: BUILDING DESIGN AND ARRANGEMENT

Summary:

EPRI states that ANSI N101.2-1972, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," and American Society for Testing and Materials (ASTM) D3842.80 provide qualification requirements for coatings. SRP Section 6.1.2, "Protective Coating Systems (Paints) - Organic Materials," states that applicants should also meet the quality assurance requirements in ANSI N101.4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities." EPRI has revised Section 4.3.2.7 of Chapter 6 to require that paints and coatings that will be exposed to the containment atmosphere be qualified in accordance with ANSI N101.4-1972 (including the quality assurance requirements) and ASTM D3842.80. The staff concludes that these revisions are acceptable; therefore, the DSER confirmatory issue is closed.

Section 4.3.2.7 of Chapter 6 of the Evolutionary Requirements Document had stated that paints and coatings that will be exposed to the containment atmosphere will be qualified in accordance with ANSI N101.2-1972 and ASTM D3842-80 to prevent the degradation of emergency core cooling system pump performance as a result of blockage of sump strainers and screens by unqualified coatings that may come off surfaces during a loss-of-coolant accident (LOCA). SRP Section 6.1.2 states that paints and coatings should also meet the quality assurance requirements in ANSI N101.4-1972. The staff has verified that EPRI has revised Section 4.3.2.7 of Chapter 6 to require that paints and coatings that will be exposed to the containment atmosphere be qualified in accordance with ANSI N101.4-1972 (including the quality assurance requirements) and ASTM D3842-80. The staff concludes that the requirements of Section 4.2.10 of Chapter 6 of the Evolutionary Requirements Document represent good design practice for minimizing the spread and facilitating the cleanup of

radioactive contamination, conform with the guidelines of Regulatory Guide 8.8 for radiation protection, and are, therefore, acceptable.

Potential Impact Number: 22869 **SRP Section Number:** 6.1.2

Document: C&S: ANSI N101.2-72
Title: Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities

Summary:

ANSI N101.2-1972 is cited in SRP 6.1.2, Section II and listed in Section VI as Reference 2. ANSI N101.2-1972 has been withdrawn.

Potential Impact Number: 22870 **SRP Section Number:** 6.2.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 19

Summary:

Criterion 19 -- Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

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The staff found that the ABWR secondary containment complies with GDC 19 in the ABWR DFSER.

Potential Impact Number: 22873 **SRP Section Number:** 2.4.6

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Locations: 52.17(a)(1), 52.17(a)(1), 52.47(a)(1)(iii), 52.79(b), App. M, Para. 2 and 9, App. N, Para. 2 and 3, App. O, Para. 3, App. Q, Para. 1

Summary:

Applications pursuant to 10CFR52 must contain technically relevant site specific information as specified by the cited sections.

Potential Impact Number: 22874 **SRP Section Number:** 2.4.6

Document: DRAFT SER CE80 CH 2
Title: SITE ENVELOPE CHARACTERISTICS

Summary:

Site bounding parameters are reviewed in SER Section 2.6. Applications referencing the CE80+ standard design must provide site- specific information on the probable maximum tsunami flood level and demonstrate that the applicable CE80+ SAR parameters are met.

Potential Impact Number: 22875 **SRP Section Number:** 2.4.6

Document: DRAFT FINAL SER ABWR CH 2
Title: SITE CHARACTERISTICS

Summary:

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A bounding site parameter for the probable maximum tsunami is provided in the ABWR SSAR and applications referencing the ABWR SSAR are to show conformance to this bounding value.

Potential Impact Number: 22877 **SRP Section Number:** 2.4.5

Document: DRAFT SER CE80 CH 2
Title: SITE ENVELOPE CHARACTERISTICS

Summary:

Site bounding parameters are reviewed in SER Section 2.6. Applications referencing the CE80+ standard design must provide site- specific information on the probable maximum tsunami flood level and demonstrate that the applicable CE80+ SAR parameters are met.

Potential Impact Number: 22878 **SRP Section Number:** 2.4.5

Document: DRAFT FINAL SER ABWR CH 2
Title: SITE CHARACTERISTICS

Summary:

A bounding site parameter for the probable maximum tsunami is provided in the ABWR SSAR and applications referencing the ABWR SSAR are to show conformance to this bounding value.

Potential Impact Number: 22887 **SRP Section Number:** 6.3

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

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Summary:

In SECY-93-087 the staff requested that the Commission approve certain positions related to severe accidents in which molten debris could be ejected from the reactor vessel under high- pressure. One of the positions addressed design measures that could prevent a high-pressure melt ejection during a severe accident. The position indicates that evolutionary and passive ALWR designs should provide a reliable depressurization system. The position proposed by the staff was approved by the Commission in an SRM dated July 21, 1993.

Potential Impact Number: 22889 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

During an earlier review, the staff found that the testing of steam isolation valves (F035 and F036) leading to the RCIC turbines in currently operating BWRs did not include actual operating conditions such as a differential pressure of about 7,000 kPa (1,000 psig) and a temperature of 286 C (546 F) expected during a steam pipe break downstream of the valves. Thus, there is no verification that the isolation valves will close during a break as a result of dynamic steam flow forces. Generic Issue (GI)-87, "Failure of HPCI Steamline Without Isolation," addresses this concern. Tests performed as part of the NRC effort to resolve GI-87 have reinforced concerns about the operability of motor-operated valves (MOVs) under these design-basis conditions. The staff concerns regarding operability of MOVs are given in Generic Letter 89-10 (June 1989) and its supplements. GE assumed closure of these valves in the steamline break analysis. Therefore, this functional requirement is incorporated into the ITAAC. In the DFSER the staff noted that the COL applicant referencing the ABWR design should verify test data showing the steam isolation valves (F035 and F036) will isolate under actual operating conditions of a differential pressure at about 7,000 kPa (1,000 psig) and a temperature of 286 C (546 F). This was DFSER COL Action Item 5.4.6-1. Since verification of the valves performance by the preoperational and power ascension testing is discussed in Chapter 14 of the SSAR, it need not be specified as a COL action item.

GE originally submitted the design description and the ITAAC for the RCIC system. At the time the DFSER was issued, the ITAAC review was in progress. Therefore, this

was DFSER Open Item 5.4.6-1. GE has since provided a revised design description and ITAAC. The adequacy and acceptability of the ABWR design description and ITAAC are evaluated in Chapter 14.3 of this report.

Potential Impact Number: 22906 **SRP Section Number:** 5.4.12

Document: C&S: IEEE 344
Title: Recommended Practice for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations

Summary:

The SRP, in specific acceptance criterion 10, cites IEEE 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." Regulatory Guide 1.100, Revision 2 endorses IEEE 344-1987.

Potential Impact Number: 22909 **SRP Section Number:** 5.4.6

Document: NRC GENERIC LETTER 89-10 Sup 3
Title: 'CONSIDERATION OF THE RESULTS OF NRC-SPONSORED TESTS OF MOTOR-OPERATED VALVES'

Summary:

BWR licensees were requested to perform a plant-specific safety assessment to verify that the generic safety assessments performed by the NRC staff and the BWR Owners' Group were applicable. In performing the plant-specific safety assessment, BWR licensees were requested to address factors such as consideration of functional valve test results; operating procedures and emergency operating procedures; the conduct of training; current torque switch bypass settings including the potential for motor overload on a first attempt to close the valve; leak detection capabilities; inspection programs for erosion-corrosion and intergranular stress corrosion cracking; water-hammer prevention practices; the environmental qualification of the MOVs and other nearby equipment; radiological consequences both on and off the plant site that could result from a pipe leak or break; and probabilistic risk considerations.

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Potential Impact Number: 22910 **SRP Section Number:** 6.2.5

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-16 the staff addressed hydrogen generation and control provisions for future designs. Because of the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, the staff recommended in SECY 90-16 that evolutionary ALWRs should be designed, as a minimum, to satisfy the following criteria:
- accommodate hydrogen generation equivalent to a 100-percent metal-water reaction of the fuel cladding;
- limit containment hydrogen concentration to no greater than 10 percent; and,
- provide containment-wide hydrogen control (such as igniters or inerting)

The staff recommended that the Commission approve its position that the requirements of 10 CFR 50.34(f)(2)(ix) remain unchanged for the evolutionary ALWRs. In the SRM for SECY 90-016 the Commission approved the staff's position that the requirements of 10 CFR 50. This issue was also addressed in SECY 93-087 with the only change being extending the above criteria to the passive plants.

Potential Impact Number: 22911 **SRP Section Number:** 6.2.5

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.47(a)(1) (ii) and 52.47(a)(1)(iv)

Summary:

(a) The requirements of this paragraph apply to all applications for design certification.
(1) An application for design certification must contain:

(ii) Demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f);

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(iv) Proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority Generic Safety Issues which are identified in the version of NUREG-0933 current on the date six months prior to application and which are technically relevant to the design;

Potential Impact Number: 22915 **SRP Section Number:** 2.3.1

Document: C&S: ANSI A58.1
Title: Minimum Design Loads in Buildings and Other Structures

Summary:

Standard ANSI A58.1 is cited (1972 version) in SRP Section 2.3.1 Acceptance Criteria, subsection II, as specific criteria necessary to meet the Commission's regulations (guidance) with respect to determining operating basis wind velocity (fastest mile of wind). The standard is also cited as Reference 9 in subsection VI and in Review Procedures, subsection III.2, as guidance for checking snow and ice load adequacy and for review of proposed extreme winds, specific wind vertical velocity distribution, and gust factors.

ANSI A58.1 has been revised and redesignated as ANSI/ASCE-7-1988.

Potential Impact Number: 22916 **SRP Section Number:** 2.3.1

Document: C&S: ASCE 7-88
Title: Minimum Design Loads for Buildings and Other Structures

Summary:

Standard ANSI A58.1 is cited (1972 version) in SRP Section 2.3.1 Acceptance Criteria, subsection II, as specific criteria necessary to meet the Commission's regulations (guidance) with respect to determining operating basis wind velocity (fastest mile of wind). The standard is also cited as Reference 9 in subsection VI and in Review Procedures, subsection III.2, as guidance for checking snow and ice load adequacy and for review of proposed extreme winds, specific wind vertical velocity distribution, and gust factors.

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ANSI A58.1 has been revised and redesignated as ANSI/ASCE-7-1988.

Potential Impact Number: 22919 **SRP Section Number:** 2.4.14

Document: C&S: ANSI N170-76
Title: Standards for Determining Design Basis Flooding at Power Reactor Sites

Summary:

American National Standards Institute (ANSI) Standard ANSI N170, "Standards for Determining Design Basis Flooding at Power Reactor Sites" (1976), is listed in Section VI as Reference 4, but is not otherwise cited in SRP Section 2.4.14. SRP Section 2.4.14 cites compliance with Reg. Guide 1.59, Position 1, as a basis for determining that no Technical Specifications or emergency procedures related to flooding are needed to ensure safety-related functions. Reg. Guide 1.59 indicates that previous guidance for determining probable maximum and seismically-induced floods on streams and in coastal areas has been replaced by ANSI N170-1976 with exceptions as noted in the Reg. Guide. In addition to methodology for determining design-basis stream flooding, ANSI N170-1976 contains methodology for estimating probable maximum surges and seiches at estuaries and coastal areas on oceans and large lakes. ANS 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.

Potential Impact Number: 22920 **SRP Section Number:** 15.2.7

Document: DRAFT SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

In accordance with the requirements of GDC 17, loss of offsite power (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. For some transient analyses (such as the loss of condenser vacuum event analysis), the applicant did not postulate the unavailability of offsite power as part of the event, but rather took it as an additional single failure to show limiting fault criteria were not exceeded. The applicant is required

to discuss each of the transient and accident analyses in the CESSAR to justify that the analyses meet the GDC 17 requirements given above. If the existing analyses do not meet the GDC 17 requirements, the applicant should reanalyze the transient and accident analyses in accordance with GDC 17 and submit the results for the staff to review.

Potential Impact Number: 22921 **SRP Section Number:** 15.2.7

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 17

Summary:

An onsite electric power system 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 anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Potential Impact Number: 22922 **SRP Section Number:** 15.2.7

Document: DRAFT SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

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In CESSAR Section 15.2, the applicant analyzes various transients resulting from a decrease in heat removal by the secondary system and identifies the limiting cases for the consideration of integrity of RCS primary system boundary and fuel rod cladding to withstand the consequences of transients. With consideration of a single failure, the applicant noted that the loss of condenser vacuum (LOCV) event with a loss of offsite power (LOOP) following a turbine trip is the most limiting case with respect to both peak RCS pressure and low DNBR. With consideration of the worst single failure of a LOOP, the limiting fuel cladding damage resulting from the LOCV event is less than 1.8 percent of total fuel rods. As required by GDC 17 (discussed in Section 15.1), a LOOP should be included in the analysis and the LOCV event with LOOP should be analyzed as a moderate-frequency event, and therefore, the calculated DNBR should meet the 95/95 DNBR limit as discussed in Section 4.4 of this DSER. Since the fuel failure was predicted, the staff concludes that the results of the LOCV are not acceptable. The applicant should provide acceptable results using the approved methods for the LOCV event analysis for the staff to review.

Potential Impact Number: 22923 **SRP Section Number:** 2.3.3

Document: FINAL SER EPRI CH 9
Title: EPRI URD-EPD: SITE SUPPORT SYSTEMS

Summary:

With regard to onsite meteorological instrumentation, the Evolutionary Requirements Document specifies the use of Regulatory Guides 1.23 and 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Access Plant and Environs Conditions During and Following an Accident."

The staff finds that the use of Regulatory Guides 1.23 and 1.97 for defining the system's capability to monitor wind speed and wind direction and to determine atmospheric stability is acceptable.

Section 4.3.6 of Chapter 9 states that the Environmental Monitoring System (EMS) will be energized from a noninterruptible power source from the onsite ac system and will comply with Regulatory Guide 1.97. Offsite EMS equipment will be provided with backup battery power.

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Section 4.4.4 of Chapter 9 states that testing and calibration requirements will be derived from manufacturer's recommendations and from Regulatory Guides 1.23 and 1.97.

Section 4.4.5 of Chapter 9 states that EMS instrument ranges will be consistent with alarm setpoints and with Regulatory Guides 1.23 and 1.97. Separate instruments with overlapping ranges will be used if the required range results in inadequate sensitivity under normal conditions.

The staff concluded that the requirements of Section 4 of Chapter 9 are, to the extent described, consistent with the regulatory criteria in Regulatory Guides 1.23 and 1.97. They do not conflict with current regulatory guidelines and are acceptable.

Potential Impact Number: 22924 **SRP Section Number:** 9.5.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff identified the following items as important review considerations for communications systems:
1. Extensive wireless communication will increase the emphasis on consideration of electromagnetic interference (EMI) in the design of the man-machine interface system (M-MIS).
2. There will be a frequency allocation plan to ensure that there is no interference between communication systems and to ensure compatibility with the EMI/RFI measures taken by the electronics and computer designers. There will be no interference between communications systems and M-MIS equipment. System designers are required to define the worst-case electromagnetic emissions from the communications equipment, including the type, magnitude, frequency content, and locations. All potential uses of the communications equipment will be considered, including maintenance activities. The communications system will also be protected from the M-MIS equipment to the extent that clear communications will be maintained;
3. The private automatic business exchange (PABX) is independent of the intraplant public address (PA) system and the intraplant sound-powered telephone system. The staff evaluated the inspection, test, analysis and acceptance criterion (ITAAC) to verify the independence of these systems;

4. The plant designer must provide adequate means to alert personnel in high-noise environments that they are to use the communication systems. The COL applicant will install noise-canceling devices or sound isolation booths or both in noisy areas and include a PA system in the communications system;
5. Specialized system portable and wireless transmitters/receivers are provided for use in operations requiring respirators or underwater diving equipment. This is acceptable to the staff.; and
6. All communications systems are inspected, checked, and tested for operability after installation to ensure proper operation and coverage. Normal and continued use of the systems provides the basis for inspections of the systems. Inspection of all communications systems after installation to ensure their proper functioning is acceptable.

Potential Impact Number: 22925 **SRP Section Number:** 9.5.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

In addition to requiring communication with offsite law enforcement authorities, in 10 CFR 73.55(f), the staff requires intraplant security communications. The intraplant security communications requirement (10 CFR 73.55[f][1]) specifies that plant security officers must be capable of continuous communication with the central and secondary alarm stations. In the CESSAR, the applicant addressed a security radio system for offsite communications, but not for onsite communications with security officers.

Based upon EPRI guidance for ALWR communications systems, the staff maintained that the plant designer should ensure that effective communication is not impeded by problems of transmission through thick concrete walls, high-noise areas, personnel protective equipment, too few communication lines, or interference with or from other electronic or electrical equipment. The staff asked the applicant to detail consideration given to assuring the capability for security officers to be in continuous intraplant communication at System 80+ standard plants. In response, the applicant stated that plant security is expected to depend on wireless intraplant communications, but that the plant owner/operator will choose the system. The staff does not consider this response adequate since the applicant does not address design provisions that may be necessary to allow this capability, such as shielding of instrument transmitters and logic cabinets from radio frequency interference (RFI), fiber-optic cabling, and radio

repeaters within buildings, and does not provide reasons for deviating from the EPRI ALWR URD. NRC Information Notice 83-83 stated: "As newer plants are built that use more solid state equipment...more cases of RFI by portable radio transmitters are likely to result.... If plant operations make the use of portable radio transmitters near RFI-sensitive equipment either necessary or likely in an emergency, then administrative prohibitions are not adequate and the licensee should consider hardware fixes." This was designated as DSER Open Item 9.5.2-3.

Subsequently, in Amendment J to CESSAR-DC, ABB-CE added the following statement in Section 13.6: "The security communications subsystem shall meet the following requirements:....Each on-site security officer, watchman, or armed response individual shall be provided with continuous communications with an individual in each continuously manned alarm station. This may be accomplished by using multi-frequency radio or microwave-transmitted two-way voice communications." Requiring continuous wireless communication between security officers, ensures adequate communication for the security organization. This additional information is acceptable to the staff. On this basis, DSER Open Item 9.5.2-3 is resolved.

Potential Impact Number: 22926 **SRP Section Number:** 9.5.2

Document: FINAL SER EPRI CH 10
Title: EPRI URD-EPD: MAN-MACHINE INTERFACE SYSTEMS

Summary:

Requirements for specific security system communications are included in Chapter 9 of the Evolutionary Requirements Document. The staff agrees that the general communication requirements of Chapter 10 are also applicable to security communication systems, as clarified by EPRI.

EPRI states a preference for telephone-type dial-up wireless equipment, with a requirement for open channel or "party line" communications for situations that require it. The staff concluded that, although the requirements intended to ensure effective communications from any part of the plant are important to security communications and address concerns discussed in NRC Information Notice 83-83, "Use of Portable Radio Transmitters Inside Nuclear Power Plants", the dial-up telephone-type portable radio preferred by EPRI might not be suitable equipment for security armed response personnel. Explicit inclusion of security communications in the analysis and definition of

specific communication needs required by the EPRI URD adequately addresses the staff's concern.

Potential Impact Number: 22928 **SRP Section Number:** 9.5.2

Document: FINAL SER EPRI CH 10
Title: EPRI URD-EPD: MAN-MACHINE INTERFACE SYSTEMS

Summary:

EPRI proposed extensive use of wireless communications systems for evolutionary plants. This will increase the emphasis on consideration of electromagnetic interference (EMI) in the design of the M-MIS. The use of this type of communication system is not specifically excluded in NRC regulations and may be necessary to meet the requirements of 10 CFR Part 73; therefore, it is acceptable.

EPRI specifies that the designer will develop a frequency allocation plan to ensure that there is no interference between communication systems and to ensure compatibility with the EMI/radiofrequency interference (RFI) measures taken in electronic and computer designs. The staff concluded that this was acceptable and that EPRI should provide guidance on a frequency allocation plan. EPRI responded that this was not practical. This is acceptable because it does not conflict with current regulatory guidance. The staff, however, will evaluate this item during its review of an individual application for FDA/DC.

EPRI states that the communication system will be protected from the M-MIS equipment to the extent that clear communications are maintained. The staff concluded that this was acceptable and that EPRI should provide design guidance on interference between communication systems and M-MIS equipment. EPRI responded that the designer is required to design the communication equipment and all M-MIS electronic equipment so that there will be no interference between the communication system and the M-MIS equipment. This is acceptable because it does not conflict with current regulatory guidance. The staff, however, will evaluate this item during its review of an individual application for FDA/DC.

EPRI requires an extensive, fairly powerful, portable, wireless communication capability. This requires extensive EMI/RFI testing of all safety electronics and probably should require testing of all control electronics for susceptibility. Existing equipment has shown definite susceptibility to interference from wireless systems significantly less

powerful than the proposed system. EPRI agreed with the staff that testing was required and added a test requirement.

Potential Impact Number: 22929 **SRP Section Number:** 9.5.2

Document: FINAL SER EPRI CH 10
Title: EPRI URD-EPD: MAN-MACHINE INTERFACE SYSTEMS

Summary:

The design of portable communication equipment will be integrated with the design of protective equipment, including diving suits and respirators, so that personnel wearing this protective equipment will be able to communicate effectively. EPRI states that the design will apply the guidance developed by EPRI on voice communication systems that are compatible with respiratory protection (EPRI NP-6559). The staff concludes that this is good design practice and is acceptable.

Potential Impact Number: 22931 **SRP Section Number:** 9.5.2

Document: FINAL SER ABWR CH 7
Title: INSTRUMENTATION AND CONTROLS

Summary:

Per the DFSER, there was a provision in the SSAR to design and provide a telephone communication system using portable, sound-powered telephones. This was the responsibility of the COL Applicant and was COL Action Item 7.7.1.15-1. There is no discussion of this COL Action Item or its resolution in the FSER.

The communication systems do not have a safety-related function. However, they are used during emergencies and because the communication system is required when plant control is at the remote shutdown station, its availability must be demonstrated assuming a main control room fire. This was DFSER Open Item 7.7.1.15-1. GE subsequently revised SSAR Chapter 9.5.2 to described the survivability of the sound-powered phones in the event of a control room fire. This item is therefore resolved.

GE also did not identify the EMI radiation levels or frequency range for the communication transmitters and receivers which may be installed in the plant. In addition, GE did not define the sensitivity of the safety-related computer systems to electromagnetic fields. Therefore, a test program with field measurements and operational descriptions is required if spurious effects upon safety-related I&C equipment due to communications is to be avoided. This was identified as DFSER Open Item 7.7.15-2. This will be verified as part of the EMI ITAAC that is included in the instrumentation and controls CDM. Therefore, this item is resolved. Based on the above, the staff concludes that the design of the paging communication system ensures its availability as required, and is, therefore, acceptable.

In the DFSER, it was stated that the staff had reviewed commitments in the SSAR concerning the design criteria and design bases for the communication systems. Provided they meet the ITAAC and DAC requirements, these commitments satisfied staff positions and industry standards, and enabled the systems to provide communications to plant personnel during accidents or incident conditions. Subject to the resolution of the Open Items 7.7.1.15-1 and 7.7.1.15-2 and COL Action Item 7.7.1.15-1, the staff concluded that the design of the paging communication system will conform to all applicable guidelines and staff positions. There are no similar conclusions in the FSER.

Potential Impact Number: 22932 **SRP Section Number:** 2.3.3

Document: NUREG 0737 SUPPLEMENT 1 Revision Date: 1/1/83
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS - REQUIREMENTS FOR EMERGENCY RESPONSE CAPABILITY

Summary:

The requirements and guidance contained in Supplement 1 to NUREG-0737 replace the corresponding requirements in the affected NUREG-0737 items. One of these NUREG-0737 items is III.A.2.2 on Meteorological Data. NUREG-0737 Supplement 1 contains guidance on meteorological measurements and information necessary for ensuring emergency response capability.

Potential Impact Number: 22933 **SRP Section Number:** 9.2.5

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Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

Standard Review Plan (SRP) Section 9.2.5, 'Ultimate Heat Sink,' states that the design of decay heat removal systems should be evaluated against Branch Technical Position (BTP) ASB 9.2, which requires an additional 20-percent uncertainty factor be included for the first 1000 seconds following shutdown and 10 percent between 1000 seconds and 10 million seconds. However, a recent comparison to the ORIGEN code (described in an attachment to EPRI's letter of August 16, 1988) has shown ANS 5.1 to be a conservative predictor of decay heat generation. On this basis, the staff concludes 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.

Potential Impact Number: 22935 **SRP Section Number:** 9.2.5

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Table B.1-2 in Appendix B to Chapter 1 indicates a commitment to comply with GL 89-13, 'Service Water Systems Problems Affecting Safety Related Equipment,' the closure document for the technical resolution of Issue 51 for existing plants. EPRI considers GL 89-13 to be appropriate for ALWRs.

The Evolutionary Requirements Document states that eliminating the possible effects of fouling and blockage of the SWS and heat sinks is a design goal. The plant designer is given specific requirements and guidance for achieving this goal, including instructions to consider designs and new requirements that further mitigate the effects of fouling. The final design of the ALWR heat sinks and water flow systems will avoid or minimize, as achievable, the problems described in this section. EPRI's position is that implementation of these requirements should resolve Issue 51 for the ALWR.

Although the Evolutionary Requirements Document includes a commitment to comply with GL 89-13, the staff identified concerns which have not yet been addressed by EPRI, including : the need for an ongoing program of surveillance and control to reduce

fouling problems in open-cycle service water systems and recent biofouling problems due to zebra mussels in cooling water systems and intake structures.

Potential Impact Number: 22938 **SRP Section Number:** 9.2.5

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

Issue 51: Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems

This issue addresses the subject of service water system (SWS) fouling at operating plants by aquatic bivalves. The SWS is the UHS that, during an accident or transient, cools the reactor building component cooling water heat exchangers, which in turn cool the RHR heat exchangers, as well as provide cooling for safety-related pumps and area cooling coils. Fouling of the safety-related SWS either by mud, silt, corrosion products, or aquatic bivalves has led to plant shutdowns, reduced power operation for repairs and modifications, and degraded modes of operation. The review criteria for this issue consist of elimination of the possible effects of fouling of SWS and UHSs.

The staff studied the conditions that allow fouling and compared alternative surveillance and control programs to minimize SWS fouling. The staff's technical findings were published in NUREG/CR-5210.

Potential Impact Number: 22942 **SRP Section Number:** 9.2.5

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

GSI-51 addresses fouling of safety-related service water systems (SSWSs) by either mud, silt, corrosion products, or aquatic bivalves. This problem has led to plant shutdowns, reduced power operation for repairs and modifications, and degraded modes of operation in nuclear power plants.

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The following are SSWS design features and interface requirements to minimize fouling of the CCWS heat exchangers and the SSWS piping, prevent flow blockage, and facilitate the maintenance of clean conditions.

- When required by the site-specific water chemistry and environmental regulations, the ultimate heat sink water must be chemically treated to reduce organic and non-organic fouling, corrosion, and scaling, and to keep mud and silt in suspension (see CESSAR Section 9.2.5.2).

- The station service water intake structure will be visually inspected, once per refueling cycle, for macroscopic biological fouling organisms, sediment, and corrosion. Inspections should be performed either by scuba divers or by dewatering the intake structure or by comparable methods. Any fouling accumulations should be removed. Also, samples of water and substrate will be collected annually to determine if biological fouling organisms have populated the water source. Upon the detection of biological fouling organisms, appropriate corrective action, such as the modification of the chemical treatment program, should be taken. However, consideration must be given to environmental regulations.

The staff will review site-specific aspects of the resolution of GSI-51 as when it reviews the site-specific application. Site-specific reviews will include a maintenance and inspection program(s) and the determination of whether or not the referencing applicant commits to meet all the applicant-identified interface requirements (COL Action Item 20.2-2).

Potential Impact Number: 22943 **SRP Section Number:** 9.2.5

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue addressed the subject of service water system (SWS) fouling at operating plants primarily by aquatic bivalves. As a result of the NRC concern for the effects on safety of open cycle water system fouling, IE Bulletin 81-03 was issued. Responses to this bulletin revealed that bivalves were observed at approximately 45% of all sites. The following related issues have been combined with the issue of whether or not the staff should develop requirements for improving the reliability of open cycle water systems:

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Issue 32, 'Flow Blockage in Essential Equipment Caused by Corbicula,' and Issue 52, 'SSW Flow Blockage by Blue Mussels.'

The recommended solution to the issue was the implementation of a baseline fouling program which was issued to licensees in Generic Letter No. 89-13. Thus, this issue was RESOLVED and requirements were established.

Potential Impact Number: 22945 **SRP Section Number:** 9.2.5

Document: NRC GENERIC LETTER 89-13 Sup 1
Title: SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

Summary:

Supplement 1 of GL 93-13 provides staff positions in response to licensees' questions related to the generic letter, including:

1. When determining whether a plant has clams in its source water, does consideration need to be given to the presence of clams in the plant vicinity (local environment) or solely in the water body (source of cooling water)? (Philadelphia Electric)

Answer The purpose of this recommended action is to enable a licensee or applicant to know if the service water system might be subject to biofouling. All potential sources of water for the service water system should be examined annually for the presence of biofouling species. If no waters in the local environment of a plant can get inside piping and components to cause biofouling degradation of the heat transfer function of the service water system, then such waters do not need to be sampled.

2. Enclosure 1 to Generic Letter 89-13 recommends varying requirements for service water systems based on intake structure configuration and location. In a service water system in which the suction point of the service water pumps is in the collecting basin for the ultimate heat sink (cooling tower) would the basin be considered the intake structure or would the source of basin makeup water be considered the intake structure? (Mississippi Power and Light)

Answer Each licensee or applicant should define the scope of the intake structure. The NRC considers that an intake structure would contain all the waters eventually used in the system. See the answer to Question II.B.1.

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Potential Impact Number: 22946 **SRP Section Number:** 9.2.5

Document: NRC GENERIC LETTER 89-13
Title: SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

Summary:

By March 1982, several reports of serious fouling events caused by mud, silt, corrosion products, or aquatic bivalve organisms in open-cycle service water systems had been received. These events led to plant shutdowns, reduced power operation for repairs and modifications, and degraded modes of operation. This situation led the NRC to establish Generic Issue 51.

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. This research program has recently been completed and the results have been published in NUREG/ CR-5210.

The NRC has concluded that the issue will be resolved when licensees and applicants implement either the recommended surveillance and control program described in Enclosure 1 or its equivalent for the service water system at their respective facilities. Many licensees experiencing service water macroscopic biofouling problems at their plants have found that these techniques will effectively prevent recurrence of such problems.

Potential Impact Number: 22948 **SRP Section Number:** 2.3.5

Document: DRAFT SER CE80 CH 2
Title: SITE ENVELOPE CHARACTERISTICS

Summary:

The applicant determined long-term relative concentrations at the low population zone (LPZ) boundary with the procedure specified in Regulatory Guide (RG) 1.145. In lieu of

site-specific data, the applicant conservatively assumed that the atmospheric conditions described in RG 1.4, Revision 2, for the period 4-30 days represented the annual average atmospheric conditions. The staff found the applicant's method for determining long-term (routine) values for X/Q (i.e., values specified in RG 1.4) are acceptable. Sites proposed for a System 80+ plant are expected to have annual average X/Q values that do not exceed 1.80 x 10-6 sec/m3 at the LPZ.

Potential Impact Number: 22952 **SRP Section Number:** 3.4.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 4

Summary:

Structures, systems and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents. These structures, systems and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Potential Impact Number: 22954 **SRP Section Number:** 15.6.5.A

Document: ADVANCE SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

In the DSER (SECY-91-355), the staff found that a credit may be given for the removal of fission products by the suppression pool provided that suppression pool DFs 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), which states that suppression pools are capable of scrubbing airborne fission products and that to ignore this capability would be an undue conservatism.

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Potential Impact Number: 22957 **SRP Section Number:** 4.3

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

The staff reviewed the reactivity coefficients used in the transient and accident analyses and concludes that they conservatively bound the expected values, including uncertainties. Further, moderator and power Doppler coefficients along with boron worth are measured as part of the startup physics testing to ensure that actual values are within those used in these analyses. Although applicant-predicted moderator temperature coefficient (MTC) values for the full range of expected operating conditions during the initial cycle are negative, the technical specifications (TS) initially proposed, allowed a positive MTC below 100-percent rated thermal power, presumably to encompass future reload cycles. However, the Electric Power Research Institute (EPRI) Advanced Light Water Reactor (ALWR) Evolutionary Plant Utility Requirements Document (URD) states that the fuel cycle design shall have a non-positive MTC over the entire fuel cycle when the reactor is critical. The staff agrees with this requirement and considers it prudent to upgrade the safety characteristics of advanced reactors to enable reactivity increases that cause a moderator temperature increase to be self-limiting, resulting in stable power operation. Therefore, the staff will require the MTC to be non-positive at all operating conditions. This was designated as DSER Open Item 4.3.2-1. In accordance with this staff requirement, ABB-CE has revised the MTC TS to ensure a non-positive value at all operating conditions (CESSAR-DC to Chapter 16). DSER Open Item 4.3.2-1 is, therefore, resolved.

Potential Impact Number: 22958 **SRP Section Number:** 4.3

Document: FINAL SER EPRI CH 4
Title: EPRI URD-EPD: REACTOR SYSTEMS

Summary:

Positive moderator coefficients are not prohibited by NRC requirements. However, the effect of allowing positive moderator coefficients must be evaluated throughout the fuel cycle for all transients and accidents, including anticipated transients without scram. In

the DSER for Chapter 4, the staff concluded that Section 7.3.1.2.3 of Chapter 4 did not adequately define when in the fuel cycle the positive moderator coefficients would be allowed above 50-percent power and what requirements would be used to decide when a positive moderator coefficient was acceptable. This was identified as an open issue in the DSER. In response to this DSER open issue, EPRI revised the Evolutionary Requirements Document to require that the fuel cycle design include a non-positive MTC over the entire fuel cycle when the reactor is critical. The staff concludes that this is acceptable; therefore, this DSER open issue is closed.

Potential Impact Number: 22959 **SRP Section Number:** 9.1.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Although the likelihood of the complete draining of the spent fuel pool was low, the use of high-density storage racks increased the probability of a zircaloy-cladding fire as compared with the use of low-density or open-frame racks. The staff concluded that the use of low-density storage racks was justified by a favorable value/impact ratio for new designs and recommended that EPRI make a commitment to use low-density storage racks, at least for the most recently discharged fuel. Therefore, the staff will expect the FDA/DC applicant to submit a design that uses low-density storage racks in the spent fuel pool for, as a minimum, the most recently discharged fuel.

Potential Impact Number: 22960 **SRP Section Number:** 9.1.2

Document: C&S: ANS 57.2
Title: Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants, Design

Summary:

ANS 57.2/ANSI N210-1976, which is cited as a guidance document, was revised in 1983. The standard provides guidance for designing spent fuel storage facilities at nuclear power stations.

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Potential Impact Number: 22961 **SRP Section Number:** 2.3.5

Document: ADVANCE SER ABWR CH 2
Title: SITE CHARACTERISTICS

Summary:

The staff also reviewed the enveloping meteorological dispersion values in Table 2.0-1 of the SSAR. The staff finds that GE determined these enveloping values using acceptable methodology given in RGs 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents from Light-Water-Cooled Power Reactors," Revision 1 and 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1. The methods used by GE for determining these enveloping values are described in Sections 2.6.7 and 15.4 of this report and, therefore, the staff finds the enveloping meteorological dispersion values acceptable.

Potential Impact Number: 22965 **SRP Section Number:** 2.4.4

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 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.

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 Impact Number: 22969 **SRP Section Number:** 6.3

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016 the staff recommended that the Commission approve the following positions on intersystem LOCAs for evolutionary ALWRs. The staff concluded that designing, to the extent practicable, low-pressure systems to withstand full RCS pressure is an acceptable means for resolving the intersystem LOCA issue. However, the staff believes that for those systems that have not been designed to withstand full RCS pressure, evolutionary ALWRs should provide:

(1) the capability for leak testing of the pressure isolation valves;

(2) valve position indication that is available in the control room when isolation valve operators are deenergized; and,

(3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

In the SRM for SECY 90-016, the Commission approved the staff's position on intersystem LOCA, provided that all elements of the low-pressure system are considered (including instrument lines, pump seals, heat exchanger tubes, and valve bonnets). This issue was also addressed in SECY 93-087 with the only change being extending the intersystem LOCA criteria to the passive plants.

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Potential Impact Number: 22970 **SRP Section Number:** 6.3

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016, the staff recommended that the Commission approve the position that the following provisions should be applied to all safety-related pumps and valves, and not limited to ASME Code Class 1, 2, and 3 components:

- Piping design should incorporate provisions for full flow testing (maximum design flow) of pumps and check valves.
- Designs should incorporate provisions to test motor-operated valves under design-basis differential pressure.
- Check valve testing should incorporate the use of advanced, non-intrusive techniques to address degradation and performance.
- A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced, non- intrusive techniques.

In the SRM for SECY 90-016, the Commission approved the staff's position as supplemented in the staff's response to the ACRS comments, dated April 27, 1990. The response agreed with the ACRS recommendations to emphasize the requirements of Generic Letter 89-10 with regard to evolutionary plants, to resolve check valve testing and surveillance issues, and indicate how these requirements are to be applied to evolutionary plants. The staff also agreed that the requirements should permit consideration of proposed alternative ways of meeting inservice and surveillance requirements. The Commission further noted that due consideration should be given to the practicality of designing testing capability, particularly for large pumps and valves.

Potential Impact Number: 22979 **SRP Section Number:** 6.2.1.1.B

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34 (f) (2) (ix)

Summary:

(ix) Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (f)(1)(xii) of this section is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that:

(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

Potential Impact Number: 22980 **SRP Section Number:** 6.2.4

Document: ADVANCE SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

GE responded to staff requests for additional information by modifying the SSAR. These changes, in response to issues raised under the review of 10 CFR 50.34(f)(2)(xv) may not have been required by the review of vent and purge systems under SRP 6.2.4 and BTP CSB 6-4. Based on the additional clarifying information provided in the SSAR, the staff concludes that the ABWR design meets TMI Action Plan Item II.E.4.4.

Potential Impact Number: 22981 **SRP Section Number:** 6.2.4

Document: DRAFT SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

The resolution of TMI Task Action Plan Items II.E.4.4(1), II.E.4.4(2), and II.E.4.4(3) is acceptable subject to (1) tests or analyses to show that containment purge or vent valves would shut without degrading containment integrity during the dynamic loads of a design-basis loss-of-coolant accident (Item II.E.4.4(1)) (Open Item 20.2-25), (2) analyses(s) to show that valve operability is assured against ascending differential

pressure and resulting dynamic loading of the design basis LOCA within the guidance provided above (Item II.E.4.4(3)) (Open Item 20.2-26), and (3) resolution of the open items in Sections 6.2.3 and 6.2.4 of this DSER.

Potential Impact Number: 22982 **SRP Section Number:** 9.5.1

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016 the staff recommended that the Commission approve the use of the enhanced fire protection guidance. The evolutionary ALWR designers must ensure that safe shutdown can be achieved, assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. Because of its physical configuration, the control room is excluded from this approach, provided an independent alternative shutdown capability that is physically and electrically independent of the control room is included in the design. Evolutionary ALWRs must provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practicable, that one shutdown division will be free of fire damage. Additionally, the evolutionary ALWR designers must ensure that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions.

In the SRM for SECY 90-016 the Commission approved the staff's position as presented in SECY 90-016 and as supplemented in by the staff's April 27, 1990 responses to the ACRS comments. The supplemented information addressed other related issues but did not result in any identified changes to the positions proposed by SECY 90-016. This issue was also addressed in SECY 93-087 with the only change being extending the enhanced fire protection criteria to the passive plants.

Potential Impact Number: 22985 **SRP Section Number:** 6.2.1.1.A

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

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Summary:

In Section 2.4.1.3 of Revision 0 of Chapter 5 of the Evolutionary Requirements Document, EPRI stated that at the preliminary design stage, a margin will be provided between calculated peak pressure and design pressure of 10 percent for dry containments and 15 percent for pressure-suppression containments. This requirement has been deleted from Section 2.4.1.3. The deletion of this requirement is acceptable because the margin between calculated peak pressure and containment design pressure is only applicable at the construction permit stage of the licensing review. The evolutionary ALWR design will be licensed under 10 CFR Part 52; therefore, the information required will be that required in a final safety analysis report. The requirements of Section 8.2.4.2 of Chapter 5 conform to the NRC criteria in SRP Section 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments," and are, therefore, acceptable.

Potential Impact Number: 22986 **SRP Section Number:** 6.2.1.1.A

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In Item II.a of SRP Section 6.2.1.1.A, the staff states that the applicant should maintain a 10-percent margin during the construction permit stage of review and that at the operating license (OL) stage of review, the calculated peak pressure "should be approximately the same as at the construction permit stage." In Item II.2 of SRP Section 6.2.1.1.A, the staff cites GDC 50 as requiring "sufficient margin" for calculated peak containment pressure. The 1 percent margin between calculated peak primary containment pressure and its design value does not meet the requirements given in GDC 50 or SRP Section 6.2.1.1.A.

The applicant stated that the single-step licensing process for System 80+ precludes considering this review to be at the CP stage and that relaxing the 10 CFR Part 100 requirement for considering the Level B operating basis earthquake would provide additional margin in containment pressure. The System 80+ design is not final until construction and operation. Experience has shown that numerous factors such as design changes, as-built deviations, and analytical developments have affected other plants, prompting applicants to reanalyze the containment response to LOCAs and MSLBs and increase the calculated peak containment pressure. Furthermore, a "final

design" should incorporate a sufficiently large margin in peak calculated containment pressure to accommodate such future changes that might increase the calculated peak containment pressure.

The applicant calculated a peak external differential of -12.6 kPa (-1.83 psid) on the primary containment caused by an inadvertent actuation of the containment spray system. This value provides a margin of 8.5 percent to the containment design value of -13.8 kPa (-2.0 psid). The applicant must justify the adequacy of this margin because it does not meet the 10-percent criterion previously discussed.

The margin in calculated peak internal primary containment pressure does not meet SRP Section 6.2.1.1.A and GDC 50. The applicant must justify the adequacy of this margin.

Potential Impact Number: 22990 **SRP Section Number:** 6.2.1.1.A

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.34(f)(2)(ix)

Summary:

Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (f)(1)(xii) of this section is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that:
...
(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

Potential Impact Number: 22992 **SRP Section Number:** 6.2.1.1.C

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

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Summary:

In Section 2.4.1.3 of Revision 0 of Chapter 5 of the Evolutionary Requirements Document, EPRI stated that at the preliminary design stage, a margin will be provided between calculated peak pressure and design pressure of 15 percent for pressure-suppression containments. This requirement has been deleted from Section 2.4.1.3. The deletion of this requirement is acceptable because the margin between calculated peak pressure and containment design pressure is only applicable at the construction permit stage of the licensing review. The evolutionary ALWR design will be licensed under 10 CFR Part 52; therefore, the information required will be that required in a final safety analysis report.

Potential Impact Number: 22993 **SRP Section Number:** 6.2.1.1.C

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

Standard Review Plan (SRP) Section 6.2.1.1.C, states that, for Mark III plants at the construction permit stage, the containment design pressure 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. GE's calculated drywell peak pressure for the FWLB is 268 kPa (39 psig). The design pressure for the drywell is 310 kPa (45 psig), which provides a margin of 15% above the peak calculated pressure in the drywell and is equal to the margin recommended in the SRP. Therefore, this design margin for containment pressure is acceptable.

The calculated wetwell peak pressure is 179 kPa (26 psig). The design pressure for the wetwell is 310 kPa (45 psig), which provides a margin of 42% above the peak calculated pressure in the wetwell. The calculated drywell-to-wetwell peak differential pressure is 110.3 kPa (16 psig) and the design drywell-to-wetwell differential pressure is 172.3 kPa (25 psig), which provides a design margin of 56%.

The staff concludes that the containment pressure transient following a LOCA in the ABWR containment is consistent with SRP 6.2.1 and, therefore, is acceptable.

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Potential Impact Number: 22994 **SRP Section Number:** 6.2.1.1.C

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.34(f)(2)(ix)

Summary:

Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (f)(1)(xii) of this section is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that:

...

(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

Potential Impact Number: 22995 **SRP Section Number:** 2.3.1

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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.

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Potential Impact Number: 22996 **SRP Section Number:** 2.4.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 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.

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 Impact Number: 23004 **SRP Section Number:** 3.7.3

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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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 responding to 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 (SSCs) when the OBE is eliminated from the design basis analysis. The staff evaluated the effect on safety of eliminating the OBE from the design load combinations for selected SSCs and has developed proposed criteria for an analysis using only the SSE in the design basis. 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.

In the SRM responding to SECY 93-087 the Commission also stated that the OBE will continue to be used as a threshold criterion for conducting a plant inspection following a seismic event at or above the OBE level.

Potential Impact Number: 23009 **SRP Section Number:** 9.3.5

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016, the staff recommended that the Commission approve the position that the following provisions should be applied to all safety-related pumps and valves, and not limited to ASME Code Class 1, 2, and 3 components:

- Piping design should incorporate provisions for full flow testing (maximum design flow) of pumps and check valves.
- Designs should incorporate provisions to test motor-operated valves under design-basis differential pressure.
- Check valve testing should incorporate the use of advanced, non-intrusive techniques to address degradation and performance.

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- A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced, non- intrusive techniques.

In the SRM for SECY 90-016, the Commission approved the staff's position as supplemented in the staff's response to the ACRS comments, dated April 27, 1990. The response agreed with the ACRS recommendations to emphasize the requirements of Generic Letter 89-10 with regard to evolutionary plants, to resolve check valve testing and surveillance issues, and indicate how these requirements are to be applied to evolutionary plants. The staff also agreed that the requirements should permit consideration of proposed alternative ways of meeting inservice and surveillance requirements. The Commission further noted that due consideration should be given to the practicality of designing testing capability, particularly for large pumps and valves.

In SECY 93-087 the staff clarified its position as follows:

- The staff recommended that full flow testing be conducted at maximum design flow with analysis to extrapolate to design pressure if it is not practicable to conduct the inservice pump testing at design flow and pressure.
- For valves the staff recommended, a qualification test (under design-basis differential pressure) be conducted prior to installation and inservice valve tests be conducted under the maximum practicable differential pressure and flow when it is not practicable to achieve design-basis differential pressure during an inservice test.

In the SRM for SECY 93-087 the Commission stated that it had no objection to the staff's position, but that further elaboration on this issue will be required from the staff.

It should be noted that all positions identified may not be applicable to the associated SRP section.

Potential Impact Number: 23011 **SRP Section Number:** 3.7.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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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 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.

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 Impact Number: 23012 **SRP Section Number:** 3.7.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY 93-087 the staff presented information on the use of seismic hazards curves to assess the seismic risk of a site proposed for a ALWR design. The evolutionary and passive ALWR designers have indicated that their applications will be consistent with the EPRI criteria for seismic hazards curves. However, based upon other reviews and estimates, the staff has concluded that the EPRI seismic hazard bounding curve is not sufficiently conservative. The discussion on seismic hazard curves provided in this section of SECY 93-087 is provided for information only.

Potential Impact Number: 23013 **SRP Section Number:** 3.2.1

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Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

This issue involved the classification of the main steam drain lines and bypass piping between the first normally-closed valve and the condenser hotwell as well as the hotwell itself in boiling water reactors. In SECY 93-087, the staff recommended that for both the evolutionary and passive ALWRs the Commission approve the position that neither the main steam drain and bypass lines from the first valve up to the condenser inlet, nor the piping between the turbine stop valve and the turbine inlet should be classified as safety-related or as seismic Category I. Rather, these lines should be analyzed using a dynamic seismic analysis to demonstrate structural integrity under SSE loading conditions. The turbine stop, control, and bypass valves and the main steam lines from the turbine control valves to the turbine shall meet all of the quality group and quality assurance guidelines specified in SRP Section 3.2.2, Appendix A. Further, that seismic analyses are performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after the SSE. In the SRM for SECY 93-087 the Commission approved the above described approach and positions for classifying the mainsteam lines and systems for boiling water reactors.

Potential Impact Number: 23014 **SRP Section Number:** 3.2.1

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 recommendation.

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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.

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 Impact Number: 23015 **SRP Section Number:** 2.5.4

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 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.

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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 Impact Number: 23016 **SRP Section Number:** 3.2.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

This issue involved the classification of the main steam drain lines and bypass piping between the first normally-closed valve and the condenser hotwell as well as the hotwell itself in boiling water reactors. In SECY 93-087, the staff recommended that for both the evolutionary and passive ALWRs the Commission approve the position that neither the main steam drain and bypass lines from the first valve up to the condenser inlet, nor the piping between the turbine stop valve and the turbine inlet should be classified as safety-related or as seismic Category I. Rather, these lines should be analyzed using a dynamic seismic analysis to demonstrate structural integrity under SSE loading conditions. The turbine stop, control, and bypass valves and the main steam lines from the turbine control valves to the turbine shall meet all of the quality group and quality assurance guidelines specified in SRP Section 3.2.2, Appendix A. Further, that seismic analyses are performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after the SSE. In the SRM for SECY 93-087 the Commission approved the above described approach and positions for classifying the mainsteam lines and systems for boiling water reactors.

Potential Impact Number: 23018 **SRP Section Number:** 3.6.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

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Summary:

Pipe rupture is a rare event that can be caused by errors in design, construction, or operation; unanticipated loads; or unanticipated corrosive environments. The staff notes that piping failures generally occur at high stress and fatigue locations, such as at the ends of a piping system where it connects to components nozzles.

Recent dynamic pipe tests conducted by EPRI and the NRC demonstrated that butt-welded piping can withstand seismic inertial loadings higher than an SSE without rupturing. Thus, the staff concludes that the likelihood of a double-ended pipe rupture caused by an OBE-level earthquake in a piping system designed to an SSE is remote. Operating experience has shown that pipe failures (splits, through-wall cracks, and double-ended pipe ruptures) are more likely to occur under conditions caused by normal operation. These conditions include erosion, corrosion, thermal constraint, fatigue, and operational transients.

The staff recommends 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. The staff recommends 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.

Further reduction in the number of postulated pipe rupture locations can be considered when compensatory measures are established to minimize the potential for pipe ruptures during normal operating and transient conditions (such as control of erosion/corrosion or use of upgraded piping materials). the guidelines for environmental qualification and compartment pressurization are currently based on the mechanistic break locations. Therefore, the staff proposed that the mechanistic pipe break and high-energy leakage crack locations determined by the piping high stress (without the OBE) and fatigue locations may be used for equipment environmental qualification and compartment pressurization purposes.

Potential Impact Number: 23019 **SRP Section Number:** 3.7.1

Document: SECY 93-087

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Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 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.

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 Impact Number: 23020 **SRP Section Number:** 3.7.1

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY 93-087 the staff presented information on the use of seismic hazards curves to assess the seismic risk of a site proposed for a ALWR design. The evolutionary and passive ALWR designers have indicated that their applications will be consistent with the EPRI criteria for seismic hazards curves. However, based upon other reviews and estimates, the staff has concluded that the EPRI seismic hazard bounding curve is not

sufficiently conservative. The discussion on seismic hazard curves provided in this section of SECY 93-087 is provided for information only.

Potential Impact Number: 23022 **SRP Section Number:** 5.4.7

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY90-016 the staff requested that the Commission approve the staff position that evolutionary PWR vendors propose design features to ensure high reliability of the shutdown decay heat removal system. In the SRM for SECY 90-016 the Commission approved the staff position, with the ACRS recommendation of April 26, 1990, that four additional specific items should be considered for mid-loop operation. The four additional specific items are as follows:

- Design provisions to help ensure continuity of flow through the core and residual heat removal system with low-liquid levels at the junction of the DHR system suction lines and the RCS.
- Provisions to ensure availability of reliable systems for decay heat removal. - Instrumentation for reliable measurements of liquid levels in the reactor vessel and at the junction of the DHR system suction lines and the RCS.
- Provisions for maintaining containment closure or for rapid closure of containment openings.

Potential Impact Number: 23023 **SRP Section Number:** 5.4.7

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016 the staff recommended that the Commission approve the following positions on intersystem LOCAs for evolutionary ALWRs. The staff concluded that

designing, to the extent practicable, low-pressure systems to withstand full RCS pressure is an acceptable means for resolving the intersystem LOCA issue. However, the staff believes that for those systems that have not been designed to withstand full RCS pressure, evolutionary ALWRs should provide:

(1) the capability for leak testing of the pressure isolation valves;

(2) valve position indication that is available in the control room when isolation valve operators are deenergized; and,

(3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

In the SRM for SECY 90-016, the Commission approved the staff's position on intersystem LOCA, provided that all elements of the low-pressure system are considered (including instrument lines, pump seals, heat exchanger tubes, and valve bonnets). This issue was also addressed in SECY 93-087 with the only change being extending the intersystem LOCA criteria to the passive plants.

Potential Impact Number: 23024 **SRP Section Number:** 5.4.7

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016, the staff recommended that the Commission approve the position that the following provisions should be applied to all safety-related pumps and valves, and not limited to ASME Code Class 1, 2, and 3 components:

- Piping design should incorporate provisions for full flow testing (maximum design flow) of pumps and check valves.
- Designs should incorporate provisions to test motor-operated valves under design-basis differential pressure.
- Check valve testing should incorporate the use of advanced, non-intrusive techniques to address degradation and performance.

- A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced, non- intrusive techniques.

In the SRM for SECY 90-016, the Commission approved the staff's position as supplemented in the staff's response to the ACRS comments, dated April 27, 1990. The response agreed with the ACRS recommendations to emphasize the requirements of Generic Letter 89-10 with regard to evolutionary plants, to resolve check valve testing and surveillance issues, and indicate how these requirements are to be applied to evolutionary plants. The staff also agreed that the requirements should permit consideration of proposed alternative ways of meeting inservice and surveillance requirements. The Commission further noted that due consideration should be given to the practicality of designing testing capability, particularly for large pumps and valves.

In SECY 93-087 the staff clarified its position as follows:

- The staff recommended that full flow testing be conducted at maximum design flow with analysis to extrapolate to design pressure if it is not practicable to conduct the inservice pump testing at design flow and pressure.
- For valves the staff recommended, a qualification test (under design-basis differential pressure) be conducted prior to installation and inservice valve tests be conducted under the maximum practicable differential pressure and flow when it is not practicable to achieve design-basis differential pressure during an inservice test.

In the SRM for SECY 93-087 the Commission stated that it had no objection to the staff's position, but that further elaboration on this issue will be required from the staff.

It should be noted that all positions identified may not be applicable to the associated SRP section.

Potential Impact Number: 23026 **SRP Section Number:** 15.2.1 - 15.2.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 17

Summary:

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An onsite electric power system 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 anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Potential Impact Number: 23027 **SRP Section Number:** 15.2.1 - 15.2.5

Document: ADVANCE SER CE CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

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 meet the GDC 17 requirements given above. If the existing analyses did meet the GDC 17 requirements, ABB-CE should have reanalyzed the transient and accident analyses in accordance with GDC 17 and submit the results for the staff to review. This was identified as DSER Open Item 15.1-5.

By letter dated December 18, 1992, ABB-CE agreed to comply with GDC 17, which requires that the LOOP not be treated as a single failure. ABB-CE included the results of reanalysis in the CESSAR-DC (Amendment R) Section 6.3.3 and Chapter 15. The staff reviewed the submittal and found that ABB-CE considered the LOOP in all the events analyzed and applied the acceptance criteria specified in the related SRP sections for the event with or without LOOP. The staff concludes that ABB-CE's

approach complies with the requirements of GDC 17. On this basis, DSER Open Item 15.1-5 is resolved.

Potential Impact Number: 23028 **SRP Section Number:** 15.8

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016 the staff requested that the Commission approve the staff position that diverse scram systems be provided for evolutionary ALWRs. In the SRM for SECY 90-016 the Commission approved the staff position. However, if the applicant can demonstrate that the consequences of an ATWS are acceptable, the staff should accept the demonstration as an alternative to the diverse scram system. This issue is also discussed in SECY 93-087 with no changes to the position indicated.

Potential Impact Number: 23035 **SRP Section Number:** 15.2.1 - 15.2.5

Document: C&S: ANSI N18.2-74
Title: Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants

Summary:

ANSI N18.2-1974 is cited in a parenthetical phrase in the Acceptance Criteria to define "moderate frequency" events. The current version of this standard is ANS 51.1-1983, which was reaffirmed in 1988.

Potential Impact Number: 23036 **SRP Section Number:** 15.2.1 - 15.2.5

Document: C&S: ANS N212-74
Title: Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants

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Summary:

ANS Trial Use Standard N212-1974 is cited in a parenthetical phrase in the Acceptance Criteria to define "moderate frequency" events. The current version of this standard is ANS 52.1- 1983, which was reaffirmed in 1988.

Potential Impact Number: 23040 **SRP Section Number:** 3.9.3

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016 the staff recommended that the Commission approve the following positions on intersystem LOCAs for evolutionary ALWRs. The staff concluded that designing, to the extent practicable, low-pressure systems to withstand full RCS pressure is an acceptable means for resolving the intersystem LOCA issue. However, the staff believes that for those systems that have not been designed to withstand full RCS pressure, evolutionary ALWRs should provide:

(1) the capability for leak testing of the pressure isolation valves;

(2) valve position indication that is available in the control room when isolation valve operators are deenergized; and,

(3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

In the SRM for SECY 90-016, the Commission approved the staff's position on intersystem LOCA, provided that all elements of the low-pressure system are considered (including instrument lines, pump seals, heat exchanger tubes, and valve bonnets). This issue was also addressed in SECY 93-087 with the only change being extending the intersystem LOCA criteria to the passive plants.

Potential Impact Number: 23041 **SRP Section Number:** 3.9.6

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Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016 the staff recommended that the Commission approve the following positions on intersystem LOCAs for evolutionary ALWRs. The staff concluded that designing, to the extent practicable, low-pressure systems to withstand full RCS pressure is an acceptable means for resolving the intersystem LOCA issue. However, the staff believes that for those systems that have not been designed to withstand full RCS pressure, evolutionary ALWRs should provide:

(1) the capability for leak testing of the pressure isolation valves;

(2) valve position indication that is available in the control room when isolation valve operators are deenergized; and,

(3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

In the SRM for SECY 90-016, the Commission approved the staff's position on intersystem LOCA, provided that all elements of the low-pressure system are considered (including instrument lines, pump seals, heat exchanger tubes, and valve bonnets). This issue was also addressed in SECY 93-087 with the only change being extending the intersystem LOCA criteria to the passive plants.

Potential Impact Number: 23044 **SRP Section Number:** 17.1

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016, the staff recommended that the Commission approve the position that features provided only for severe-accident protection need not be 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. In the SRM for SECY 90-016 the Commission approved the staff's position on equipment survivability. SECY 90- 016 further clarified 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 instances where safety-related equipment provided for DBAs 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 it is needed to perform its intended function. In addition, Appendices A and B to Regulatory Guide 1.155, "Station Blackout," provide guidance on the type of quality assurance activities and specifications which the staff concludes are appropriate for equipment utilized to prevent and mitigate the consequences of severe accidents. This issue was also addressed in SECY 93-087 with the only change being extending the above positions to the passive plants.

It should be noted that the above positions are related to severe accident issues and may not be directly applicable to this SRP section. However, this issue should be considered as a candidate for future work per IPD 7.0.

Potential Impact Number: 23045 **SRP Section Number:** 2.5.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 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.

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 Impact Number: 23046 **SRP Section Number:** 3.6.1

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 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.

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 Impact Number: 23047 **SRP Section Number:** 3.8.1

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Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 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.

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 Impact Number: 23048 **SRP Section Number:** 3.8.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 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.

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 Impact Number: 23049 **SRP Section Number:** 3.8.3

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 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.

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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 Impact Number: 23050 **SRP Section Number:** 3.8.4

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 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.

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 Impact Number: 23051 **SRP Section Number:** 3.8.5

Document: SECY 93-087

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Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 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.

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 Impact Number: 23052 **SRP Section Number:** 3.9.3

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 recommendation.

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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.

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 Impact Number: 23053 **SRP Section Number:** 3.10

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 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.

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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 Impact Number: 23055 **SRP Section Number:** 3.9.6

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY 90-016, the staff recommended that the Commission approve the position that the following provisions should be applied to all safety-related pumps and valves, and not limited to ASME Code Class 1, 2, and 3 components:

- Piping design should incorporate provisions for full flow testing (maximum design flow) of pumps and check valves.
- Designs should incorporate provisions to test motor-operated valves under design-basis differential pressure.
- Check valve testing should incorporate the use of advanced, non-intrusive techniques to address degradation and performance.
- A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced, non- intrusive techniques.

In the SRM for SECY 90-016, the Commission approved the staff's position as supplemented in the staff's response to the ACRS comments, dated April 27, 1990. The response agreed with the ACRS recommendations to emphasize the requirements of Generic Letter 89-10 with regard to evolutionary plants, to resolve check valve testing and surveillance issues, and indicate how these requirements are to be applied to evolutionary plants. The staff also agreed that the requirements should permit consideration of proposed alternative ways of meeting inservice and surveillance requirements. The Commission further noted that due consideration should be given to the practicality of designing testing capability, particularly for large pumps and valves.

In SECY 93-087 the staff clarified its position as follows:

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- The staff recommended that full flow testing be conducted at maximum design flow with analysis to extrapolate to design pressure if it is not practicable to conduct the inservice pump testing at design flow and pressure.
- For valves the staff recommended, a qualification test (under design-basis differential pressure) be conducted prior to installation and inservice valve tests be conducted under the maximum practicable differential pressure and flow when it is not practicable to achieve design-basis differential pressure during an inservice test.

In the SRM for SECY 93-087 the Commission stated that it had no objection to the staff's position, but that further elaboration on this issue will be required from the staff.

It should be noted that all positions identified may not be applicable to the associated SRP section.

Potential Impact Number: 23062 **SRP Section Number:** 2.5.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY 93-087 the staff presented information on the use of seismic hazards curves to assess the seismic risk of a site proposed for a ALWR design. The evolutionary and passive ALWR designers have indicated that their applications will be consistent with the EPRI criteria for seismic hazards curves. However, based upon other reviews and estimates, the staff has concluded that the EPRI seismic hazard bounding curve is not sufficiently conservative. The discussion on seismic hazard curves provided in this section of SECY 93-087 is provided for information only.

Potential Impact Number: 23064 **SRP Section Number:** 10.3

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

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Summary:

This issue involved the classification of the main steam drain lines and bypass piping between the first normally-closed valve and the condenser hotwell as well as the hotwell itself in boiling water reactors. In SECY 93-087, the staff recommended that for both the evolutionary and passive ALWRs the Commission approve the position that neither the main steam drain and bypass lines from the first valve up to the condenser inlet, nor the piping between the turbine stop valve and the turbine inlet should be classified as safety-related or as seismic Category I. Rather, these lines should be analyzed using a dynamic seismic analysis to demonstrate structural integrity under SSE loading conditions. The turbine stop, control, and bypass valves and the main steam lines from the turbine control valves to the turbine shall meet all of the quality group and quality assurance guidelines specified in SRP Section 3.2.2, Appendix A. Further, that seismic analyses are performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after the SSE. In the SRM for SECY 93-087 the Commission approved the above described approach and positions for classifying the mainsteam lines and systems for boiling water reactors.

Potential Impact Number: 23065 **SRP Section Number:** 10.4.1

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

This issue involved the classification of the main steam drain lines and bypass piping between the first normally-closed valve and the condenser hotwell as well as the hotwell itself in boiling water reactors. In SECY 93-087, the staff recommended that for both the evolutionary and passive ALWRs the Commission approve the position that neither the main steam drain and bypass lines from the first valve up to the condenser inlet, nor the piping between the turbine stop valve and the turbine inlet should be classified as safety-related or as seismic Category I. Rather, these lines should be analyzed using a dynamic seismic analysis to demonstrate structural integrity under SSE loading conditions. The turbine stop, control, and bypass valves and the main steam lines from the turbine control valves to the turbine shall meet all of the quality group and quality assurance guidelines specified in SRP Section 3.2.2, Appendix A.

Further, that seismic analyses are performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after the SSE. In the SRM for SECY 93-087 the Commission approved the above described approach and positions for classifying the mainsteam lines and systems for boiling water reactors.

Potential Impact Number: 23066 **SRP Section Number:** 17.1

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY 93-087 the reliability assurance program (RAP) is discussed. The ALWR RAP would apply to those plant systems structures or components that are risk-significant (or significant contributors to plant safety) as quantified by the design certification PRA. SECY 93-087 contains detailed information on how a RAP is to be developed and maintained throughout the life of a plant. SECY 93- 087 also contains an interim staff position on RAP for design certification of the ALWRs. The discussion provided by SECY 93-087 on RAP was for information only.

Potential Impact Number: 23067 **SRP Section Number:** 17.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY 93-087 the reliability assurance program (RAP) is discussed. The ALWR RAP would apply to those plant systems structures or components that are risk-significant (or significant contributors to plant safety) as quantified by the design certification PRA. SECY 93-087 contains detailed information on how a RAP is to be developed and maintained throughout the life of a plant. SECY 93- 087 also contains an interim staff

position on RAP for design certification of the ALWRs. The discussion provided by SECY 93-087 on RAP was for information only.

Potential Impact Number: 23068 **SRP Section Number:** 17.3

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY 93-087 the reliability assurance program (RAP) is discussed. The ALWR RAP would apply to those plant systems structures or components that are risk-significant (or significant contributors to plant safety) as quantified by the design certification PRA. SECY 93-087 contains detailed information on how a RAP is to be developed and maintained throughout the life of a plant. SECY 93- 087 also contains an interim staff position on RAP for design certification of the ALWRs. The discussion provided by SECY 93-087 on RAP was for information only.

Potential Impact Number: 23070 **SRP Section Number:** 3.11

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

Final SER, NUREG-1462, Section 3.11, provides new guidance on the qualification of mechanical equipment and on mild environment radiation thresholds for electronic equipment. NUREG-1462 also discusses the NRC staff endorsement of Topical Report CENPD-255-A, Rev. 3 which provides a methodology to be used by CE applicants to qualify safety related electrical equipment located in a harsh environment.

Potential Impact Number: 23071 **SRP Section Number:** 3.11

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Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

Final SER, NUREG-1469, Section 3.11, provides new guidance on the qualification of mechanical equipment, on mild environment radiation thresholds for electronic equipment, and on equipment wetting qualification requirements. NUREG-1469 discusses the NRC staff endorsement of GE Topical Report NEDE-24326-1 (proprietary) and GESSAR II which outline the methodology used by GE to qualify nuclear steam system safety-related equipment that is subject to a harsh environment. NUREG-1469 also discusses a 60 year operating life.

Potential Impact Number: 23072 **SRP Section Number:** 3.11

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

SER, NUREG-1242, Section 4.8.2, discusses a new NRC staff position on source terms used to develop radiation environments associated with design basis accidents. NUREG-1242 also discusses a 60 year operating life.

Potential Impact Number: 23077 **SRP Section Number:** 9.2.6

Document: FINAL SER CE80 CH 8
Title: ELECTRIC POWER SYSTEMS

Summary:

Regarding core cooling for an SBO event, ABB-CE was required to confirm that:
(1) The plant will have sufficient condensate storage to remove decay heat for the duration of an SBO in accordance with RG 1.155, Section 3.3.2;
(2) The equipment and systems will be operable during an SBO event, and;
(3) By confirming that the AAC source provides appropriate cooling, the areas in which SBO equipment is located will be habitable.

In the DSER, the staff stated it would verify that these three aspects of the design are included as design commitments. This was designated as DSER Open Item 8.5-1.

In Amendment Q, ABB-CE supplemented the information on SBO as follows: ABB-CE indicated that the EFW system contains two, seismically designed, the EFW system contains two, seismically designed, 100-percent-capacity emergency feedwater storage tanks (EFWSTs). Each tank has a safety-related condensate storage volume of 350,000 gallons, which is sufficient condensate to maintain the unit in hot standby condition for 8 hours by using the EFW turbine-driven pumps.

Potential Impact Number: 23085 **SRP Section Number:** 2.3.1

Document: DRAFT SER CE80 CH 2
Title: SITE ENVELOPE CHARACTERISTICS

Summary:

Pursuant to 10 CFR Part 52.47(a)(1)(iii) an applicant seeking design certification is required to (1) provide site parameters postulated for the design, and (2) analyze and evaluate the design in terms of these parameters. In the CE80+ DSER, the staff listed the following parameters to be included in the site parameter envelope: Groundwater; Maximum Flood (or Tsunami) Level; Precipitation (Maximum Rainfall Rate for Roof Design); Precipitation (Snow Load); Design Temperatures (Ambient Air and Cooling Water); Extreme Wind; Tornado; Soil Properties (Minimum Soil-bearing Capacity); Seismology (Design Ground Response Spectra); Aircraft Hazards; and, Meteorology (Short-term Dilution Factor X/Q and Long-term Dilution Factor X/Q).

Potential Impact Number: 23086 **SRP Section Number:** 2.4.11

Document: DRAFT SER CE80 CH 2
Title: SITE ENVELOPE CHARACTERISTICS

Summary:

Pursuant to 10 CFR Part 52.47(a)(1)(iii) an applicant seeking design certification is required to (1) provide site parameters postulated for the design, and (2) analyze and evaluate the design in terms of these parameters. In the CE80+ DSER, the staff listed

the following parameters to be included in the site parameter envelope: Groundwater; Maximum Flood (or Tsunami) Level; Precipitation (Maximum Rainfall Rate for Roof Design); Precipitation (Snow Load); Design Temperatures (Ambient Air and Cooling Water); Extreme Wind; Tornado; Soil Properties (Minimum Soil-bearing Capacity); Seismology (Design Ground Response Spectra); Aircraft Hazards; and, Meteorology (Short-term Dilution Factor X/Q and Long-term Dilution Factor X/Q).

Potential Impact Number: 23087 **SRP Section Number:** 2.4.12

Document: DRAFT SER CE80 CH 2
Title: SITE ENVELOPE CHARACTERISTICS

Summary:

Pursuant to 10 CFR Part 52.47(a)(1)(iii) an applicant seeking design certification is required to (1) provide site parameters postulated for the design, and (2) analyze and evaluate the design in terms of these parameters. In the CE80+ DSER, the staff listed the following parameters to be included in the site parameter envelope: Groundwater; Maximum Flood (or Tsunami) Level; Precipitation (Maximum Rainfall Rate for Roof Design); Precipitation (Snow Load); Design Temperatures (Ambient Air and Cooling Water); Extreme Wind; Tornado; Soil Properties (Minimum Soil-bearing Capacity); Seismology (Design Ground Response Spectra); Aircraft Hazards; and, Meteorology (Short-term Dilution Factor X/Q and Long-term Dilution Factor X/Q).

Potential Impact Number: 23088 **SRP Section Number:** 2.4.12

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.79(b)

Summary:

An application referencing certified design must describe those portions of the design which are site-specific, such as the service water intake structure and the ultimate heat sink. An application referencing a certified design must also demonstrate compliance with the interface requirements established for the design.

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Potential Impact Number: 23090 **SRP Section Number:** 6.2.1.1.C

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In Item II.a of SRP Section 6.2.1.1.A, the staff states that the applicant should maintain a 10-percent margin during the construction permit stage of review and that at the operating license (OL) stage of review, the calculated peak pressure "should be approximately the same as at the construction permit stage." In Item II.2 of SRP Section 6.2.1.1.A, the staff cites GDC 50 as requiring "sufficient margin" for calculated peak containment pressure. The 1 percent margin between calculated peak primary containment pressure and its design value does not meet the requirements given in GDC 50 or SRP Section 6.2.1.1.A.

The applicant stated that the single-step licensing process for System 80+ precludes considering this review to be at the CP stage and that relaxing the 10 CFR Part 100 requirement for considering the Level B operating basis earthquake would provide additional margin in containment pressure. The System 80+ design is not final until construction and operation. Experience has shown that numerous factors such as design changes, as-built deviations, and analytical developments have affected other plants, prompting applicants to reanalyze the containment response to LOCAs and MSLBs and increase the calculated peak containment pressure. Furthermore, a "final design" should incorporate a sufficiently large margin in peak calculated containment pressure to accommodate such future changes that might increase the calculated peak containment pressure.

The applicant calculated a peak external differential of -12.6 kPa (-1.83 psid) on the primary containment caused by an inadvertent actuation of the containment spray system. This value provides a margin of 8.5 percent to the containment design value of -13.8 kPa (-2.0 psid). The applicant must justify the adequacy of this margin because it does not meet the 10-percent criterion previously discussed.

The margin in calculated peak internal primary containment pressure does not meet SRP Section 6.2.1.1.A and GDC 50. The applicant must justify the adequacy of this margin.

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Potential Impact Number: 23092 **SRP Section Number:** 6.2.7

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

The ABB-CE System 80+ primary containment is a welded steel vessel fabricated to the requirements of ASME Code, Section III. The ASME Code requires that the vessel materials meet the fracture toughness requirements of Subsection NE-2000. The staff concludes the design is in compliance with the requirements of GDC 51 because the steel vessel is made of materials that will meet the fracture toughness requirements of the ASME Code. This will ensure that the SCV materials will not undergo brittle fracture and the probability of a rapidly propagating fracture will be minimized.

The criteria used in the analysis, design, and construction of the steel containment structure to account for anticipated loadings and postulated conditions that may be imposed on the structure during its service lifetime conform with established criteria, codes, standards, and guides and the industry standard, ASME Code, Section III, Division 1, Subsection NE, 1989 Edition.

Potential Impact Number: 23094 **SRP Section Number:** 9.5.3

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

In the FSER, the staff describes the applicant's proposed lighting systems in detail. The FSER indicates that ABB-CE submitted information on the design of the normal, emergency, self-contained dc lighting units and security lighting to demonstrate that the lighting in the normal and vital plant areas as well as passageways to and from these areas is adequate. After installation, each lighting system will be inspected, checked, and tested to verify that it is operable and provides proper coverage.

In the DSER, it was indicated that the applicant should address the following staff concerns and EPRI guidelines:

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(1) The staff will verify that the completion of the hazard analyses and plant emergency procedures are included as commitments and that appropriate inspections, tests, and/or analyses are included as part of ITAAC to verify implementation of their design commitments.

(2) State what method will be used to distinguish between the normal, emergency, and security lighting cables and circuits to ensure they are physically identified and separate.

(3) Confirm that the Class 1E distribution system supplies at least one of the circuits supplying lighting fixtures for the normal lighting system in safety-related areas (other than main control room) and in access routes to these areas. The other lighting circuit can be supplied from a non-Class 1E electrical division backed-up by the combustion turbine generator.

(4) Integrate the emergency lighting system in the main control room with the normal lighting system and design it so that alternate emergency lighting fixtures are fed from separate safety divisions.

(5) Design the emergency lighting installations that serve the main control room and those other areas of the plant where safe shutdown operation may be performed so that these installations will continue to function during and after a design basis earthquake.

(6) Confirm which part of the emergency lighting system will not be qualified as Class 1E.

In the FSER, the staff discusses CESSAR-DC amendments addressing the above items and concludes that the amendments are acceptable.

The staff accepted the applicant's design and resolution of item 3 above and provided the following discussion: The normal lighting system is energized as long as power from an offsite power source or a standby non-safety source (Combustion Turbine) is available. The Combustion Turbine is designed to start automatically within two minutes from the onset of a LOOP event. ABB-CE stated that all lighting fixtures and other components of the lighting system located in normally occupied areas or in areas containing safety equipment are supported so as to enhance the earthquake survivability of these components and to ensure that they do not present a personnel or equipment hazard when subjected to a seismic loading of a DBE. This addresses Item (3) and is acceptable because it ensures that the normal lighting system is energized

as long as power from an offsite power source or a standby non-safety source (CTG) is available under normal and abnormal plant conditions.

Analyst Note:

The staff stated positions in the EPRI Evolutionary Plant FSER that portions of the continuous ac lighting in safety-related areas outside the main control room and in the access routes to those areas should be provided from a Class 1E distribution system with identified portions classified/qualified Class 1E and seismic Category I and lighting fixtures seismically supported as a minimum. The EPRI requirements reviewed by the staff and the positions stated therein relate to EPRI's proposed alternate ac source-backed (by a combustion turbine-generator) non-safety "normal" ac lighting system and "emergency" lighting system consisting of only self-contained battery-operated units outside the control room and Class 1E UPS powered lighting in the control room. For the CE System 80+ design, since the applicant proposed an "emergency" lighting system that consists of both Class 1E ac lighting for "vital" areas (as determined by hazard analyses) and Class 1E dc self contained, battery operated lighting units, the staff accepted the "normal" system as a non-safety system with no portion powered from Class 1E sources. The apparent conflict between the staff position that portions of the "normal" system should be Class 1E and the staff's acceptance of ABB-CE's "normal" system as a non-safety system thus does not represent a conflict or change in position.

Potential Impact Number: 23095 **SRP Section Number:** 9.5.3

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

In response to RAI Q500.20, the applicant proposed to delete the reference to a UPS and change Section 9.5.3.2.2 to read instead: "The security lighting system is considered part of the permanent non-safety systems and is fed from the alternate ac (AAC) source (combustion turbine), which is located in a secure vital area for protection." However, the AAC could take 10 minutes to start (from the onset of a LOOP event), and additional time for load sequencing before security lighting would be restored. The staff concluded in the DSER that the proposed change is inconsistent with CESSAR Section 9.5.3.1 and does not conform with the EPRI URD requirement for uninterruptible power for those portions of the security lighting that are essential to

plant protection following interruption of normal power. Inconsistency between CESSAR Section 9.5.3.1 and the changes proposed for CESSAR Section 9.5.3.2.2, and the applicant's reason for deviating from the URD requirements for uninterruptible power for those portions of the security lighting that are essential to plant protection, were designated as DSER Open Item 9.5.3-2.

In the FSER, the staff indicates that in a CESSAR-DC amendment, ABB-CE stated in CESSAR-DC Section 8.3.1.1.5.1, that the AAC is designed to start automatically within two minutes from the onset of an LOOP event. As stated in CESSAR-DC Section 9.5.3.2.2, the security lighting is part of the permanent non-safety systems which is fed from the AAC. Selected portions of the security lighting system essential to maintaining adequate plant protection are powered from an uninterruptible power supply. Requiring uninterruptible power for those portions of the security lighting that are essential to plant protection is consistent with the URD requirements. This additional information has also removed the inconsistency between CESSAR-DC Section 9.5.3.1 and the changes proposed for CESSAR-DC Section 9.5.3.2.2. On this basis, DSER Open Item 9.5.3-2 is resolved.

Potential Impact Number: 23096 **SRP Section Number:** 9.5.3

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

In the DSER, the staff indicated that in CESSAR Section 9.5.3.2.2, the applicant states that the security lighting system will provide a minimum illumination of 0.2 foot-candle at ground level. In CESSAR Table 9.5.3-1, the applicant lists 2 to 5 foot-candles as typical illuminance ranges for normal exterior area lighting. Those illumination levels would give a range of ratios of typical to minimum illumination of 10:1 to 25:1. However, in NUREG/CR-1327, "Security Lighting Planning Document for Nuclear Fixed Site Facilities," the NRC contractor recommends that for security lighting of the protected area's isolation zone, the average to minimum illumination ratio should not exceed 4:1, with less than 3:1 preferred. Responding to RAI Q500.27, the applicant proposed to add the following to Section 9.5.3.2.2: "The Owner-Operator shall provide a security lighting system that will meet closed-circuit television (CCTV) illumination requirements within camera viewing areas to permit prompt assessment of intrusion alarms." This change was designated as DSER Confirmatory Item 9.5.3-1.

In the FSER the staff indicates that by CESSAR-DC amendment, ABB-CE incorporated the above statement to CESSAR-DC Section 9.2.3.2.2, resolving the staff's concern about adequate illumination for CCTV and resolving DSER Confirmatory Item 9.5.3-1.

Potential Impact Number: 23098 **SRP Section Number:** 9.5.3

Document: FINAL SER EPRI CH 9
Title: EPRI URD-EPD: SITE SUPPORT SYSTEMS

Summary:

In the DSER for Chapter 9, the staff stated that Section 5.2.12 specified that the security system will be powered by a noninterruptible electric power source, with the exception, because of the size of the load, of protected area boundary lighting. The physical security plan was to include the compensatory measures to be taken if the boundary lighting was lost.

The staff also identified backup power for security lighting as a confirmatory issue because EPRI had committed to revise Section 5.2.12 of Chapter 9, to resolve an inconsistency between this section and Chapter 11, by requiring that the security equipment onsite secondary power supply be located in a vital area.

EPRI has revised the design requirements to provide for a separate, dedicated, uninterruptible backup power source located in a vital area and to include a security diesel generator, uninterruptible power supply cabinets, and batteries. The staff concludes that these revisions satisfactorily address its concerns. Therefore, this DSER confirmatory issue is closed. In a letter dated January 24, 1992, EPRI proposed a change to Section 5.2.12.3 that would allow only selected parts of security lighting to be backed up with interruptible power. Security lighting requirements are also included in Chapter 11 of the Evolutionary Requirements Document. The staff concludes that this change is compatible with existing regulatory requirements and is acceptable.

Potential Impact Number: 23099 **SRP Section Number:** 9.5.3

Document: FINAL SER EPRI CH 11
Title: EPRI URD-EPD: ELECTRIC POWER SYSTEMS

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Summary:

The necessity to enter high radiation areas on a frequent basis to relamp burned out lighting fixtures can be a source of unnecessary personnel radiation exposures. In an effort to reduce the doses associated with relamping and maintenance of lighting fixtures, the plant designer will locate lighting fixtures in low radiation zones, use long-life lighting, and provide easy access to lighting fixtures. These features meet the intent of Regulatory Guide 8.8 to maintain occupational doses as low as is reasonably achievable and are acceptable.

Potential Impact Number: 23100 **SRP Section Number:** 9.5.3

Document: FINAL SER EPRI CH 11
Title: EPRI URD-EPD: ELECTRIC POWER SYSTEMS

Summary:

The staff concludes that at least a portion of the continuous ac lighting in safety-related areas outside the main control room and in the access routes to those areas should be provided from a Class 1E distribution system capable of being powered from a Class 1E diesel generator. The lighting transformers and their associated lighting panels should be Class 1E and seismic Category I. The lighting circuits up to the lighting fixtures should be treated as Class 1E and routed in seismic Category I raceways. The lighting fixtures themselves should, as a minimum, be seismically supported (if the fixtures can be seismically qualified, they should be so qualified). The circuits to individual lighting fixtures should be staggered with the staggered circuits fed from separate electrical divisions. One of the circuits feeding the lighting fixtures should be supplied from the Class 1E distribution system. The other circuit to the fixtures should be supplied from a separate non-Class 1E electrical division backed up by the combustion turbine generator. This system should be provided in addition to the dc self-contained, battery-operated lighting units.

The staff will pursue these details of the lighting system design with the ALWR plant designers during its review of an individual application for FDA/DC in accordance with the criteria in SRP Section 9.5.3.

Potential Impact Number: 23102 **SRP Section Number:** 9.5.3

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Document: FINAL SER EPRI CH 11
Title: EPRI URD-EPD: ELECTRIC POWER SYSTEMS

Summary:

The staff concluded that emergency lighting system provide illumination units such that the lighting provided will achieve a minimum illumination of 10 foot-candles at the printed or written material and on scales and legends. EPRI revised Section 8.5.1 in Chapter 11 to require that the emergency lighting system provide a minimum illumination level of 10 foot-candles at all workstations in the plant where emergency operations will be performed that could require the reading of printed or written material or the reading of scales and legends. With regard to the lighting intensity, the new revision is acceptable since it is now clear that the requirement is specifying an illumination level for the lighted area rather than the source strength for the emergency lighting units. With regard to the areas to which the requirement applies, the revision is acceptable with the understanding that the requirement is not limited to control rooms or local control stations but applies, in general, to the areas of the plant where emergency operations will be performed that require the reading of printed or written material. EPRI stated that this was the case. The staff, therefore, concludes that this open issue is closed.

Potential Impact Number: 23103 **SRP Section Number:** 9.5.3

Document: FINAL SER EPRI CH 11
Title: EPRI URD-EPD: ELECTRIC POWER SYSTEMS

Summary:

EPRI requires that the emergency lighting system of the main control room be integrated with the normal lighting system and be designed so that alternate emergency lighting fixtures are fed from separate safety divisions. In the rationale portion of this section, EPRI states that the emergency lighting circuits should be staggered, with the staggered circuits fed from separate safety divisions (powered from redundant, independent safety uninterruptible power supplies) so as to ensure that adequate lighting is retained in the main control room in the event of a circuit failure. The staff concludes that EPRI requirements require two separate and independent safety power supplies fed from redundant safety divisions to power the emergency lighting in the main control room. This was considered necessary to ensure that the emergency lighting meets the single-failure criterion. This is acceptable. The staff is not entirely

clear, however, how the emergency lighting system in the main control room will be integrated with the normal lighting system as required by EPRI. If some fixtures in the main control room are fed from, and part of, the normal lighting system while other fixtures are separately fed from, and part of, the emergency lighting system, with the combination providing the normal control room illumination levels as the staff believes is the intent of the requirement, this would likely be an acceptable design. If, however, the integration is done by normally powering all the fixtures in the control room from the normal lighting system and then switching a portion to the emergency lighting supply on loss of the normal lighting supply, serious separation issues could be involved. The staff will review this aspect of the lighting system in its review of an individual application for FDA/DC.

With regard to the qualification of emergency lighting, EPRI requires that the emergency lighting installations that will serve the main control room and those other areas of the plant where safe shutdown operations may be performed be designed to remain functional during and after a design-basis earthquake. This requirement, in conjunction with the staff's requirement to supply at least a portion of the safety-related continuous ac lighting outside the main control room from a Class 1E distribution system (Section 8.2.1 of this chapter), is acceptable.

Potential Impact Number: 23105 **SRP Section Number:** 9.5.3

Document: FINAL SER EPRI CH 11
Title: EPRI URD-EPD: ELECTRIC POWER SYSTEMS

Summary:

In its letter dated June 8, 1989, the staff stated that it was unclear whether the 8-hour battery-powered emergency lights would be installed in high-radiation areas or outdoor locations. The staff's concern resulted from difficulties that existing plants have had with such installations. In its letter dated October 19, 1989, EPRI responded that the 8-hour battery-powered emergency lights will be installed throughout ALWRs referencing the Evolutionary Requirements Document, as necessary, in accordance with the guidance of SRP Section 9.5.1, "Fire Protection Program," in all areas needed for the operation of safe-shutdown equipment and in the access routes leading to these areas. In addition, similar lighting units with at least a 1.5-hour battery power supply will be provided throughout the plant to ensure personnel safety and property protection in accordance with the requirements of the Life Safety Code and the National Electric Code. EPRI also stated that although specific high-radiation areas or outdoor locations

that may require battery-powered emergency lights are not addressed in the Evolutionary Requirements Document, it expects that some units may be required in outdoor locations, but few, if any, in high-radiations areas.

EPRI requires that additional dc self-contained battery-operated lighting units be installed throughout the plant to provide emergency lighting for personnel safety in accordance with the National Electric Code and the Life Safety Code of the National Fire Protection Association. Additionally, EPRI has committed to comply with SRP Section 9.5.1 (see Table B.1-2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document). This is acceptable.

Potential Impact Number: 23106 **SRP Section Number:** 9.5.3

Document: FINAL SER EPRI CH 11
Title: EPRI URD-EPD: ELECTRIC POWER SYSTEMS

Summary:

EPRI requires that the non-safety batteries provide at least 30 minutes of continued operation of the security lighting system in the event of an interruption of ac power. Because this exceeds current regulatory requirements and provides sufficient time to load security lighting onto the dedicated backup power supply, the staff concludes that this is acceptable.

EPRI also requires that the security lighting system be designed to provide a minimum illumination level of a 1 foot-candle when measured horizontally at ground level and states that higher illumination levels may be necessary in certain areas, such as portals and vital areas, to provide adequate monitoring capability. Because the requirement for a minimum illumination level of 1 foot-candle is in excess of the 0.2 foot-candle required by 10 CFR Part 73, this requirement is acceptable.

Experience has shown that inattention to the integration of exterior lighting systems with the isolation zone's closed-circuit television (CCTV) system, particularly uniformity of lighting (e.g., light sources or reflections in the field of view, glare, blooming, and excessive light/dark ratios) can be detrimental to alarm assessment. In its letter dated September 15, 1989, EPRI stated that light/dark ratio was an engineering detail beyond the scope of the Evolutionary Requirements Document. The staff concludes that this response is acceptable. However, it will determine if the integration of the exterior

lighting systems with the CCTV system is acceptable when it evaluates a site's security system during the review of an individual application for FDA/DC.

The staff concludes that the requirements in Section 8 of Chapter 11 are consistent with the requirements of 10 CFR 73.55(c)(5) for security lighting.

Potential Impact Number: 23107 **SRP Section Number:** 9.5.3

Document: FINAL SER ABWR CH 8
Title: ELECTRIC POWER SYSTEMS

Summary:

GE indicated that there will be four lighting systems: the normal ac lighting system, the standby ac lighting system, the emergency dc lighting system, and the guide lamp lighting system. The SER provides a description of the four lighting systems, based upon the staff's review of GE's submittal(s).

GE provided information to demonstrate that the lighting is adequate in plant areas containing safety-related equipment as well as passageways to and from these areas. GE indicated that the ABWR design for each lighting system will meet or exceed the lighting level requirements of the Illuminating Engineering Society Lighting Handbook and will have the capability of being functionally tested on a periodic basis.

The staff concluded that a lighting system that meets the above described commitments will provide adequate levels of light to permit the required operation and maintenance of equipment in safety-related equipment areas, and passageways to and from these areas, under normal operating conditions. This design is, therefore, acceptable.

GE indicated that the ac standby and dc emergency lighting systems which are normally energized will be periodically inspected and bulb replacement will be performed. In addition, the guide lamp system will be inspected and tested periodically to ensure operability of lights and switching circuits. The COL applicant must include these periodic tests, inspections, and bulb replacement in appropriate plant procedures.

For off-normal conditions, it was not clear that the design adequately considers lighting needs for areas containing safety-related equipment and for passageways to and from these areas where plant operations are or may be required by emergency procedures.

Therefore, GE needed to further address the adequacy of the lighting in areas containing safety-related equipment under off-normal conditions. This was DFSER Open Item 8.3.5-1.

GE indicated that because of the redundancy provided by the four lighting systems described in the SSAR, the complete loss of lighting in any of the critical areas is not credible. The standby lighting and emergency lighting systems provide totally independent low level illumination in areas vital to safe shutdown of the reactor. In addition, the safety-related control systems will automatically bring the plant to safe-shutdown conditions. The control systems do not require the lighting system to perform their safety function of bringing the plant to a safe-shutdown condition.

In the FSER, the staff concludes that a lighting system design which meets the above described commitments will provide reasonable assurance that sufficient illumination will be available to perform emergency procedures as may be required during off normal conditions. The lighting system design is, therefore, acceptable. GE included the above commitments in Section 9.5.3.2.5 of the SSAR, which is acceptable. Therefore, this item is resolved.

Potential Impact Number: 23108 **SRP Section Number:** 12.2

Document: C&S: ANS 18.1
Title: RADIOACTIVE SOURCE TERM FOR NORMAL OPERATION OF LIGHT WATER REACTORS

Summary:

ANS 18.1 (the most recent version is dated 1984) superseded ANSI N237-1976, which is cited in the Acceptance Criteria subsection as it relates to the establishment of typical long-term concentrations of principal radionuclides in fluid streams of light-water cooled nuclear power plants.

Potential Impact Number: 23109 **SRP Section Number:** 12.2

Document: ADVANCE SER ABWR CH 12
Title: RADIATION PROTECTION

APPENDIX II
Potential Impacts

Summary:

The estimates of fission and activation products concentrations in the ABWR systems containing reactor water are based on American National Standards Institute/American Nuclear Society (ANSI/ANS)-18.1 "Radioactive Source Term for Normal Operation of LWR's" (1984), adjusted using the assumptions in RG 1.112 (Rev. 0), "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors." Allowances are included for the buildup of activation products resulting from corrosion and wear on the basis of operating experience of reactors of similar design. Neutron and prompt gamma source terms are based on reactor core physics calculations. The source terms used to determine in-plant-post- accident radiation levels are based on the assumptions in NUREG-0737, Item II.B.2.

Potential Impact Number: 23113 **SRP Section Number:** 2.3.4

Document: ADVANCE SER ABWR CH 2
Title: SITE CHARACTERISTICS

Summary:

The bounding atmospheric relative concentration values for LPZ (the 8-hour time period from 0 to 8 hours, the 16-hour period from 8 to 24 hours, the 3-day period from 1 to 4 days, and the 26-day period from 4 to 30 days) were determined by the most limiting design basis accident loss-of- coolant accident (LOCA) not to exceed the dose reference values given in 10 CFR Part 100 (300 rem for the thyroid and 25 rem for the whole-body). As shown in Table 2.0-1 of the SSAR, the two-hour LPZ atmospheric relative concentration and an annual average (8760 hours) concentration values were obtained by logarithmic extrapolation of these calculated LPZ bounding atmospheric relative concentrations.

The staff concludes in Section 15.4 of this report that GE's proposed bounding atmospheric relative concentrations (Chi/Q) for the EAB and for the LPZ of the ABWR plant, in conjunction with the engineered safety features systems provided in the ABWR design, are sufficient to provide reasonable assurance that the radiological consequences of a postulated LOCA will be within the dose reference values set forth in 10 CFR Part 100.

APPENDIX II
Potential Impacts

Potential Impact Number: 23115 **SRP Section Number:** 2.3.4

Document: DRAFT SER CE80 CH 2
Title: SITE ENVELOPE CHARACTERISTICS

Summary:

Pursuant to 10 CFR Part 52.47 (a)(1)(iii) an applicant seeking design certification is required to (1) provide site parameters postulated for the design, and (2) analyze and evaluate the design in terms of these parameters. The applicant has provided a list of site parameters in CESSAR Table 2.0-1...

After reviewing Table 2.0-1 the staff has concluded that the following site parameters should be included in Table 2.0-1...

Short-term dilution factor (X/Q): 4.97x10-4; EAB=500 meters...

Potential Impact Number: 23116 **SRP Section Number:** 2.3.2

Document: DRAFT SER CE80 CH 2
Title: SITE ENVELOPE CHARACTERISTICS

Summary:

Pursuant to 10 CFR Part 52.47(a)(1)(iii) an applicant seeking design certification is required to (1) provide site parameters postulated for the design, and (2) analyze and evaluate the design in terms of these parameters. In the CE80+ DSER, the staff listed the following parameters to be included in the site parameter envelope: Groundwater; Maximum Flood (or Tsunami) Level; Precipitation (Maximum Rainfall Rate for Roof Design); Precipitation (Snow Load); Design Temperatures (Ambient Air and Cooling Water); Extreme Wind; Tornado; Soil Properties (Minimum Soil-bearing Capacity); Seismology (Design Ground Response Spectra); Aircraft Hazards; and, Meteorology (Short-term Dilution Factor X/Q and Long-term Dilution Factor X/Q).

Potential Impact Number: 23118 **SRP Section Number:** 15.0

Document: FINAL SER CE80 CH 15

APPENDIX II
Potential Impacts

Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

GDC 17 in 10 CFR Part 50, Appendix A requires, in part: An onsite electric power system 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 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.

Potential Impact Number: 23123 **SRP Section Number:** 3.9.2

Document: NRC MEMORANDUM 09/30/93, from E. Beckjord to F. Gillespie
Title: GENERIC ISSUE NO. 146, 'SUPPORT FLEXIBILITY OF EQUIPMENT AND COMPONENTS'

Summary:

RES recommended revising pertinent sections of the SRP to assure adequate treatment of equipment and component anchorage stiffness assumptions.

Potential Impact Number: 23124 **SRP Section Number:** 3.9.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

APPENDIX II
Potential Impacts

Summary:

Generic Safety Issue No. 146, "Support Flexibility of Equipment and Components," addresses the issue that seismic loading on equipment and pipe mounted components may have been under estimated.

Potential Impact Number: 23131 **SRP Section Number:** 6.5.2

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016, the staff recommended that the Commission approve an interim approach for evolutionary ALWRs for the methods used in the calculation of the source term. In the SRM for SECY 90-016 the Commission approved this approach to the source term with a qualification on incorporating the results of research into regulations. In SECY 93-087, the staff provided an overview of the current source term status and provided information on the development of a new source term to replace the source term of TID-14844.

Potential Impact Number: 23146 **SRP Section Number:** 6.5.2

Document: C&S: ANS 56.5-79
Title: PWR and BWR Containment Spray System Design Criteria

Summary:

SRP Section 6.5.2 cites ANSI/ANS 56.5-1979 for PWR and BWR containment Spray System Design Criteria. This standard was reaffirmed in 1987.

Potential Impact Number: 23148 **SRP Section Number:** 6.3

Document: NRC GENERIC LETTER 88-17

APPENDIX II
Potential Impacts

Title: LOSS OF DECAY HEAT REMOVAL

Summary:

This generic letter contains guidance and recommendations related to operation following placement of the NSSS on shutdown cooling, or following the attainment of conditions under which shutdown cooling would normally be initiated. SRP sections which address areas important to safety during shutdown and low power operations such as: operations, training, fire protection, and instrumentation should be evaluated against the guidance contained in this generic letter.

Potential Impact Number: 23149 **SRP Section Number:** 9.5.1

Document: NRC GENERIC LETTER 88-17
Title: LOSS OF DECAY HEAT REMOVAL

Summary:

This generic letter contains guidance and recommendations related to operation following placement of the NSSS on shutdown cooling, or following the attainment of conditions under which shutdown cooling would normally be initiated. SRP sections which address areas important to safety during shutdown and low power operations such as: operations, training, fire protection, and instrumentation should be evaluated against the guidance contained in this generic letter.

Potential Impact Number: 23152 **SRP Section Number:** 9.5.1

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

To demonstrate adequate treatment of shutdown risk for ALWRs, the staff will require: 1) adequate vendor assessment of shutdown and low-power risk, identifying design-specific vulnerabilities and weaknesses . 2) documentation showing consideration and incorporation of design features that minimize shutdown and low-power risk vulnerabilities. The shutdown and Low-power risk issue is currently

under review by the staff, as discussed in NUREG-1449. The staff plans to assess current regulatory requirements and, where necessary, will develop and implement appropriate regulatory actions to address the issues, including new guidance and new requirements.

Potential Impact Number: 23158 **SRP Section Number:** 9.5.1

Document: DRAFT SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The NRC has evaluated risks during shutdown and low power operation. This task had the objective of developing a thorough understanding of the manner in which activities and operations during shutdown are planned and implemented; determining the root causes of earlier events was also a part of this task. The staff evaluation of shutdown and low-power operations was issued in NUREG-1449, "NRC Staff Evaluation of Shutdown and Low Power Operation," September 1993. Because an event that occurs during shutdown and at low power has safety significance, the staff has determined that concerns with the shutdown operation should be satisfactorily addressed before the NRC issues a final design approval for the System 80+ design. ABB-CE was required to demonstrate adequate treatment of shutdown risk by submitting information, including (1) documentation showing adequate assessment of shutdown and low-power risk, with identification of design-specific vulnerabilities and weaknesses and (2) documentation showing consideration and incorporation of design features that minimize shutdown and low-power risk vulnerabilities. The staff has reviewed the shutdown risk report submitted by ABB-CE and found it acceptable. In addition, the staff asked the applicant to provide a description for any deviations and a justification of the deviations from GL 88-17 recommendations.

Potential Impact Number: 23164 **SRP Section Number:** 9.5.1

Document: ADVANCE SER ABWR CH 19
Title: SEVERE ACCIDENTS

Summary:

APPENDIX II
Potential Impacts

Various incidents occurring at nuclear power plants during low power and shutdown operation modes over the past several years have raised NRC staff concerns regarding plant vulnerability during these operating modes. The shutdown events have caused plants to lose their ability to maintain core cooling, provide make-up coolant to the reactor, and to maintain electrical power to essential equipment. These events, as well as others, prompted the staff to begin a comprehensive review of low-power and shutdown operations including hot shutdown, cold shutdown, and refueling at all nuclear plants. The staff issued NUREG-1449, Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States, to document technical findings associated with shutdown conditions. The staff is using these technical findings to prepare appropriate regulatory actions to address shutdown risk. The fundamental conclusion of NUREG-1449 is that public health and safety have been adequately protected while plants were in shutdown conditions. However, numerous and significant events have indicated that substantial safety improvements should be made in several areas including fire protection. The staff utilized the technical findings and insights from NUREG-1449 in its safety evaluation of the ABWR.

Potential Impact Number: 23170 **SRP Section Number:** 6.5.1

Document: C&S: ANSI N509-80
Title: Nuclear Power Plant Air Cleaning Units and Components

Summary:

SRP section 6.5.1 Acceptance Criteria specifies ANSI 509-1980. This code has been revised and is now designated as ASME N509-1989 "Nuclear Power Plant Air-Cleaning Units and components."

Potential Impact Number: 23171 **SRP Section Number:** 6.5.1

Document: C&S: ANSI N510-80
Title: Testing of Nuclear Air Treatment Systems

Summary:

APPENDIX II
Potential Impacts

SRP section 6.5.1 Acceptance Criteria specifies ANSI 510-1980. This code has been revised and is now designated as ASME N510-1989 "Nuclear Power Plant Air-Cleaning Units and Components."

Potential Impact Number: 23173 **SRP Section Number:** 6.5.1

Document: NRC GENERIC LETTER 80-11 Revision Date: 1/31/80
Title: IE BULLETIN NO. 80-03, LOSS OF CHARCOAL FROM STANDARD TYPE II, 2 INCH, TRAY ADSORBER CELLS

Summary:

This generic letter transmitted NRC Bulletin 80-03 to licensees for action.

Potential Impact Number: 23174 **SRP Section Number:** 6.5.1

Document: NRC BULLETIN 80-03
Title: LOSS OF CHARCOAL FROM STANDARD TYPE II, 2 INCH, TRAY ADSORBER CELLS

Summary:

The purpose of this NRC Bulletin was to request licensees take actions to determine if charcoal absorber cells in use, or proposed for use, have the potential for the loss of charcoal incidental to handling, storage or use.

Potential Impact Number: 23175 **SRP Section Number:** 6.5.1

Document: ADVANCE SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

APPENDIX II
Potential Impacts

The staff requested that the SSAR Section 6.5.1.3.3 be revised to address IE Bulletin 80-03 to state that the charcoal tray and screen will be all welded construction to preclude the potential loss of charcoal from adsorber cells per IE Bulletin 80-03.

Potential Impact Number: 23177 **SRP Section Number:** 6.3

Document: NRC GENERIC LETTER 83-36
Title: NUREG-0737 TECHNICAL SPECIFICATIONS

Summary:

This Generic Letter provided additional guidance on the implementation of TMI Task Action Plan Items. Included was a discussion on item II.K.3.17

Potential Impact Number: 23178 **SRP Section Number:** 6.1.2

Document: DRAFT SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In a letter of April 30, 1992 (letter number LD-92-063), the applicant states that the coatings will be selected based on ASTM D3842-86, "Standard Guide for Selection of Test Methods for Coatings for Use in Light Water Nuclear Power Plants," ASTM D3911-89, "Standard Test Method for Evaluating Coatings Used in Light Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions," and ASTM D3843-89, "Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities." The staff has reviewed ASTM D3842-86 and ASTM D3911-89 and found them acceptable.

Potential Impact Number: 23181 **SRP Section Number:** 9.5.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

APPENDIX II
Potential Impacts

Location: Part 50.63(c)(2)

Summary:

10 CFR 50.63 (c) (2) includes requirements applicable to an alternate AC source.

Potential Impact Number: 23182 **SRP Section Number:** 9.5.5

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

Regulatory Position C.3 of Reg. Guide 1.155 includes staff guidance applicable to an alternate AC source.

Potential Impact Number: 23183 **SRP Section Number:** 9.5.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Part 50.63 (c) (2)

Summary:

10 CFR 50.63 (c) (2) includes requirements applicable to an alternate AC source.

Potential Impact Number: 23184 **SRP Section Number:** 9.5.6

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

APPENDIX II
Potential Impacts

Regulatory Position C.3 of Reg. Guide 1.155 includes staff guidance applicable to an alternate AC source.

Potential Impact Number: 23188 **SRP Section Number:** 6.1.2

Document: C&S: ASTM D3842
Title: STANDARD GUIDE FOR SELECTION OF TEST METHODS FOR COATINGS FOR USE IN LIGHT WATER NUCLEAR POWER PLANTS

Summary:

Evolutionary plant SERs cite ASTM D3842-86, "Standard Guide for Selection of Test Methods for Coatings for Use in Light Water Nuclear Power Plants," as acceptable for use in specifications for protective coating systems.

Potential Impact Number: 23189 **SRP Section Number:** 6.1.2

Document: C&S: ASTM D3911
Title: STANDARD TEST METHOD FOR EVALUATING COATINGS USED IN LIGHT WATER NUCLEAR POWER PLANTS AT SIMULATED DESIGN BASIS ACCIDENT (DBA) CONDITIONS

Summary:

Evolutionary plant SERs cite ASTM D3911-89, "Standard Test Method for Evaluating Coatings Used in Light Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions," as acceptable for use in specifications for protective coating systems.

Potential Impact Number: 23195 **SRP Section Number:** 10.3

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

APPENDIX II
Potential Impacts

Summary:

Seismic Qualification: Appendix A to 10 CFR Part 100 requires that structures, systems, and components necessary to ensure the capability to mitigate the radiological consequences of accidents that could result in exposures comparable to the dose guideline exposures of Part 100 be designed to remain functional during and after a safe shutdown earthquake (SSE). Thus, the main steamline, portions of its associated piping, and the main condenser are required to remain functional if the SSE occurs. As such, these components are required to be analyzed to demonstrate their capability to maintain their structural integrity. Appendix A to 10 CFR Part 100 requires that the engineering method used to ensure that the safety functions are maintained during and after an SSE involve the use of either a suitable dynamic analysis or a suitable qualification test.

The staff's position is that the main steam piping from the outermost isolation valve up to the seismic interface restraint and branch lines up to the first closed valve must conform to Appendix A to SRP Section 3.2.2 and RG 1.29. The main steamline from the seismic interface restraint up to, but not including, the turbine stop valve (including branch lines to the first normally closed valve) must be classified as Quality Group B and inspected in accordance with applicable portions of ASME Code, Section XI, Inservice Inspection, but may be classified as non-seismic Category I if it has been analyzed using a dynamic seismic analysis to demonstrate its structural integrity under SSE loading conditions. However, all pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 are applicable to ensure that the quality of the piping material is commensurate with its importance to safety during both operational and accident conditions.

The seismic interface restraint must provide a structural barrier between the seismic Category I portion of the main steamline in the reactor building and the non-seismic Category I portions of the main steamline in the turbine building. The seismic interface restraint will be located inside the seismic Category I building.

Potential Impact Number: 23197 **SRP Section Number:** 9.5.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.63(a)(2)

Summary:

APPENDIX II
Potential Impacts

10 CFR 50.63(a)(2) provides requirements for ensuring adequate core cooling in the event of a station blackout for the specified duration.

Potential Impact Number: 23198 **SRP Section Number:** 9.5.4

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

Regulatory Position C.3 includes staff guidance applicable to ensuring the ability to cope with a station blackout for the specified duration.

Potential Impact Number: 23199 **SRP Section Number:** 9.5.8

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.63(a)(2)

Summary:

10 CFR 50.63(a)(2) provides requirements for ensuring adequate core cooling in the event of a station blackout for the specified duration.

Potential Impact Number: 23200 **SRP Section Number:** 9.5.8

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

Regulatory Position C.3 includes staff guidance applicable to ensuring the ability to cope with a station blackout for the specified duration.

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Potential Impacts

Potential Impact Number: 23201 **SRP Section Number:** 9.5.7

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.63(a)(2)

Summary:

10 CFR 50.63(a)(2) provides requirements for ensuring adequate core cooling in the event of a station blackout for the specified duration.

Potential Impact Number: 23202 **SRP Section Number:** 9.5.7

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

Regulatory Position C.3 includes staff guidance applicable to ensuring the ability to cope with a station blackout for the specified duration.

Potential Impact Number: 23204 **SRP Section Number:** 9.3.4

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

Regulatory Position C.3 includes staff guidance applicable to ensuring the ability to cope with a station blackout for the specified duration.

Potential Impact Number: 23209 **SRP Section Number:** 15.2.8

APPENDIX II
Potential Impacts

Document: DRAFT SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

The Semiscale test data for feedwater line breaks (as discussed in Section 4.3.3.1 of NUREG/CR-4945, dated July 1987) indicated that the SG heat transfer capacity remains unchanged until the SG liquid inventory is nearly depleted. This is followed by a rapid reduction to 0 percent heat transfer with little further reduction in the SG water inventory. In light of these test data, the applicant is required to discuss the FLB method and justify that the method is conservative in comparison with the Semiscale test data. If the method is found to be nonconservative, the applicant is required to reanalyze the FLB event by using the model that is supported by the test data, including the Semiscale data. The applicant should submit the results for the staff to review. This is an open item. See Open Item 15.3.2-1.

The location mentioned in the DSER may be in error. Subsection 4.3.3.1 addresses Steam Line Breaks. Subsection 4.3.3.2 on page 71 states that "The Combustion Engineering (CE) FSAR Appendix 15B assumes for a feed line break that there is 100% heat transfer until the liquid inventory is depleted, followed by a step change reduction in the heat transfer to 0%, which is nonconservative based on Semiscale scaled results. ... CE FSAR heat transfer assumptions in the secondary may be nonconservative because Semiscale shows degradation beginning at about 10% inventory." This is probably the relevant text.

Potential Impact Number: 23210 **SRP Section Number:** 3.10

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

The staff recommended various positions related to the elimination of the operating basis earthquake (OBE) from the design of systems, structures, and components.

APPENDIX II
Potential Impacts

Potential Impact Number: 23211 **SRP Section Number:** 3.10

Document: NRC MEMORANDUM 05/05/92, from Robert L. Baer to James A. Norberg Revision Date: 5/5/92
Title: RECOMMENDATIONS FOR SRP REVISIONS RELATED TO SNUBBERS

Summary:

RES proposed SRP changes as part of the resolution of Generic Issue 113. Generic Issue 113 addresses the dynamic qualification and testing of large bore hydraulic snubbers.

Potential Impact Number: 23214 **SRP Section Number:** 3.10

Document: C&S: IEEE 382
Title: Standard for Qualification of Actuators for Power Operated Valve Assemblies With Safety-Related Functions for Nuclear Power Plants

Summary:

IEEE Std. 382-1985 establishes minimum requirements for, and provides guidance regarding, the methods and procedures for qualification of power-driven valve actuators with safety-related functions.

Potential Impact Number: 23215 **SRP Section Number:** 3.10

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

Section 3.10, discusses assuring operability of active pumps and valves by limiting the stresses to the material elastic limit when the component is subjected to normal operating, seismic and dynamic loads.

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Potential Impact Number: 23223 **SRP Section Number:** 5.4.1.1

Document: DRAFT SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

Section 5.4.1.1 contains a reference to ASTM E-208. The DSER contained an open item requesting that the applicant should verify that specific editions of all national codes and standards referenced in the CESSAR have been identified.

Potential Impact Number: 23224 **SRP Section Number:** 5.4.1.1

Document: DRAFT SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

Section 5.4.1.1 contains a reference to ASTM A-370. The DSER contained an open item requesting that the applicant should verify that specific editions of all national codes and standards referenced in the CESSAR have been identified.

Potential Impact Number: 23252 **SRP Section Number:** 3.5.1.4

Document: NRC MEMORANDUM 11/07/83, from Harold Denton to Victor Stello
Revision Date: 11/7/83
Title: POSITION ON USE OF PRA IN TORNADO MISSILE PROTECTION LICENSING ACTIONS

Summary:

Discusses staff position on the use of probabilistic risk assessment (PRA) in tornado missile protection licensing actions.

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Potential Impact Number: 23261 **SRP Section Number:** 3.5.1.6

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Locations: 52.47(a)(1)(iii), 52.47(a)(1)(iii), 52.79(a)(1), Appendix M, paragraph 2, Appendix O paragraph 3

Summary:

Applications for Standard Design Certification must contain the site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters.

If an application for a Combined License references an Early Site Permit, the application need not contain information or analysis submitted in connection with the permit. The application must contain information sufficient to demonstrate that the design falls within the parameters specified in the Early Site Permit and to resolve any other environmental issue not considered in any previous proceeding on the site or the design.

Applications filed for a manufacturing license (authorization to manufacture but not construct or install at a specific location) shall contain the technical information required by Section 50.34(a)(1) except that the preliminary safety analysis report shall be designated as a 'design report' and any required information or analyses relating to site matters shall be predicated on postulated site parameters which must be specified in the application.

Submittals for a review of a standard design shall include the site parameters postulated for the design, and an analysis and evaluation of the design in terms of such postulated site parameters.

Potential Impact Number: 23262 **SRP Section Number:** 3.5.1.6

Document: FINAL SER CE80 CH 2
Title: SITE CHARACTERISTICS

Summary:

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Pursuant to 10 CFR Part 52.47(a)(1)(iii) an applicant seeking design certification is required to (1) provide site parameters postulated for the design, and (2) analyze and evaluate the design in terms of these parameters. In the CE80+ FSER, the staff listed the following parameters to be included in the site parameter envelope: Groundwater; Maximum Flood (or Tsunami) Level; Precipitation (Maximum Rainfall Rate for Roof Design); Precipitation (Snow Load); Design Temperatures (Ambient Air and Cooling Water); Extreme Wind; Tornado; Soil Properties (Minimum Soil-bearing Capacity); Seismology (Design Ground Response Spectra); Aircraft Hazards; and, Meteorology (Short-term Dilution Factor X/Q and Long-term Dilution Factor X/Q).

As stated in Section 2.2.1 of CESSAR-DC, ABB-CE proposes that a site for the System 80+ plant be accepted without further review if the distances from the plant meet the following requirements which are from SRP Section 3.5.1.6, "Aircraft hazards":

(1) The plant-to-airport distance (D) is between 8 and 16 kilometers (5 and 10 miles), and the projected annual number of operations is less than 195 D2 (500 D2 for a distance between 5 and 10 miles); or if D is greater than 16 kilometers (10 miles), and the projected annual number of operations is less than 390 D2 (1000 D2 for a distance greater than 10 miles).

(2) The plant is at least 8 kilometers (5 miles) from the edge of military training routes including low-level training routes, except for those associated with a usage greater than 1000 flights per year, or where activities (such as practice bombing) may create unusual stress conditions.

(3) The plant is at least 3.2 kilometers (2 miles) beyond the nearest edge of a Federal airway, holding pattern, or airport.

If these site proximity acceptance criteria are not met, or if sufficiently hazardous military activities are identified, the COL applicant will be required to submit a detailed review of the aircraft hazards in the COL application. These parameters are included in CESSAR-DC Table 2.0-1.

The proposed site requirements in CESSAR-DC Section 2.1.1 meet the criteria of SRP Section 3.5.1.6 and are, therefore, acceptable. The criteria of SRP Section 3.5.1.6 are based on the assumption that adherence to these criteria results in very low offsite release probability (less than 10-7/reactor-year).

The staff reiterated the site-specific nature of aircraft hazards and further discussed the COL action described above in FSER Section 3.5.1.6.

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Potential Impact Number: 23263 **SRP Section Number:** 11.4

Document: FINAL SER EPRI CH 12
Title: EPRI URD-EPD: RADIOACTIVE WASTE PROCESSING SYSTEMS

Summary:

With regard to shipping containers for the processed wet solid wastes, EPRI has identified an interface requirement for the plant designer in Section 5.2.1.4 of Chapter 12. This interface requirement requires the plant designer to interface with the shipping container supplier and with disposal sites to ensure compatibility with system designs.

Although the Evolutionary Requirements Document does not include an explicit interface requirement for the shipping container for processed dry solid wastes, the staff concludes that the design of dry waste shipping containers also is not within the scope of the Evolutionary Requirements Document. For the above reasons, the staff will review the design of shipping containers both for wet and dry solid wastes against the acceptance criteria in SRP Section 11.4, 'Solid Waste Management Systems,' and applicable sections of 10 CFR Part 61 during its review of individual applications for FDA/DC.

Potential Impact Number: 23264 **SRP Section Number:** 11.4

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

TMI Action Plan Item IV.C.1 required that lessons learned from TMI be extended to other key NRC programs where a potential exists for nuclear accidents including, but not restricted to, the transportation of nuclear materials, waste management, research reactors, fuel facilities, and Category I materials licensees.

Studies performed by NMSS resulted in the issuance of a Draft BTP on Waste Form on October 30, 1981 which incorporated the resin degradation experience gained from the EPICOR-II system design used at TMI-2. This BTP was issued in February 1983. As a result, all necessary work on this item has been completed.

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Potential Impact Number: 23265 **SRP Section Number:** 11.4

Document: ADVANCE SER ABWR CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

By SSAR amendments up to and including 32, GE identified COL License Information (Section 11.4.3) which states the COL applicant will provide detailed information to demonstrate that the wet waste solidification process will result in a product that complies with 10 CFR 61.56. GE has also included this action item in the final certified SSAR.

Because the establishment and implementation of a process control program for solidifying the evaporator concentrates, using an approved solidification agent, and the dewatering process of the spent resins and filter sludges are dependent on the as-procured equipment for the ABWR standard design, the staff will review the PCP and the dewatering process for each referencing applicant against BTP ETSB 11-3.

By SSAR amendments up to and including 32, GE identified COL License Information (Section 11.4.3.1, Item 3) which states the COL applicant will provide a PCP for solidifying the evaporator concentrates using an approved solidification agent and the dewatering process of the spent resins and filter sludges. GE has also included this action item in the final certified SSAR.

In section 11.4.2 the staff concluded that the solid waste management system design met the requirements of 10 GDC 60, 63, and 64 and complied with 10 CFR 61.56 and 10 CFR Part 71.

Potential Impact Number: 23268 **SRP Section Number:** 10.3.6

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

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Cleaning and cleanliness control for Control Element Drive Mechanism materials are in accordance with ANSI/ASME NQA-2-1983, 'Quality Assurance Requirements for Nuclear Power Plants,' and RG 1.37, 'Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants.' The staff has previously reviewed ANSI/ASME NQA-2-1983 and finds it acceptable.

NQA-2-1983 is a departure from ANSI N45.2.1-1973 used in the current SRP 4.5.1 (1981).

Potential Impact Number: 23269 **SRP Section Number:** 10.3.6

Document: FINAL SER CE80 CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

ABB-CE identified in the CESSAR-DC the specific materials for the steam and feedwater system. Specifically, ABB-CE stated that carbon-manganese and chromium-molybdenum steels are to be used for the main steam and main feedwater systems.

ABB-CE's design and layout of piping systems considers the effect on the piping material from fluid velocity, bend location, and the location of flash points. The staff believes ABB-CE has demonstrated an appreciation of water characteristics, piping configuration, and materials selection, and that these factors will mitigate erosion and corrosion of piping systems. On this basis, DSER Open Item 10.3-5 is resolved.

Potential Impact Number: 23270 **SRP Section Number:** 10.3.6

Document: FINAL SER CE80 CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

In Amendment N, ABB-CE revised the CESSAR-DC to specify the main steam, hot reheat, condensate, main feedwater piping systems, and the heater drain piping systems upstream of the drain control valves are carbon steel or equivalent. However,

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. Stainless steel will be used as tubing in various heat exchangers and as cladding of tubesheets in low chloride environments, with alloy content increasing with increasing chloride levels. For chloride levels above 800 ppm, high concentrations of dissolved solids (above 1000 ppm), or water contaminated by sewage discharges, titanium tubing will be used. These material selections have been used in numerous plants for twenty or more years, and there have been no failures under the stated environments. Accordingly, ABB-CE has provided adequate assurance of the safety and structural integrity of these systems for the 60 year life of the plant. On this basis, DSER Open Item 10.3-4 is resolved.

Potential Impact Number: 23274 **SRP Section Number:** 10.3.6

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Section 5.3.2 of Chapter 1 imposes the following requirements on materials in feedwater, steam, and condensate systems: - The use of copper alloys is prohibited for PWR components that will be in contact with feedwater, steam, or condensate. Copper alloys may be used in BWRs in certain applications for service conditions of 200 F or less; however, all such applications will be identified by the plant designer, who will document the basis for their acceptability on the basis of current experience.

- The ALWR design will incorporate into the secondary system component design the material considerations identified in EPRI NP-2294, "Guide to Design of Secondary Systems and Their Components To Minimize Oxygen-Induced Corrosion."

- Corrosion/erosion-resistant materials will be used for all components exposed to wet steam or flashing liquid flow where significant erosion could occur. The degree of corrosion/erosion resistance of the material will be consistent with the temperature, moisture content, and velocity of the wet steam to which the component is exposed. Plain carbon steel with no deliberate alloying additions other than carbon and manganese will [not] be used for this application.

The staff concludes that the implementation of these requirements will ensure that materials in feedwater, steam, and condensate systems will perform as designed.

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Potential Impact Number: 23278 **SRP Section Number:** 4.6

Document: NRC GENERIC LETTER 80-107
Title: BWR SCRAM DISCHARGE SYSTEM

Summary:

This letter transmits the staff's Generic Safety Evaluation Report (SER) regarding BWR Scram Discharge System dated December 1, 1980. The SER recommends certain short-term actions which are considered necessary to assure the continued safe operation of General Electric (GE) BWR's with inadequate SDV-IV hydraulic coupling until permanent design changes are made.

In addition, the criteria for permanent design changes affecting all operating GE BWR's is included in the SER.

Potential Impact Number: 23279 **SRP Section Number:** 6.2.1.1.A

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

Standard Review Plan (SRP) Section 6.2.1.1.C, states that, for Mark III plants at the construction permit stage, the containment design pressure 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. GE's calculated drywell peak pressure for the FWLB is 268 kPa (39 psig). The design pressure for the drywell is 310 kPa (45 psig), which provides a margin of 15% above the peak calculated pressure in the drywell and is equal to the margin recommended in the SRP. Therefore, this design margin for containment pressure is acceptable.

The calculated wetwell peak pressure is 179 kPa (26 psig). The design pressure for the wetwell is 310 kPa (45 psig), which provides a margin of 42% above the peak calculated pressure in the wetwell. The calculated drywell-to-wetwell peak differential

pressure is 110.3 kPa (16 psig) and the design drywell-to-wetwell differential pressure is 172.3 kPa (25 psig), which provides a design margin of 56%.

The staff concludes that the containment pressure transient following a LOCA in the ABWR containment is consistent with SRP 6.2.1 and, therefore, is acceptable.

Potential Impact Number: 23280 **SRP Section Number:** 6.2.1.1.B

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

Standard Review Plan (SRP) Section 6.2.1.1.C, states that, for Mark III plants at the construction permit stage, the containment design pressure 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. GE's calculated drywell peak pressure for the FWLB is 268 kPa (39 psig). The design pressure for the drywell is 310 kPa (45 psig), which provides a margin of 15% above the peak calculated pressure in the drywell and is equal to the margin recommended in the SRP. Therefore, this design margin for containment pressure is acceptable.

The calculated wetwell peak pressure is 179 kPa (26 psig). The design pressure for the wetwell is 310 kPa (45 psig), which provides a margin of 42% above the peak calculated pressure in the wetwell. The calculated drywell-to-wetwell peak differential pressure is 110.3 kPa (16 psig) and the design drywell-to-wetwell differential pressure is 172.3 kPa (25 psig), which provides a design margin of 56%.

The staff concludes that the containment pressure transient following a LOCA in the ABWR containment is consistent with SRP 6.2.1 and, therefore, is acceptable.

Potential Impact Number: 23281 **SRP Section Number:** 10.3.6

Document: FINAL SER CE80 CH 17
Title: QUALITY ASSURANCE

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Summary:

The CE 80+ FSER (Chapter 17) states that NQA-2 has superseded the ANSI standards referenced by RGs 1.30. 1.37, 1.38, 1.39, 1.94, and 1.116.

Potential Impact Number: 23282 **SRP Section Number:** 4.5.2

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

ABB-CE proposes to use Stellite, a cobalt-based alloy, as a hard-facing material. As discussed in Section 4.5.1 of this report, ABB-CE states that cobalt-based alloys will be avoided unless no proven alternative exists. Cobalt-free alloys with the wear and corrosion properties of the Stellite (cobalt-base) alloys are under development, but have not been fully demonstrated to be as usable as the Satellites. Accordingly, the staff acknowledges that at this time the availability of alternative materials for Satellites is limited. On this basis, DSER Open Item 4.5.2-1 is resolved.

The current SRP 4.5.2 (1981) provides no guidance for using cobalt alloys such as stellites as reactor internal and core support materials.

Potential Impact Number: 23283 **SRP Section Number:** 4.5.2

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

In CESSAR-DC Section 4.5.2.1, ABB-CE indicates that Inconel 690 will be used 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 and secondary coolant loops because it is more corrosion-resistant than Inconel 600. The use of Inconel 690 and the equivalent weld metals (Types 52 and 152) will give reasonable assurance of the material integrity of the components in contact with reactor coolant. On this basis, DSER Open Item 4.5.2-4 is resolved.

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Potential Impact Number: 23284 **SRP Section Number:** 4.5.2

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

The ferrite content limits for austenitic stainless weld metal in the CESSAR-DC were broader than those in industry guidelines (EPRI, "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, ALWR Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991) and staff guidance (USNRC, "Technical Report on Materials Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," NUREG-0313, Revision 2, January 1988). In CESSAR-DC Amendment L, ABB-CE revised the CESSAR-DC to be consistent with industry guidelines and staff guidance. The revision should assure an acceptable level of structural integrity for stainless steel welds over the life of the plant. On this basis, DSER Open Item 4.5.1-5 is resolved.

As discussed in Section 4.5.1 of the SER, the ferrite content limits for austenitic stainless steel castings and weld metal given in the CESSAR-DC were broader than those in industry guidelines and staff guidance. In Amendment N to the CESSAR-DC, ABB-CE revised the ferrite content of weld metal to be consistent with industry guidelines. In Amendment Q to the CESSAR-DC, ABB-CE reduced the upper limit of ferrite content in stainless steel castings with normal operating temperatures above 260 C (500 F) (as expressed in ferrite number (FN)) to 20 to mitigate the effects of thermal embrittlement over the life of the castings in service. This reduction will offer reasonable assurance of an acceptable level of fracture toughness for stainless steel castings over the life of the plant. On this basis, DSER Open Item 4.5.2-5 is resolved.

In Section 4.5.2 of the FSER, the staff found that the materials used for constructing the components of the reactor internals and core support structures have been identified by specification and conform with the requirements of NG-2000 of Section III and Parts A, B, and C of Section II of the ASME Code.

Potential Impact Number: 23285 **SRP Section Number:** 4.5.2

Document: FINAL SER CE80 CH 5

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Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The concern relating to the environmental effects on fatigue is discussed in Section 5.2.3 of the SER. This is applicable to all materials. The design of System 80+ components will address the potential influence of environmental effects on the fatigue life of materials over the 60-year design life. The issue of environmental effects on fatigue is currently under consideration by a special steering committee for cyclic life and environmental effects in nuclear applications of the PVRC.

System 80+ components will be designed to the ASME B&PV Code requirements. ABB-CE addressed the issue of environmental fatigue of Class 1 components in Amendment R to Section 3.9.1 of CESSAR-DC and the staff's evaluation is provided in Section 3.12.5.7 of the SER. On this basis, DSER Open Item 5.2.3-6 is resolved.

Potential Impact Number: 23286 **SRP Section Number:** 4.5.2

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

The applicant is proposing to use Type 304 austenitic stainless steel. However, these materials are susceptible to intergranular stress corrosion cracking. In the DSER, the staff indicated that the applicant should consider using low-carbon, wrought austenitic stainless steel, which includes Types 304L, 316L, 304NG, 316NG, and modified Type 347 (Open Item 4.5.2-3).

ABB-CE proposes to use Type 304 austenitic stainless steel. However, this material is susceptible to IGSCC when the oxygen content of the reactor coolant exceeds 0.010 ppm at temperatures above 200 F during normal operations. During startup and operation of the ABB-CE NSSS, these temperature and chemical conditions are maintained through specified chemistry control. On this basis, DSER Open Item 4.5.2-3 is resolved.

Potential Impact Number: 23287 **SRP Section Number:** 4.5.2

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Potential Impacts

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

In Section 5.3.A.1 of the DSER for Chapter 1, the staff stated that EPRI should revise the material selection for reactor coolant pressure boundary piping to prevent IGSCC of austenitic stainless steel. The staff has recommended that licensees and applicants follow Revision 2 of NUREG-0313 to prevent IGSCC in stainless steel. In the EPRI Evolutionary Plant FSER the staff concludes that the implementation of the requirements in Section 5.3.1.8 of Chapter 1, following Revision 2 of NUREG-0313, and using adequate field and shop fabrication processes will ensure that austenitic stainless steel will be resistant to IGSCC in service. The staff further indicated that it will evaluate field and shop fabrication processes during its review of an individual application for a combined license.

Section 5.3.1.8 of Chapter 1 specifies that austenitic stainless steel must be resistant to intergranular stress corrosion cracking (IGSCC). All austenitic stainless steel that is in contact with BWR reactor coolant at a temperature above 200 F during power operation or that is part of the BWR RCS pressure boundary or that is in borated water at any temperature in a PWR and that is welded without subsequent solution heat treatment will meet the requirements below.

For PWR service, only low-carbon wrought austenitic stainless steel, which includes Types 304L, 316L, 304NG, 316NG, and modified 347, will be used.

For BWR service, only low-carbon wrought austenitic stainless steel, Types 304NG, 316NG, and modified 347 with a maximum carbon content of 0.02 percent, will be used.

These materials will be tested for resistance to sensitization in accordance with Section 5.4.2.4 of Chapter 1 of the Evolutionary Requirements Document.

Incoming components, parts, raw materials, and heat-treated parts will be examined for excessive intergranular attack (IGA) unless a minimum of 0.030-inch of metal is removed from all as-received surfaces during fabrication.

Potential Impact Number: 23292 **SRP Section Number:** 4.6

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Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Generic Issue 25 Discussion:

This issue concerns the slow loss of control air pressure in the scram system of BWR's. Air pressure dropping at a certain rate will first allow some of the CRD scram outlet valves to open slightly, thus filling the scram discharge volume with water but allowing little or no control rod movement. The technical resolution of this issue resulted in a generic safety evaluation report identifying new guidance for the scram discharge volume designs. The generic safety evaluation report is documented in a memorandum for G. Lainas, et al., from P. Check, "BWR Scram Discharge System Safety Evaluation," December 1, 1980.

Generic Issue 41 Discussion:

On June 28, 1980, during a routine shutdown of the Browns Ferry Unit 3 reactor, a manual scram from approximately 36% power failed to insert about 40% of the control rods. As part of the response to this event at Browns Ferry Unit 3, followup investigations and tests were performed by the licensees. As a result, a number of other deficiencies were uncovered.

The corrective measures are divided into a short-term program and a long-term program. The short-term actions were implemented by I&E Bulletins. The long-term program is the concern of this issue.

The objective of the long-term program was the improvement of the SDV design. The long-term program was devised by an NRR task force working with a subgroup of the BWR Owners' Group and with OIE. Their generic recommendations have been published. [NRC Letter to All BWR Licensees, "BWR Scram Discharge System," December 9, 1980.]

Implementation is being tracked by MPA No.B-58. Because all generic work is complete and implementation is in progress, this issue is considered RESOLVED.

Potential Impact Number: 23295 **SRP Section Number:** 4.5.2

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Potential Impacts

Document: FINAL SER ABWR CH 4
Title: REACTOR

Summary:

The controls imposed on the reactor coolant chemistry satisfy the chemistry limits specified in EPRI report NP-4947, "BWR Hydrogen Water Chemistry Guidelines 1987 Revision," December 1988. This will provide reasonable assurance that the reactor internal and the core support structures will be adequately protected during operation from conditions that could lead to stress corrosion, including irradiation-assisted stress corrosion cracking and loss of component structural integrity.

The materials to be used for the construction of components of the reactor internals and core support structures were identified in SSAR Section 4.5.2 by specification and found to be in conformance with the requirements of NG-2000 of Section III and Parts A, B, and C of Section II of the ASME Code.

Potential Impact Number: 23305 **SRP Section Number:** 6.6

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

GE has committed in SSAR Subsection 6.6.7.2 to a monitoring program for erosion-corrosion that provides assurances that procedures or administrative controls are in place to assure that the NUMARC program (or another equally effective program) is implemented and the structural integrity of all high-energy (two-phase as well as single- phase) carbon-steel systems 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.

Potential Impact Number: 23306 **SRP Section Number:** 6.5.1

Document: ADVANCE SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

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Potential Impacts

Summary:

GE committed to provide an analysis to demonstrate that the use of the system during the inerting, deinerting, pressure control, or purging of the primary containment during normal plant operation does not impair its functional capability during a DBA. Therefore, the staff identified the submittal of the subject analysis as Open Item 6.5.1-2 in the DFSER. However, 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). BY SSAR Amendment 32, GE stated that the normal operation of the SGTS use is much less than 90 hours per year for both trains combined during startup, power, and hot shut down modes of operation. 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 to perform its intended function in the event of a LOCA, since this would be a change to SSAR certified design information. The staff finds this acceptable. Therefore, DFSER Open Item 6.5.1-2 is resolved.

Potential Impact Number: 23308 **SRP Section Number:** 4.6

Document: FINAL SER ABWR CH 4
Title: REACTOR

Summary:

From the Discussion of NUREG-0933 Issue 25 in Section 20.2.3:

The ABWR design incorporates two features to prevent the loss or impairment of the scram function because of slow loss of control air in the system: (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. Based upon this information, the staff concluded that GE adequately addressed issue 25 for the ABWR design.

From the Discussion in Section 4.6:

The safety concerns associated with a pipe break, given in NUREG-0803, "Safety Concerns Associated With a Pipe Break in the BWR Scram System," are not applicable for ABWR. The ABWR design does not include scram discharge volume piping. The

water displaced by the CRD during the scram will be routed to the reactor pressure vessel.

Potential Impact Number: 23313 **SRP Section Number:** 6.3

Document: NRC MEMORANDUM 04/02/93, from W. Minners to F. Gillespie
Title: PROPOSED RESOLUTION OF GENERIC ISSUE 105, 'INTERFACING SYSTEMS LOCA IN LWR'S'

Summary:

RES proposed a new SRP section as part of the resolution of Generic Issue 105.

Potential Impact Number: 23314 **SRP Section Number:** 6.3

Document: NRC MEMORANDUM 06/03/93, from E. Beckjord to J. Taylor
Title: TECHNICAL RESOLUTION OF GENERIC ISSUE 105, 'INTERFACING SYSTEMS LOSS OF COOLANT ACCIDENT IN LWR'S'

Summary:

In the memorandum presenting the resolution of Generic Issue 105, RES informed the EDO that a draft SRP section covering design review of systems interfacing with the reactor coolant system in ALWRs was provided to NRR for information and use as appropriate

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Potential Impact Number: 23315 **SRP Section Number:** 6.3

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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This Commission paper expresses the NRC position on interfacing system LOCA in advanced reactor designs, both evolutionary and passive.

Potential Impact Number: 23316 **SRP Section Number:** 6.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.63

Summary:

10 CFR 50.63 (a)(2) in part 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.

Potential Impact Number: 23321 **SRP Section Number:** 3.4.1

Document: NUREG 1229
Title: REGULATORY ANALYSIS FOR RESOLUTION OF USI A-17

Summary:

Section 6.7 of this regulatory analysis documents that the staff plans, as part of the resolution of USI A-17, to include in the standard review plan for future plants specific guidance regarding protection from internal flooding and water intrusion events.

Potential Impact Number: 23322 **SRP Section Number:** 5.4.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.63(a)

Summary:

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10 CFR 50.63 (a)(2) provides requirements for ensuring adequate core cooling in the event of a station blackout for the specified duration.

Potential Impact Number: 23323 **SRP Section Number:** 5.4.6

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

Regulatory Position C.3 includes staff guidance applicable to ensuring the ability to cope with a station blackout for the specified duration.

Potential Impact Number: 23324 **SRP Section Number:** 9.4.1

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

Regulatory Position C.3 includes staff guidance applicable to ensuring the ability to cope with a station blackout for the specified duration.

Potential Impact Number: 23325 **SRP Section Number:** 9.4.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.63(a)(2)

Summary:

10 CFR 50.63(a)(2) provides requirements for ensuring adequate core cooling and containment integrity in the event of a station blackout for the specified duration.

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Potential Impact Number: 23327 **SRP Section Number:** 3.4.1

Document: C&S: ANSI N170
Title: Standards for Determining Design Basis Flooding at Power Reactor Sites

Summary:

Regulatory Guide 1.59 is cited in SRP Section 3.4.1 as reviewer guidance. Regulatory Guide 1.59 references ANSI N170-1976. The current version of this standard is ANS 2.8-1992. Note that ANSI N170-1976 is not directly cited in SRP Section 3.4.1.

Potential Impact Number: 23328 **SRP Section Number:** 12.3 - 12.4

Document: NUREG 1446
Title: STANDARDS FOR PROTECTION AGAINSH RADIATION - 10 CFR PART 20

Summary:

On May 21, 1991, the Nuclear Regulatory Commission issued a revision to its standards for protection against ionizing radiation, 10 CFR Part 20.

NUREG-1446 (microfiche - 60207:182) is a comparative text of the new and old portions of 10CFR20 organized to show the full text of each set of regulations in a side-by-side format for ease of comparison. Paragraph numbers in the old and new portions of 10CFR20 do not exhibit one-for-one correspondence between the old and new paragraph content. NUREG-1446 combines relevant text, wherever it appears, in the old portion of 10CFR20 to present it side-by-side with new paragraphs. There are instances of "No Corresponding Provision" in both directions.

Potential Impact Number: 23331 **SRP Section Number:** 12.1

Document: ADVANCE SER CE CH 12
Title: RADIATION PROTECTION

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Summary:

The amended 10 CFR Part 20 requires that all licensees develop, document, and implement a radiation protection program. This program should encompass the ALARA concept and includes provisions for maintaining radiation doses and intakes of radioactive materials ALARA.

Potential Impact Number: 23332 **SRP Section Number:** 12.1

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION
Location: Part 20.1101

Summary:

20.1101 Radiation protection programs. (a) Each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of this part. (See 20.2102 for recordkeeping requirements relating to these programs.) (b) The licensee shall use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA). (c) The licensee shall periodically (at least annually) review the radiation protection program content and implementation.

Potential Impact Number: 23334 **SRP Section Number:** 9.3.4

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue addresses rupture of low pressure piping systems which interface with the reactor coolant pressure boundary due to pressure isolation system failure. This issue was assigned a high priority for resolution.

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Potential Impact Number: 23335 **SRP Section Number:** 9.3.4

Document: NRC MEMORANDUM 04/02/93, from W. Minners to F. Gillespie
Title: PROPOSED RESOLUTION OF GENERIC ISSUE 105, 'INTERFACING SYSTEMS LOCA IN LWR'S'

Summary:

RES proposed a new SRP section as part of the resolution of Generic Issue 105.

Potential Impact Number: 23336 **SRP Section Number:** 9.3.4

Document: NRC MEMORANDUM 06/03/93, from E. Beckjord to J. Taylor
Title: TECHNICAL RESOLUTION OF GENERIC ISSUE 105, 'INTERFACING SYSTEMS LOSS OF COOLANT ACCIDENT IN LWR'S'

Summary:

In the memorandum presenting the resolution of Generic Issue 105, RES informed the EDO that a draft SRP section covering design review of systems interfacing with the reactor coolant system in ALWRs was provided to NRR for information and use as appropriate.

Potential Impact Number: 23338 **SRP Section Number:** 9.3.4

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016 the staff recommended that the Commission approve the following positions on intersystem LOCAs for evolutionary ALWRs. The staff concluded that designing, to the extent practicable, low-pressure systems to withstand full RCS pressure is an acceptable means for resolving the intersystem LOCA issue. However,

the staff believes that for those systems that have not been designed to withstand full RCS pressure, evolutionary ALWRs should provide:

(1) the capability for leak testing of the pressure isolation valves;

(2) valve position indication that is available in the control room when isolation valve operators are deenergized; and,

(3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

In the SRM for SECY 90-016, the Commission approved the staff's position on intersystem LOCA, provided that all elements of the low-pressure system are considered (including instrument lines, pump seals, heat exchanger tubes, and valve bonnets). This issue was also addressed in SECY 93-087 with the only change being extending the intersystem LOCA criteria to the passive plants.

Potential Impact Number: 23340 **SRP Section Number:** 9.3.4

Document: NRC BULLETIN 86-03
Title: POTENTIAL FAILURE OF MULTIPLE ECCS PUMPS DUE TO SINGLE FAILURE OF AIR-OPERATED VALVE IN MINIMUM FLOW RECIRCULATION LINE

Summary:

This Bulletin required actions to determine whether or not a single-failure vulnerability exists in the minimum flow recirculation line of any ECCS pumps that could cause a failure of more than one ECCS train.

The miniflow lines for the CVCS pumps could be affected by the actions requested in this Bulletin.

Potential Impact Number: 23341 **SRP Section Number:** 9.3.4

Document: NRC BULLETIN 88-04

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Title: POTENTIAL SAFETY-RELATED PUMP LOSS

Summary:

This Bulletin addressed the potential for dead-heading of ECCS pumps that have a common mini-flow line. In addition, questions concerning the installed mini-flow line capacity were also addressed. These positions could have an impact on the CVCS system pumps.

Potential Impact Number: 23342 **SRP Section Number:** 9.3.4

Document: NRC GENERIC LETTER 89-04
Title: GUIDANCE OF DEVELOPING ACCEPTABLE INSERVICE TESTING PROGRAMS

Summary:

In cases where only the minimum-flow return line is available for pump testing, regardless of the test interval, the staff's position is that flow instrumentation which meets the requirements of IWP-4110 and 4120 must be installed in the mini-flow return line. Installation of this instrumentation is necessary to provide flow rate measurements during pump testing so this data can be evaluated with the measured pump differential pressure to monitor for pump hydraulic degradation.

NRC Bulletin 88-04, dated May 5, 1988, advised licensees of the potential for pump damage while running pumps in the minimum-flow condition. The above guidelines for meeting the Code or performing alternative testing is not intended to supersede the thrust of this Bulletin. Licensees should ensure that if pumps are tested in the low flow condition, the flow is sufficient to prevent damage to the pump.

Potential Impact Number: 23347 **SRP Section Number:** 9.3.4

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

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In this Section EPRI developed an approach for addressing the high rate of reactor coolant pump (RCP) seal failures that challenge the makeup capacity of the emergency core cooling systems in PWRs.

In its DSER for Chapter 3, the staff concluded that some of the design requirements proposed by EPRI might not be practical unless adequate pump seals were developed and/or independent seal cooling was provided. The staff pointed out the difficulties in meeting some of EPRIs design requirements concerning seal leak rates and meeting the time limitations imposed by the station blackout analysis.

Issue 23 has not yet been generically resolved by the staff and has a high safety priority. Unless the final agency resolution of this issue dictates other actions for ALWR evolutionary plant designs, the staff 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.

Potential Impact Number: 23350 **SRP Section Number:** 4.5.2

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

The staff has recommended that licensees and applicants follow Revision 2 of NUREG-0313, Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping, to prevent IGSCC in stainless steel. In addition, it is important that adequate field and shop fabrication processes be used to minimize the sensitization of materials to IGSCC. Therefore, the staff will evaluate this matter during its review of an individual application for a combined license.

Potential Impact Number: 23351 **SRP Section Number:** 4.5.2

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

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In the DSER (SECY-91-153), the staff noted that the materials used for the construction of RCPB components had been identified by specification and were in conformance with Section III of the ASME Code and NUREG-0313. However, the staff requested that GE use Revision 2 (not Revision 1 as proposed by GE) of NUREG-0313. GE revised the SSAR to reference NUREG-0313, Revision 2. GE's compliance with the provisions of the Code for material specifications and conformance with NUREG-0313, Revision 2, satisfy the quality standards of GDC 1 and 30 and 10 CFR 50.55a.

Potential Impact Number: 23352 **SRP Section Number:** 4.5.1

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

The staff has recommended that licensees and applicants follow Revision 2 of NUREG-0313, Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping, to prevent IGSCC in stainless steel. In addition, it is important that adequate field and shop fabrication processes be used to minimize the sensitization of materials to IGSCC. Therefore, the staff will evaluate this matter during its review of an individual application for a combined license.

Potential Impact Number: 23353 **SRP Section Number:** 4.5.1

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In the ABWR DFSER, the NRC staff requested that GE use Revision 2 (not Revision 1 as proposed by GE) of NUREG-0313. GE revised the SSAR to reference NUREG-0313, Revision 2. GE's compliance with the provisions of the Code for material specifications and conformance with NUREG-0313, Revision 2, satisfy the quality standards of GDC 1 and 30 and 10 CFR 50.55a.

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Potential Impact Number: 23354 **SRP Section Number:** 5.2.3

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

Section 5.2.3 of the FSER indicates that the controls to avoid stress corrosion cracking in RCPB components constructed of austenitic stainless steels conform to the recommendations of RG 1.44, RG 1.37, and NUREG-0313, Revision 2. Cold work of austenitic stainless steel is controlled by applying limits on hardness, bend radii, and surface finish on ground surfaces. The controls to be followed during material selection, fabrication, examination, and protection, in accordance with these recommendations, in order to prevent excessive yield strength, sensitization, and contamination provide reasonable assurance that the RCPB components of austenitic stainless steels will be in a metallurgical condition that minimizes susceptibility to stress corrosion cracking during service.

Section 5.3.1 of the FSER also notes 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 recommendations of RGs 1.37, Revision 0, 1.44, Revision 0, and NUREG 0313, Revision 2.

Potential Impact Number: 23355 **SRP Section Number:** 5.2.3

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

The staff has recommended that licensees and applicants follow Revision 2 of NUREG-0313, Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping, to prevent IGSCC in stainless steel. In addition, it is important that adequate field and shop fabrication processes be used to minimize the sensitization of materials to IGSCC. Therefore, the staff will evaluate this matter during its review of an individual application for a combined license.

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Potential Impact Number: 23356 **SRP Section Number:** 5.4.2.1

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

The staff has recommended that licensees and applicants follow Revision 2 of NUREG-0313, Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping, to prevent IGSCC in stainless steel. In addition, it is important that adequate field and shop fabrication processes be used to minimize the sensitization of materials to IGSCC. Therefore, the staff will evaluate this matter during its review of an individual application for a combined license.

Potential Impact Number: 23358 **SRP Section Number:** 10.3.6

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In the ABWR FSER, the NRC staff approved GE's use Revision 2 of NUREG-0313 to address Intergranular Stress Corrosion Cracking (IGSCC).

Potential Impact Number: 23359 **SRP Section Number:** 10.3.6

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

The staff has recommended that licensees and applicants follow Revision 2 of NUREG-0313, Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping, to prevent IGSCC in stainless steel. In addition, it is important that adequate field and shop fabrication processes be used to minimize the

sensitization of materials to IGSCC. Therefore, the staff will evaluate this matter during its review of an individual application for a combined license.

Potential Impact Number: 23362 **SRP Section Number:** 4.5.2

Document: NRC GENERIC LETTER 88-01
Title: NRC POSITION ON IGSCC IN BWR AUSTENITIC STAINLESS STEEL PIPING

Summary:

The purpose of this Generic Letter is to seek information regarding implementation of new staff positions covering technical areas related to intergranular stress corrosion cracking in BWR austenitic stainless steel piping. Attachment A to the letter defines staff postions on numerous subjects, including: Staff Positions on Materials; Staff Position on Processes: Staff Position on Water Chemistry; and Staff Position on Weld Overlay Reinforcement.

The technical bases for these postions are detailed in NUREG-0313, Rev. 2, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." This revision to NUREG-0313 was a major task in the staff long range plan to deal with BWR pipe cracking that was presented to the Commission in SECY 84-301. This revision includes the relevant recommedations of the Piping Review Committee Task Group on Pipe Cracking issued as NUREG-1061, Vol. 1, "Report of USNRC Piping Review Committee," and consideration of public comments on that document. NUREG-0313, Rev. 2, describes the technical bases for the staff positions on materials, processes, and primary coolant chemistry to minimize and control IGSCC problems.

Potential Impact Number: 23363 **SRP Section Number:** 4.5.1

Document: NRC GENERIC LETTER 88-01
Title: NRC POSITION ON IGSCC IN BWR AUSTENITIC STAINLESS STEEL PIPING

Summary:

The purpose of this Generic Letter is to seek information regarding implementation of new staff positions covering technical areas related to intergranular stress corrosion

cracking in BWR austenitic stainless steel piping. Attachment A to the letter defines staff postions on numerous subjects relating to SRP 5.2.3, including: Staff Positions on Materials; Staff Position on Processes: Staff Position on Water Chemistry; and Staff Position on Weld Overlay Reinforcement. The letter applies to all BWR piping made of austenitic stainless steel that is four inches or larger in nominal diameter and contains reactor coolant at a temperature above 200 degrees F during power operation regardless of Code classification. It also applies to reactor vessel attachments and appurtenances such as jet pump instrumentation penetration assemblies and head spray and vent components.

The technical bases for these postions are detailed in NUREG-0313, Rev. 2, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." This revision to NUREG-0313 was a major task in the staff long range plan to deal with BWR pipe cracking that was presented to the Commission in SECY 84-301. This revision includes the relevant recommedations of the Piping Review Committee Task Group on Pipe Cracking issued as NUREG-1061, Vol. 1, "Report of USNRC Piping Review Committee," and consideration of public comments on that document. NUREG-0313, Rev. 2, describes the technical bases for the staff positions on materials, processes, and primary coolant chemistry to minimize and control IGSCC problems.

Potential Impact Number: 23364 **SRP Section Number:** 6.1.1

Document: NRC GENERIC LETTER 88-01
Title: NRC POSITION ON IGSCC IN BWR AUSTENITIC STAINLESS STEEL PIPING

Summary:

The purpose of this Generic Letter is to seek information regarding implementation of new staff positions covering technical areas related to intergranular stress corrosion cracking in BWR austenitic stainless steel piping. Attachment A to the letter defines staff postions on numerous subjects, including: Staff Positions on Materials; Staff Position on Processes: Staff Position on Water Chemistry; and Staff Position on Weld Overlay Reinforcement.

The technical bases for these postions are detailed in NUREG-0313, Rev. 2, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." This revision to NUREG-0313 was a major task in the staff long range plan to deal with BWR pipe cracking that was presented to the Commission in

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SECY 84-301. This revision includes the relevant recommedations of the Piping Review Committee Task Group on Pipe Cracking issued as NUREG-1061, Vol. 1, "Report of USNRC Piping Review Committee," and consideration of public comments on that document. NUREG-0313, Rev. 2, describes the technical bases for the staff positions on materials, processes, and primary coolant chemistry to minimize and control IGSCC problems.

Potential Impact Number: 23365 **SRP Section Number:** 6.1.1

Document: RG 1.31 **Revision:** 3
Title: CONTROL OF FERRITE CONTENT IN STAINLESS STEEL WELD METAL

Summary:

This guide describes a method acceptable to the NRC staff for implementing NRC requirements/regulations with regard to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems in light-water-cooled nuclear power plants.

Potential Impact Number: 23366 **SRP Section Number:** 5.2.3

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

ABB-CE initially proposed to fabricate the pressurizer surge line and the RCS valves from cast austenitic stainless steel (SA351 CF8M). Thermal aging of cast austenitic stainless steel at the reactor temperature reduces the fracture toughness of the materials (U.S. NRC, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG/CR-4513, June 1991). ABB-CE needed to address the effect of thermal embrittlement for the 60-year plant design life.

In addition, it is difficult to inspect cast austenitic stainless steel using ultrasonic techniques. As discussed in the FSER, 10 CFR 50.55a(g) requires ASME Code Class 1, 2, and 3 components to be designed to enable the performance of inservice examination and preservice inspection in accordance with ASME Code Section XI.

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This regulation has no relief provisions for preservice inspection. A material that is difficult to inspect may not be able to conform to 10 CFR 50.55a(g).

Because of the thermal aging and inspection concerns, ABB-CE considered alternatives to cast materials and redesigned the pressurizer surge line using wrought materials.

In addition, ABB-CE considered alternative materials to cast stainless steels for other components. Where cast stainless steel was determined to be the best material selection for a specific application, such as large-diameter valve bodies, ABB-CE provided additional controls. These controls involved further restriction of the permissible ferrite content of the casting to minimize the effects of thermal aging. ABB-CE also considered the component service temperature for the 60-year design life since at lower operating temperatures, thermal aging is of less concern. The standard range of permissible ferrite content specified for stainless steel castings is from 8 percent to 30 percent.

In the FSER, the staff noted that the ferrite content limits for austenitic stainless steel castings and weld metal given in the CESSAR-DC were broader than those in industry guidelines (EPRI URD) and staff guidance (NUREG-CR/4513). ABB-CE eventually limited the ferrite content of austenitic stainless steel castings with normal operating temperatures above 500 F (260 C) to a maximum of 20 percent. ABB-CE also modified the upper ferrite limit to 15 percent for austenitic stainless steel weld metal. The staff believes the lower ferrite content limits for austenitic stainless steel castings and weld metal will provide reasonable assurance that components of these materials maintain adequate fracture toughness for their 60-year life.

Potential Impact Number: 23367 **SRP Section Number:** 5.3.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

Staff evaluation in Section 5.3.1 indicates that in Section 5.2.3 of this FSER, the staff noted the ferrite content limits for austenitic stainless steel castings and weld metal given in the CESSAR-DC were broader than those in industry guidelines (Ref. 1) and staff guidance (Ref. 2).

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References

(1) Electric Power Research Institute, 'Advanced Light Water Reactor Utility Requirements Document,' NP-6780-L, Volume 2, ALWR Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991.

(2) U.S. Nuclear Regulatory Commission, 'Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping,' NUREG-0313, Rev. 2, January 1988.

Potential Impact Number: 23368 **SRP Section Number:** 5.3.1

Document: NRC GENERIC LETTER 88-01
Title: NRC POSITION ON IGSCC IN BWR AUSTENITIC STAINLESS STEEL PIPING

Summary:

The purpose of this Generic Letter is to seek information regarding implementation of new staff positions covering technical areas related to intergranular stress corrosion cracking in BWR austenitic stainless steel piping. Attachment A to the letter defines staff postions on numerous subjects, including: Staff Positions on Materials; Staff Position on Processes: Staff Position on Water Chemistry; and Staff Position on Weld Overlay Reinforcement.

The technical bases for these postions are detailed in NUREG-0313, Rev. 2, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." This revision to NUREG-0313 was a major task in the staff long range plan to deal with BWR pipe cracking that was presented to the Commission in SECY 84-301. This revision includes the relevant recommedations of the Piping Review Committee Task Group on Pipe Cracking issued as NUREG-1061, Vol. 1, "Report of USNRC Piping Review Committee," and consideration of public comments on that document. NUREG-0313, Rev. 2, describes the technical bases for the staff positions on materials, processes, and primary coolant chemistry to minimize and control IGSCC problems.

Potential Impact Number: 23369 **SRP Section Number:** 5.4.2.1

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Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

Staff evaluation in Section 5.4.2 indicates that in Section 5.2.3 of this chapter, the staff notes that the ferrite content limits for austenitic stainless steel given in the CESSAR-DC are broader than those in industry guidelines and staff guidance. In Amendment L to the CESSAR-DC, ABB-CE modified the upper ferrite limit to 15 percent for austenitic stainless steel weld metal. The staff believes the lower ferrite content limits for austenitic stainless steel castings and weld metal will provide reasonable assurance that components of these materials maintain adequate fracture toughness for their 60 year life.

Potential Impact Number: 23370 **SRP Section Number:** 6.3

Document: NUREG 0718 **Revision:** 1 , 6/1/81
Title: LICENSING REQUIREMENTS FOR PENDING APPLICATIONS FOR CONSTRUCTION PERMITS AND MANUFACTURING LICENSE

Summary:

NUREG 0718 identifies the extent to which TMI Action Plan items apply to pending Construction Permit and Manufacturing License applications. Item II.E.2.1 is listed as "Category 2", indicating that a requirement of this type is customarily left for the OL stage.

Potential Impact Number: 23374 **SRP Section Number:** 4.5.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This item responds to a Piping Review Committee recommendation that regular grade 316 SS and 304 SS materials in BWR recirculation piping be replaced with an alloy which is resistant to intergranular stress corrosion cracking. This item is a regulatory

impact issue that requires updating of Regulatory Guide 1.44 to reflect the staff's findings in NUREG-0313, Rev. 2.

Potential Impact Number: 23376 **SRP Section Number:** 5.4.2.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This item responds to a Piping Review Committee recommendation that regular grade 316 SS and 304 SS materials in BWR recirculation piping be replaced with an alloy which is resistant to intergranular stress corrosion cracking. This item is a regulatory impact issue that requires updating of Regulatory Guide 1.44 to reflect the staff's findings in NUREG-0313, Rev. 2.

Potential Impact Number: 23377 **SRP Section Number:** 10.3.6

Document: NRC GENERIC LETTER 88-01
Title: NRC POSITION ON IGSCC IN BWR AUSTENITIC STAINLESS STEEL PIPING

Summary:

The purpose of this Generic Letter is to seek information regarding implementation of new staff positions covering technical areas related to intergranular stress corrosion cracking in BWR austenitic stainless steel piping. Attachment A to the letter defines staff positions on numerous subjects relating to SRP 5.2.3, including: Staff Positions on Materials; Staff Position on Processes: Staff Position on Water Chemistry; and Staff Position on Weld Overlay Reinforcement.

The technical bases for these postions are detailed in NUREG-0313, Rev. 2, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." This revision to NUREG-0313 was a major task in the staff long range plan to deal with BWR pipe cracking that was presented to the Commission in SECY 84-301. This revision includes the relevant recommedations of the Piping Review Committee Task Group on Pipe Cracking issued as NUREG-1061, Vol. 1, "Report of USNRC Piping Review Committee," and consideration of public comments

on that document. NUREG-0313, Rev. 2, describes the technical bases for the staff positions on materials, processes, and primary coolant chemistry to minimize and control IGSCC problems.

Potential Impact Number: 23382 **SRP Section Number:** 5.4.2.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Pipe cracking resulting from intergranular stress corrosion cracking can cause a LOCA, with the subsequent possibility of core melt. In 1988, NUREG-0313, Revision 2, was published and Generic Letter 88-01 was issued outlining the NRC positions on IGSCC in BWR austenitic stainless steel piping; MPA B-84 was established for implementation purposes. Thus, this issue was resolved with new requirements established.

Potential Impact Number: 23385 **SRP Section Number:** 12.3 - 12.4

Document: C&S: ANSI N101.6-72
Title: Guidelines on Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants

Summary:

ANSI N101.6-1972 is cited in SRP section 12.3-12.4 as a guidance document under the Regulatory Guide and other document Acceptance Criteria list at item number 20.

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.

Potential Impact Number: 23386 **SRP Section Number:** 12.3 - 12.4

Document: C&S: ANSI N16.2

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Title: Criticality Accident Alarm System

Summary:

ANSI N16.2-1969 is cited in SRP section 12.3-12.4 as a guidance document under the Regulatory Guide and other document Acceptance Criteria list at item number 19.

ANSI N16.2, "Criticality Accident Alarm Systems" is now numbered ANS 8.3. The 1986 version is current.

Potential Impact Number: 23387 **SRP Section Number:** 12.3 - 12.4

Document: NUREG 1430
Title: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS

Summary:

SRP Section 12.3-12.4 references the B&W STS (NUREG-0103) as they relate to radiation protection considerations. Improved STS have been promulgated for industry use (microfiche - 63882:002, 63881:020, and 63880:001).

Potential Impact Number: 23388 **SRP Section Number:** 12.3 - 12.4

Document: NUREG 1431
Title: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS

Summary:

SRP Section 12.3-12.4 references the Westinghouse STS (NUREG-0452) as they relate to radiation protection considerations. Improved STS have been promulgated for industry use (microfiche - 63886:001, 63885:001, and 63883:001).

Potential Impact Number: 23389 **SRP Section Number:** 12.3 - 12.4

APPENDIX II
Potential Impacts

Document: NUREG 1432
Title: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS

Summary:

SRP Section 12.3-12.4 references the Combustion Engineering STS (NUREG-0212) as they relate to radiation protection considerations. Improved STS have been promulgated for industry use (microfiche - 63961:271, 63965:246, and 63964:164).

Potential Impact Number: 23390 **SRP Section Number:** 12.3 - 12.4

Document: NUREG 1433
Title: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/4

Summary:

SRP Section 12.3-12.4 references the General Electric STS (NUREG-0123) as they relate to radiation protection considerations. Improved STS have been promulgated for industry use (microfiche - 63895:001, 63909:001, and 63910:001).

Potential Impact Number: 23391 **SRP Section Number:** 11.5

Document: NRC BULLETIN 80-10
Title: CONTAMINATION OF NONRADIOACTIVE SYSTEM AND RESULTING POTENTIAL FOR UNMONITORED,UNCONTROLLED RELEASE OF RADIOACTIVITY TO ENVIRONMENT

Summary:

Action to be Taken by Licensee with an Operating License; 1. Review your facility design and operation to identify systems that are considered as nonradioactive (or described as nonradioactive in the FSAR), but could possibly become radioactive through interfaces with radioactive systems, i.e., a nonradioactive system that could become contaminated due to leakage, valving errors or other operating conditions in radioactive systems. In particular, special consideration should be given to the following

systems: auxiliary boiler system, demineralized water system, isolation condenser system, PWR secondary water clean-up system, instrument air system, and the sanitary waste system.
2. Establish a routine sampling/analysis or monitoring program for these systems in order to promptly identify any contaminating events which could lead to unmonitored, uncontrolled liquid or gaseous releases to the environment, including releases to on-site leaching fields or retention ponds.

Potential Impact Number: 23394 **SRP Section Number:** 11.5

Document: ADVANCE SER ABWR CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

As stated at the beginning of Section 11 of this report, the staff will review specific compliance of the COL sampling program and QA for radiological monitoring programs with ANSI N13.1 and RGs 1.21 and 4.15 on a plant-specific basis. During such a plant-specific review, the staff will evaluate the plant-specific features and programs provided by the COL applicant to address the issue of possible contamination of nonradioactive systems and resulting potential for unmonitored, uncontrolled release to the environment discussed in IE Bulletin 80-10. Therefore, the staff's review of the sampling and analysis program for the ABWR standard design is limited to the identification of the streams required to be sampled, monitored and controlled.

Potential Impact Number: 23396 **SRP Section Number:** 11.5

Document: ADVANCE SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

IE Bulletin No. 80-10 identified an issue concerning the potential contamination of nonradioactive systems which could result in unmonitored, uncontrolled releases of radioactivity to the environment. The ABWR design employs several methods to prevent such an occurrence. The methods include ensuring that no cross-connections exist between nonradioactive and potentially radioactive systems (such as with the DTS

system), by providing sampling and monitoring of both radioactive and nonradioactive effluents before discharge, and by separating nonradioactive systems from potentially radioactive systems by barriers, along with radiation monitors to detect leakage across the barrier (such as with the RCW system). These methods are discussed in detail in various SSAR sections (e.g., 9.2.11 and 11.5) and are reviewed in various sections of this report (e.g., 9.2.11 and 11.5). These methods ensure that all discharges to the environment are monitored and controlled. Based on this information, the staff concludes that the ABWR design has provided sufficient design features and design requirements to ensure that the issues discussed in IE Bulletin No. 80-10 are resolved.

Potential Impact Number: 23399 **SRP Section Number:** 11.5

Document: ADVANCE SER CE CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

The staff finds the specific ranges for the high-range and normal range noble gas effluent monitors and containment area high-range monitor conform to the ranges required and specified for such monitors in NUREG-0737, Three Mile Island (TMI) Item II.F.1, Attachments 1 and 3.

Based on the above information, the staff considers that the DSER Confirmatory Item 11.5-1 is resolved. The staff also finds that the accident monitoring instrumentation provided in the CESSAR-DC design, for monitoring noble gases in effluents, and sampling continuously and analyzing the plant effluent for post-accident releases of radioiodine and particulates in conjunction with the COL applicant's operation and maintenance manual, prepared as discussed above, will meet the requirements of NUREG-0737, TMI Item II.F.1, Attachments 1 and 2 and, therefore, will comply with the applicable portions of 10 CFR 50.34(f)(2)(xvii) (these incorporate the subject TMI requirements).

Based on the above discussion, the staff concludes that the CESSAR-DC process and effluent radiological monitoring and sampling system complies with 10 CFR Part 20, Section 20.1302, and with GDC 60, 63, and 64. The staff further concludes that the system design complies with 10 CFR Part 50, Section 50.34(f)(2) (xvii) as it relates to noble gas effluent monitoring and sampling, and analyzing plant gaseous effluents for radioiodine and particulates during and following an accident (NUREG-0737, TMI Item

II.F.1, Attachments 1 and 2 requirements). Thus, the system meets the acceptance criteria of SRP Section 11.5 and is, therefore, acceptable.

Potential Impact Number: 23400 **SRP Section Number:** 11.5

Document: ADVANCE SER CE CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

The staff has reviewed the CESSAR-DC design regarding the issues identified in NRC Bulletin 80-10, 'Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release to the Environment.' Specifically, the staff has reviewed the CESSAR-DC design features (CESSAR-DC Sections 9.4, 11.2, 11.3, 11.5 and 12.3) 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.

The staff considers that CESSAR-DC Sections 9.4, 11.2, 11.3, 11.5 and 12.3 satisfactorily address the concerns raised in BL 80-10 and that the design features provided in CESSAR-DC design are 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.

Potential Impact Number: 23401 **SRP Section Number:** 11.5

Document: ADVANCE SER CE CH 20
Title: GENERIC ISSUES

Summary:

The staff reviewed CESSAR-DC Sections 9.4, 11.2, 11.3, 11.5, and 12.3 to determine if the System 80+ design adequately addresses the concerns identified in BL 80-10. Based on this evaluation, the staff concludes that the CESSAR-DC sections identified above satisfactorily address the concerns raised in BL 80-10 and that the design features provided in the System 80+ design are 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. Therefore, the staff determines that the System 80+ design resolves the concerns identified in BL 80-10.

Potential Impact Number: 23402 **SRP Section Number:** 11.5

Document: C&S: ANSI N13.10-74
Title: Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents

Summary:

Review procedures in SRP Section 11.2 compare an applicant's monitoring instrumentation specifications and performance criteria with those contained in ANSI N13.10-1974. Reg. Guide 4.15, cited as specific criteria in SRP Section 11.5, also cites ANSI N13.10-1974. ANSI N13.10-1974 was reaffirmed and redesignated as ANSI/IEEE N42.18-1980. In 1991, ANSI/IEEE N42.18-80 was reaffirmed.

Potential Impact Number: 23403 **SRP Section Number:** 11.5

Document: C&S: ANSI N13.1-69
Title: Guide for Sampling Airborne Radioactive Materials in Nuclear Facilities

Summary:

Specific criteria and review procedures in SRP Section 11.2 compare an applicant's provisions for gaseous sampling from ducts and stacks with ANSI N13.1-1969. Reg. Guides 4.15, 1.21, and 1,97, cited as specific criteria in SRP Section 11.5, also cite ANSI N13.1-1969. ANSI N13.1-1969 was reaffirmed in 1982 and again in 1993.

Potential Impact Number: 23405 **SRP Section Number:** 11.5

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION
Location: Part 20.106

APPENDIX II
Potential Impacts

Summary:

A licensee shall not possess, use, or transfer licensed material so as to release to an unrestricted area radioactive material in concentrations which exceed the limits specified in Appendix B, Table II of this part, except as authorized pursuant to Section 20.302 or paragraph (b) of this section.

Potential Impact Number: 23406 **SRP Section Number:** 5.4.2.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

SRP Section 5.4.2.1 does not presently cover the use of austenitic stainless steels. However, the staff reviewed the use of austenitic stainless steel in Section 5.4.2 (which covered the review of SRP Section 5.4.2.1) of this FSER. Specifically;

"As discussed in Sections 4.5.1, 4.5.2, and 5.2.3 of this report, ABB-CE limited the carbon content of austenitic stainless steels to .065 percent compared to the usual specification limit of .08 percent and imposed controls on welding, fabrication, and water chemistry during start-up and operation to mitigate the occurrence of IGSCC in these stainless steels. These mitigating actions have proven successful in operating reactors for more than 15 years and provide reasonable assurance of control of IGSCC, and therefore, the integrity of these components. On this basis, Open Item 5.4.2-10 is resolved."

Potential Impact Number: 23408 **SRP Section Number:** 12.5

Document: NUREG 1446
Title: STANDARDS FOR PROTECTION AGAINSH RADIATION - 10 CFR PART 20

Summary:

On May 21, 1991, the Nuclear Regulatory Commission issued a revision to its standards for protection against ionizing radiation, 10 CFR Part 20.

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Potential Impacts

NUREG-1446 (microfiche - 60207:182) is a comparative text of the new and old portions of 10CFR20 organized to show the full text of each set of regulations in a side-by-side format for ease of comparison. Paragraph numbers in the old and new portions of 10CFR20 do not exhibit one-for-one correspondence between the old and new paragraph content. NUREG- 1446 combines relevant text, wherever it appears, in the old portion of 10CFR20 to present it side-by-side with new paragraphs. There are instances of "No Corresponding Provision" in both directions.

Potential Impact Number: 23409 **SRP Section Number:** 5.3.1

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The FSER notes 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 recommendations of RGs 1.37, Revision 0, 1.44, Revision 0, and NUREG 0313, Revision 2.

Potential Impact Number: 23410 **SRP Section Number:** 5.3.1

Document: NUREG 0313 **Revision:** 2 , 1/1/88
Title: TECHNICAL REPORT ON MATERIAL SELECTION AND PROCESSING GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING. FINAL REPORT

Summary:

This report updates and supersedes the technical recommendations of NUREG-0313, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, published in July 1977, and its subsequent revision published in July 1980. This report provides the technical bases for the NRC staff's revised recommended methods to control the intergranular stress corrosion cracking susceptibility of BWR piping. For piping that does not fully comply with the material selection, testing, and processing guideline combinations of this document, varying degrees of augmented inservice inspection are recommended. This revision also

includes guidance and NRC staff recommendations (not requirements) regarding crack evaluation and weld overlay repair methods for long-term operation or for continuing interim operation of plants until a more permanent solution is implemented. (Microfiche location 44241:127)

Potential Impact Number: 23411 **SRP Section Number:** 5.3.1

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

The staff has recommended that licensees and applicants follow Revision 2 of NUREG-0313, Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping, to prevent IGSCC in stainless steel. In addition, it is important that adequate field and shop fabrication processes be used to minimize the sensitization of materials to IGSCC. Therefore, the staff will evaluate this matter during its review of an individual application for a combined license.

Potential Impact Number: 23412 **SRP Section Number:** 4.5.1

Document: NUREG 0313 **Revision:** 2 , 1/1/88
Title: TECHNICAL REPORT ON MATERIAL SELECTION AND PROCESSING GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING. FINAL REPORT

Summary:

This report updates and supersedes the technical recommendations of NUREG-0313, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, published in July 1977, and its subsequent revision published in July 1980. This report provides the technical bases for the NRC staff's revised recommended methods to control the intergranular stress corrosion cracking susceptibility of BWR piping. For piping that does not fully comply with the material selection, testing, and processing guideline combinations of this document, varying degrees of augmented inservice inspection are recommended. This revision also includes guidance and NRC staff recommendations (not requirements) regarding crack

evaluation and weld overlay repair methods for long-term operation or for continuing interim operation of plants until a more permanent solution is implemented. Microfiche No: 44241:127

Potential Impact Number: 23413 **SRP Section Number:** 4.5.2

Document: NUREG 0313 **Revision:** 2 , 1/1/88
Title: TECHNICAL REPORT ON MATERIAL SELECTION AND PROCESSING GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING. FINAL REPORT

Summary:

This report updates and supersedes the technical recommendations of NUREG-0313, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, published in July 1977, and its subsequent revision published in July 1980. This report provides the technical bases for the NRC staff's revised recommended methods to control the intergranular stress corrosion cracking susceptibility of BWR piping. For piping that does not fully comply with the material selection, testing, and processing guideline combinations of this document, varying degrees of augmented inservice inspection are recommended. This revision also includes guidance and NRC staff recommendations (not requirements) regarding crack evaluation and weld overlay repair methods for long-term operation or for continuing interim operation of plants until a more permanent solution is implemented. Microfiche No: 44241:127

Potential Impact Number: 23414 **SRP Section Number:** 5.2.3

Document: NUREG 0313 **Revision:** 2 , 1/1/88
Title: TECHNICAL REPORT ON MATERIAL SELECTION AND PROCESSING GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING. FINAL REPORT

Summary:

This report updates and supersedes the technical recommendations of NUREG-0313, Technical Report on Material Selection and Processing Guidelines for BWR Coolant

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Pressure Boundary Piping, published in July 1977, and its subsequent revision published in July 1980. This report provides the technical bases for the NRC staff's revised recommended methods to control the intergranular stress corrosion cracking susceptibility of BWR piping. For piping that does not fully comply with the material selection, testing, and processing guideline combinations of this document, varying degrees of augmented inservice inspection are recommended. This revision also includes guidance and NRC staff recommendations (not requirements) regarding crack evaluation and weld overlay repair methods for long-term operation or for continuing interim operation of plants until a more permanent solution is implemented.

Analyst Notes:

Reg. Guide 1.31 position C.4 specifies ferrite content limits between 5 and 20% (Ferrite Numbers from 5 to 20) for stainless steel weld metal filler materials. NUREG-0313, Revision 2 identifies a lower ferrite content limit of 7.5% which is more restrictive than the lower limit specified in Reg. Guide 1.31.

NUREG-0313, Revision 2 guidelines for prevention of IGSCC in BWR RCPB piping are in many cases different (including stated carbon and ferrite content limits) than the current guidance of Reg. Guide 1.44 for control of the use of sensitized stainless steel. NUREG-0313, Revision 2 guidelines are in some cases more restrictive and/or more comprehensive (provide more initial material selection options and more post weld treatment/repair options to reduce susceptibility to IGSCC) than Reg. Guide 1.44 guidance.

Microfiche No: 44241:127

Potential Impact Number: 23415 **SRP Section Number:** 5.4.2.1

Document: NUREG 0313 **Revision:** 2 , 1/1/88
Title: TECHNICAL REPORT ON MATERIAL SELECTION AND PROCESSING GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING. FINAL REPORT

Summary:

This report updates and supersedes the technical recommendations of NUREG-0313, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, published in July 1977, and its subsequent revision

published in July 1980. This report provides the technical bases for the NRC staff's revised recommended methods to control the intergranular stress corrosion cracking susceptibility of BWR piping. For piping that does not fully comply with the material selection, testing, and processing guideline combinations of this document, varying degrees of augmented inservice inspection are recommended. This revision also includes guidance and NRC staff recommendations (not requirements) regarding crack evaluation and weld overlay repair methods for long-term operation or for continuing interim operation of plants until a more permanent solution is implemented. Microfiche No: 44241:127

Potential Impact Number: 23416 **SRP Section Number:** 10.3.6

Document: NUREG 0313 **Revision:** 2 , 1/1/88
Title: TECHNICAL REPORT ON MATERIAL SELECTION AND PROCESSING GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING. FINAL REPORT

Summary:

This report updates and supersedes the technical recommendations of NUREG-0313, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, published in July 1977, and its subsequent revision published in July 1980. This report provides the technical bases for the NRC staff's revised recommended methods to control the intergranular stress corrosion cracking susceptibility of BWR piping. For piping that does not fully comply with the material selection, testing, and processing guideline combinations of this document, varying degrees of augmented inservice inspection are recommended. This revision also includes guidance and NRC staff recommendations (not requirements) regarding crack evaluation and weld overlay repair methods for long-term operation or for continuing interim operation of plants until a more permanent solution is implemented. Microfiche No: 44241:127

Potential Impact Number: 23417 **SRP Section Number:** 4.2

Document: C&S: ASTM C776-77
Title: Specification for Sintered Uranium Dioxide Pellets

APPENDIX II
Potential Impacts

Summary:

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.

Potential Impact Number: 23418 **SRP Section Number:** 4.2

Document: FINAL SER ABWR CH 4
Title: REACTOR

Summary:

The staff considers the fuel burnup limit a safety question and has several fuel operating concerns at burnup levels above those currently approved for BWRs. 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 in the ABWR and may be extended only with NRC review and approval. This was identified as DFSER Confirmatory Item 4.2-1.

Potential Impact Number: 23419 **SRP Section Number:** 4.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Section 3.6 of Chapter 1 of the Evolutionary Requirements Document requires that the ALWR core be designed for up to a 24-month fuel cycle and that fuel mechanical designs have a peak bundle burnup of at least 45,000 and 55,000 megawatt-days per metric ton of uranium (MWD/MTU) for BWRs and PWRs, respectively. These minimum fuel burnup requirements are inconsistent with the EPRI-proposed requirements of 50,000 and 60,000 MWD/MTU specified in Sections 4.2.2.2 and 7.2.2.2 of Chapter 4 for BWRs and PWRs, respectively. In addition, although these values are inconsistent with each other, they are greater than NRC-approved fuel burnup levels. To support this high fuel burnup operation, each ALWR design application will need to include

sufficient high fuel burnup data to demonstrate fuel integrity in the areas of fission gas release, cladding corrosion due to oxidation and hydriding, and reduction in cladding material strength.

Potential Impact Number: 23420 **SRP Section Number:** 4.5.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The staff considers Alloy 600 undesirable for use in the ALWR because of the stress corrosion cracking experienced in existing nuclear plants. It specifically discourages the use of Alloy 600 in steam generators and pressurizers. Therefore, it will require that the applicant for any standard design application identify the use of Alloy 600 and provide information concerning its use. Those applications will be reviewed and approved by the staff on a case-by-case basis. In addition, the use of other Ni-Cr-Fe alloys such as Alloy 690 or 800 should be considered in applications for which primary water stress corrosion cracking is a concern. These applications also will be reviewed on a case-by-case basis.

Also, where Iconel Alloy-X-750 is used, the application will be in accordance with EPRI NP-6202, 'Material specification for Alloy-X-750 in LWR Internal Components. The staff considers these requirements acceptable because improperly heat treated Alloy-X-750 is susceptible to cracking and EPRI NP- 6202 provides the technical basis for the use of this alloy.

Potential Impact Number: 23421 **SRP Section Number:** 5.2.3

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The staff considers Alloy 600 undesirable for use in the ALWR because of the stress corrosion cracking experienced in existing nuclear plants. It specifically discourages the use of Alloy 600 in steam generators and pressurizers. Therefore, it will require that the

applicant for any standard design application identify the use of Alloy 600 and provide information concerning its use. Those applications will be reviewed and approved by the staff on a case-by-case basis. In addition, the use of other Ni-Cr-Fe alloys such as Alloy 690 or 800 should be considered in applications for which primary water stress corrosion cracking is a concern. These applications also will be reviewed on a case-by-case basis.

Also, where Iconel Alloy-X-750 is used, the application will be in accordance with EPRI NP-6202, 'Material specification for Alloy-X-750 in LWR Internal Components. The staff considers these requirements acceptable because improperly heat treated Alloy-X-750 is susceptible to cracking and EPRI NP- 6202 provides the technical basis for the use of this alloy.

Potential Impact Number: 23422 **SRP Section Number:** 5.2.3

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In Amendment Q to CESSAR-DC Table 5.2-2, ABB-CE listed materials and information for RCPB components and included Inconel 690 materials in place of Inconel 600 materials. Operating experience indicates that Inconel 600 and Inconel 182 (shielded metal arc welding filler material) are susceptible to cracking. ABB-CE considered alternate materials that were more resistant to cracking. ABB-CE will use Inconel 690 for all applications in which Inconel 600 was to be used. The staff views the Inconel 690 alloy as the preferred nickel alloy in the primary and secondary coolant loops because of its improved corrosion resistance compared to Inconel 600. The use of Inconel 690 will provide reasonable assurance of the material integrity of the components and tubing in contact with reactor coolant and most secondary water chemistries. On this basis, DSER Open Item 5.2.3-3 is resolved.

Potential Impact Number: 23423 **SRP Section Number:** 5.4.2.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

APPENDIX II
Potential Impacts

Summary:

In Section 5.3.A.3.b of the DSER for Chapter 1, the staff identified a concern regarding the use of Alloy 600 in future ALWRs. Section 5.3.1.3.1 of Chapter 1 of the Evolutionary Requirements Document specifies that Alloy 600 can be used in ALWRs. EPRI has revised Section 5.3.1.3 to place restrictions on the use of Alloy 600. EPRI states that the designer will not use Alloy 600 in the steam generators and pressurizers. For PWR applications, Alloy 600 will be given a special thermal treatment to improve resistance to stress corrosion cracking. However, in general, the staff considers Alloy 600 undesirable for use in the ALWR because of the stress corrosion cracking experienced in existing nuclear plants. It specifically discourages the use of Alloy 600 in steam generators and pressurizers. Therefore, it will require that the applicant for any standard design application identify the use of Alloy 600 and provide information concerning its use. Those applications will be reviewed and approved by the staff on a case-by-case basis. In addition, the use of other Ni-Cr-Fe alloys such as Alloy 690 or 800 should be considered in applications for which primary water stress corrosion cracking is a concern. The implementation of this requirement will ensure that Ni-Cr-Fe alloys will perform in service as designed. These applications also will be reviewed on a case-by-case basis. In addition, the EPRI Evolutionary Plant FSER states that pickling of wetted surfaces for all Ni-Cr-Fe alloys is prohibited for both BWRs and PWRs. Section 5.3.1.7 of Chapter 1 specifies that the application of Inconel Alloy X-750 will be in accordance with EPRI NP-6202. 'Material Specification for Alloy X-750 in LWR Internal Components.'

Potential Impact Number: 23424 **SRP Section Number:** 5.4.2.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In Section 5.4.2 (review of SRP Section 5.4.2.1 items) the staff noted the steam generator tubes are fabricated of thermally treated Inconel 690 materials. In Section 5.2.3 the staff noted in Amendment Q to CESSAR-DC Table 5.2-2, ABB-CE listed materials and information for RCPB components and included Inconel 690 materials in lieu of Inconel 600 materials. Operating experience indicates that Inconel 600 and Inconel 182 (shielded metal arc welding filler material) are susceptible to cracking. ABB-CE considered alternate materials that were more resistant to cracking. ABB-CE will use Inconel 690 for all applications where Inconel 600 was to be used. 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. The use of Inconel 690 will provide reasonable assurance of the material integrity of the components and tubing in contact with reactor coolant and most secondary water chemistries. On this basis, Open Item 5.2.3-3 is resolved.

Potential Impact Number: 23428 **SRP Section Number:** 4.5.1

Document: C&S: ASME NQA-2
Title: QUALITY ASSURANCE REQUIREMENTS FOR NUCLEAR POWER PLANTS

Summary:

The CE 80+ ASER (Chapter 17) states that NQA-2 has superseded the ANSI standards referenced by RGs 1.30. 1.37, 1.38, 1.39, 1.94, and 1.116.

Potential Impact Number: 23432 **SRP Section Number:** 5.3.1

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

Cleaning and cleanliness control are in accordance with ANSI/ASME NQA-2-1983, Quality Assurance Requirements for Nuclear Power Plants, and RG 1.37, Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants. The staff reviewed ANSI/ASME NQA-2-1983 and found it acceptable. (Also see CE 80+ FSER Ch. 17 for staff position that NQA-2-1983 supercedes standards referenced by several Reg. Guides, including Reg. Guide 1.37)

Potential Impact Number: 23433 **SRP Section Number:** 5.3.1

Document: C&S: ANSI N45.2.1
Title: Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants

APPENDIX II
Potential Impacts

Summary:

The ANSI N 45.2.1-1973 Standard is referenced in Section 4.5.1 of the Standard Review Plan (1981) for onsite cleaning and cleanliness control. Continued use of ANSI N 45.2.1-1973 would require consideration of reviewing the updated version, ANSI N 45.1-1-1980, for consistency. ANSI N 45.2.1 is not included in Table 6 for a code comparison by PNL.

Potential Impact Number: 23434 **SRP Section Number:** 5.3.1

Document: C&S: ASME NQA-2
Title: QUALITY ASSURANCE REQUIREMENTS FOR NUCLEAR POWER PLANTS

Summary:

The CE 80+ FSER (Chapter 17) states that NQA-2 has superseded the ANSI standards referenced by RGs 1.30. 1.37, 1.38, 1.39, 1.94, and 1.116.

Potential Impact Number: 23435 **SRP Section Number:** 5.4.2.1

Document: C&S: ASME NQA-2
Title: QUALITY ASSURANCE REQUIREMENTS FOR NUCLEAR POWER PLANTS

Summary:

The CE 80+ FSER (Chapter 17) states that NQA-2 has superseded the ANSI standards referenced by RGs 1.30. 1.37, 1.38, 1.39, 1.94, and 1.116.

Potential Impact Number: 23436 **SRP Section Number:** 5.4.2.1

Document: C&S: ANSI N45.2.1
Title: Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants

APPENDIX II
Potential Impacts

Summary:

The ANSI N 45.2.1-1973 Standard is referenced in Section 4.5.1 of the Standard Review Plan (1981) for onsite cleaning and cleanliness control. Continued use of ANSI N 45.2.1-1973 would require consideration of reviewing the updated version, ANSI N 45.2-1-1980, for consistency.

Potential Impact Number: 23437 **SRP Section Number:** 5.4.2.1

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

Cleaning and cleanliness control are in accordance with ANSI/ASME NQA-2-1983, Quality Assurance Requirements for Nuclear Power Plants, and RG 1.37, Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants. The staff reviewed ANSI/ASME NQA-2-1983 and found it acceptable. (Also see CE 80+ FSER Ch. 17 for staff position that NQA-2-1983 supercedes standards referenced by several Reg. Guides, including Reg. Guide 1.37)

Potential Impact Number: 23440 **SRP Section Number:** 10.3.6

Document: C&S: ANSI N45.2.1
Title: Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants

Summary:

The ANSI N 45.2.1-1973 Standard is referenced in Section 4.5.1 of the Standard Review Plan (1981) for onsite cleaning and cleanliness control. Continued use of ANSI N 45.2.1-1973 would require consideration of reviewing the updated version, ANSI N 45.1-1-1980, for consistency.

Potential Impact Number: 23447 **SRP Section Number:** 5.3.1

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Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The radiation induced shifts in reference temperatures for Reactor Vessel beltline materials can be predicted with reasonable accuracy and conservatism using the methodology of RG 1.99, Revision 2.

Potential Impact Number: 23448 **SRP Section Number:** 5.2.3

Document: C&S: ASTM A262
Title: DETECTING SUSCEPTIBILITY TO INTERGRANULAR ATTACK IN STAINLESS STEEL

Summary:

ASTM A262 is cited as a reference in SRP Section 5.2.3. RGs 1.37 and 1.44 are both cited by SRP Section 5.2.3 for the control of cracking in stainless steel. RG 1.44 endorses (with conditions) the 1970 version, while RG 1.37 endorses the 1968 version. The citation in the SRP is not date specific. The current version of this standard is ASTM A262-1991.

Potential Impact Number: 23449 **SRP Section Number:** 5.2.3

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In the ASER, the staff noted that ABB-CE proposed to use American Society for Testing and Materials (ASTM) A-708, in lieu of the ASTM A-262 procedures recommended in RG 1.44, "Control of the Use of Sensitized Stainless Steel," for verifying non-sensitization of austenitic stainless steel materials. The staff finds the proposed alternative (ASTM A-708) a more stringent test than ASTM A-262, Practice E. The ASTM A-708 test uses the same test solution and same temperature as ASTM A-262,

Practice E; however, A-708 requires 72 hours in the test solution while A-262, Practice E requires only 24 hours (with an option for longer test times upon agreement between supplier and user). Also, Note 26 of A-262, Practice E acknowledges that results in the literature indicate that Practice E is more sensitive when run for longer times. Accordingly, as acknowledged in SRP 5.4.1, paragraph III.2, ASTM A-708 is an acceptable alternative test for ASTM A-262, Practice A or E. On this basis, Open Item 5.2.3-4 was resolved.

In the FSER, the staff clarified that ABB-CE 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 (recommended in RG 1.44), "Control of the Use of Sensitized Stainless Steel," has been used by ABB-CE since the mid-1970s for verifying non-sensitization of austenitic stainless steel materials and weldments. ABB-CE 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 ABB-CE'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 tests for ASTM A-262, Practice A or E. On this basis, DSER Open item 5.2.3-4 is resolved.

Potential Impact Number: 23451 **SRP Section Number:** 5.3.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

Appendix H to 10 CFR Part 50 requires the surveillance program to meet ASTM E185-82. ABB-CE contends that its surveillance program is designed on the basis of Appendix H to 10 CFR Part 50 and ASTM E185-82. ASTM E185-82 has been applicable to plants designed for 40 years, i.e., 32 effective full-power years (EFPYs) at end-of-life. ABB-CE submitted additional information on its surveillance program in response to Q252.1.

The staff required that the schedule in ASTM E185-82 be supplemented to address the period between 40 and 60 years. The staff, in the DSER, required that ABB-CE propose a capsule withdrawal schedule beyond 32 EFPYs to demonstrate compliance with 10 CFR Part 50, Appendix H, to the end of the plant's design life of 60 years.

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In Amendment L to the CESSAR-DC, ABB-CE revised the capsule withdrawal schedule to withdraw a surveillance capsule at 6, 15, 32 and 48 (EOL) EFPY over the 60 year design life of the reactor vessel. This revision to the withdrawal schedule is consistent with the discussion provided for this open item, with the exception that a surveillance capsule will not be withdrawn at 3 EFPY. The predicted shift for the beltline forging material to be used for the System 80+ reactor vessel is predicted to have approximately a 8 C (15 F) shift at a fluence corresponding to 3 EFPY. The predicted shift after 6 EFPY for the beltline forging is only 10 C (18 F). The material selection and requirements for the System 80+ reactor vessel beltline will ensure minimal damage due to irradiation and minimal changes in properties. The beltline materials in the reactor pressure vessels of this design will have tighter chemistry limits and more stringent requirements for initial properties than the latest fabricated vessels of currently operating, licensed reactor vessels. The staff agrees that there is not a sufficient difference in predicted shift over this short time span relative to the overall life of the reactor vessel to justify the need for two surveillance capsules being analyzed in the first 6 EFPYs. On this basis, Open Item 5.3.1-3 is resolved

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Potential Impact Number: 23452 **SRP Section Number:** 5.3.1

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The surveillance program must generate sufficient information to permit the determination of conditions under which the reactor vessel will be operated with an adequate margin against rapidly propagating fracture throughout its service lifetime. Based on a 40-year design life, ASTM E-185-82 recommends three materials surveillance capsules be installed in the reactor vessel beltline. However, in the DFSER, the staff identified that for the ABWR, the design life is expected to be increased to 60 years. Accordingly, a need exists for GE to reassess the number of materials surveillance capsules to be provided to account for the additional 20-year increase in the expected life of the vessel. The staff noted that GE should revise its SSAR to address changes to the material surveillance program, including the number and location of the vessel surveillance capsules, for the ABWR reactor vessel to account for the 60-year design life of the plant. This was DFSER Open Item 5.3.1-1.

GE addressed this open item in the SSAR. Specifically, GE committed to provide four surveillance capsules instead of previously proposed three capsules. GE also required

that the capsules placement be designed to produce a lead factor of approximately 1.2 to 1.5. Furthermore, GE specified that the weld metal specimens be made from the same heat of weld wire, lot of flux and using the same welding practice as used for the beltline weld. GE also calculated that the predicted end of license adjusted reference temperature of the reactor vessel will be less than 38 degrees C (100 degrees F). The staff finds this response acceptable because these requirements will ensure that the surveillance program will generate sufficient information to determine the conditions under which the reactor vessel will be operated throughout its 60 years service lifetime.

Potential Impact Number: 23463 **SRP Section Number:** 5.3.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," (56 FR 22300; May 15, 1991) establishes a PTS screening criterion (RTPTS) below which no additional action is required for protection from PTS events.

In response to Q252.13, ABB-CE stated that the design meets the PTS screening criterion. The ABB-CE assessment is based on the assumed beltline forging material properties (0.06 weight-percent copper, 1.00 weight-percent nickel, and -12 C (10 F) initial RTNDT) and 6 x 1019 n/cm2 end-of-life fluence (60 years).

The reactor vessel beltline materials are acceptable and meet the requirements of 10 CFR 50.61. DSER Open Item 5.2.2.3-1 is resolved as follows: In Amendment N, ABB-CE modified the CESSAR-DC to state that the pressurized thermal shock calculation margin used was 27 C (50F) for base metal. The staff has performed a statistical analysis of all the plate surveillance data that were received from the licensees. The surveillance data are maintained in the PR-EDB. Surveillance data received by the NRC from the licensees after the issuance of RG 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," Revision 2, indicate that the standard deviation for the increase in reference temperature for the plate material should be 13 C (26 F). This would result in a margin of 28.9 C (52 F). The staff has raised this issue in the PTS review of other plants, such as Indian Point 3 and Beaver Valley 1.

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Since the staff's statistical analysis indicates that the standard deviation for the increase in reference temperature is greater than 9.5 C (17 F), the licensee's use of 9.5 C (17 F) as the amount of margin may not assure that the RTPTS value is less than the PTS screening criteria at the end of a license. A staff contractor is currently reviewing the data base to update the methodology for calculating RTPTS. When this review is completed, a margin term different than 9.5 C (17 F) may be necessary to assure that the RTPTS value is less than the PTS screening criterion at the end of a license.

The staff believes that ABB-CE's approach of using a margin of 27.8 C (50 F) rather than 28.9 C (52 F) has enough conservatism to provide adequate assurance of fracture toughness in a PTS event for its design life. Information is being developed to improve the accuracy of calculating RTPTS. This issue is further addressed in the resolution of Open Item 5.3.2-3. On this basis, Open Item 5.2.2.3-1 is resolved.

Potential Impact Number: 23464 **SRP Section Number:** 4.3

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

Maximum fast neutron fluxes greater than 1 million electron volt (Mev) at the vessel inside diameter are presented. For reactor operation at the full power rating and an 80 percent capacity factor, the calculated vessel fluence greater than 1 Mev at the inner vessel wall does not exceed 6.2 x 1019 n/cm2 over the 60-year design life of the vessel. ABB-CE had to provide justification that the calculated fluence complies with the requirements of 10 CFR 50.61, this was designated as DSER Open Item 4.3.2-2. The calculated exposure includes a 30-percent uncertainty factor. The staff reviewed ABB-CE's calculational method and found that calculated values of RTPTS for the maximum value of the fluence and the appropriate base metal and weld chemistry comply with the requirements of 10 CFR 50.61. On this basis, DSER Open Item 4.3.2-2 is resolved.

Potential Impact Number: 23466 **SRP Section Number:** 5.3.2

Document: DRAFT SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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Summary:

The applicant states that the pressure-temperature limit curves will be developed based on Appendix G to ASME Section III requirements and the embrittlement estimation in RG 1.99, Revision 2. However, there are uncertainties in neutron embrittlement prediction procedures. For example, RG 1.99, Revision 1, would predict a shift of 70 C (126 F) in the reference temperature based on the phosphorous content, which is not addressed in RG 1.99, Revision 2. Thus, for establishing the reactor vessel pressure-temperature limits before valid plant-specific surveillance data become available, the applicant should estimate the shift in the reference temperature as the larger of: (a) that predicted by RG 1.99, Rev. 2; or, (b) a calculation accounting for the phosphorous content (an acceptable formula is specified in the DSER).

Potential Impact Number: 23467 **SRP Section Number:** 5.3.2

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

As discussed in the resolution of Open Item 5.3.1-1, the phosphorus content in the reactor vessel forgings and weld metal is controlled to a maximum of 0.012 percent. The possible effect of phosphorus on predicted shifts in low copper RPV beltline materials was evaluated by ORNL and the NRC using the EDB database. Only a marginal correlation was observed between phosphorus content and predicted shift for low copper materials. The conclusion that there was any significant correlation was judged to be weak. Even in those cases where a possible trend with phosphorus content was suggested by the data, the differences between the observed and predicted shifts were within the margins applied by RG 1.99, Revision 2. In addition, in the cases where a trend with phosphorus was suggested, it was generally the high phosphorus (>0.012 percent) materials which indicated the possible trend. Based on this previous assessment of the effects of impurity elements on the irradiation response of materials, the existing controls on residual elements are concluded to be sufficient. The limits on copper, phosphorus and other residual elements will minimize the extent of radiation damage to the RPV beltline materials. 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. On this basis, Open Item 5.3.2-1 is resolved.

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Potential Impact Number: 23468 **SRP Section Number:** 5.3.2

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

Appendices G and H to 10 CFR Part 50 require the applicant to predict the amount of increase in reference temperature, RTNDT, resulting from neutron irradiation. The increase in RTNDT resulting from neutron irradiation is then added to the initial RTNDT and the margin to establish the adjusted reference temperature. The staff's recommended method for calculating the increase in RTNDT resulting from neutron irradiation is contained in RG 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," Revision 2. The relationships contained in the guide were derived by statistical analysis of 216 material data points that were reported from the testing of irradiated materials.

For approval of any specific plant license, COL applicants must submit plant-specific calculations of RTNDT, stress intensity factors, and pressure- temperature curves similar to those in RG 1.99, Revision 2, and SRP Section 5.3.2. This is COL Action Item 5.3.2-1.

Potential Impact Number: 23470 **SRP Section Number:** 4.5.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

In Section 5.3.1.1 of Chapter 1 of the EPRI Evolutionary Plant FSER the staff concluded the following manual grinding controls will ensure that wrought austenitic stainless steel will perform in service as design:

"- Grinding performed before solution heat treatment requires no additional control except that after heavy grinding, light grinding or flapping will be required.

- Grinding will be restricted except for fit-up, nondestructive testing, or to remove scratches or surface defects.

- All grinding of austenitic stainless steel surfaces exposed to reactor water in service that are not subsequently solution heat treated will be performed in accordance with a written procedure. The procedure will involve finishing the ground surface with successively finer grit sizes to remove the bulk of cold-worked material, with the final grit size no coarser than #120 grit."

The above requirements were originally required for BWRs, but only recommended for PWRs. In their evaluation the staff required these controls be applied to PWRs.

Potential Impact Number: 23471 **SRP Section Number:** 4.5.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

In Section 5.3.1.1 of Chapter 1 of the EPRI Evolutionary Plant FSER the staff concluded the following manual grinding controls will ensure that wrought austenitic stainless steel will perform in service as design:

"- Grinding performed before solution heat treatment requires no additional control except that after heavy grinding, light grinding or flapping will be required.

- Grinding will be restricted except for fit-up, nondestructive testing, or to remove scratches or surface defects.

- All grinding of austenitic stainless steel surfaces exposed to reactor water in service that are not subsequently solution heat treated will be performed in accordance with a written procedure. The procedure will involve finishing the ground surface with successively finer grit sizes to remove the bulk of cold-worked material, with the final grit size no coarser than #120 grit."

The above requirements were originally required for BWRs, but only recommended for PWRs. In their evaluation the staff required these controls be applied to PWRs.

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Potential Impact Number: 23472 **SRP Section Number:** 5.2.3

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

In Section 5.3.1.1 of Chapter 1 of the EPRI Evolutionary Plant FSER the staff concluded the following manual grinding controls will ensure that wrought austenitic stainless steel will perform in service as design:

"- Grinding performed before solution heat treatment requires no additional control except that after heavy grinding, light grinding or flapping will be required.

- Grinding will be restricted except for fit-up, nondestructive testing, or to remove scratches or surface defects.

- All grinding of austenitic stainless steel surfaces exposed to reactor water in service that are not subsequently solution heat treated will be performed in accordance with a written procedure. The procedure will involve finishing the ground surface with successively finer grit sizes to remove the bulk of cold-worked material, with the final grit size no coarser than #120 grit."

The above requirements were originally required for BWRs, but only recommended for PWRs. In their evaluation the staff required these controls be applied to PWRs.

Potential Impact Number: 23473 **SRP Section Number:** 5.3.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

In Section 5.3.1.1 of Chapter 1 of the EPRI Evolutionary Plant FSER the staff concluded the following manual grinding controls will ensure that wrought austenitic stainless steel will perform in service as design:

"- Grinding performed before solution heat treatment requires no additional control except that after heavy grinding, light grinding or flapping will be required.

- Grinding will be restricted except for fit-up, nondestructive testing, or to remove scratches or surface defects.

- All grinding of austenitic stainless steel surfaces exposed to reactor water in service that are not subsequently solution heat treated will be performed in accordance with a written procedure. The procedure will involve finishing the ground surface with successively finer grit sizes to remove the bulk of cold-worked material, with the final grit size no coarser than #120 grit."

The above requirements were originally required for BWRs, but only recommended for PWRs. In their evaluation the staff required these controls be applied to PWRs.

Potential Impact Number: 23474 **SRP Section Number:** 6.1.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

In Section 5.3.1.1 of Chapter 1 of the EPRI Evolutionary Plant FSER the staff concluded the following manual grinding controls will ensure that wrought austenitic stainless steel will perform in service as design:

"- Grinding performed before solution heat treatment requires no additional control except that after heavy grinding, light grinding or flapping will be required.

- Grinding will be restricted except for fit-up, nondestructive testing, or to remove scratches or surface defects.

- All grinding of austenitic stainless steel surfaces exposed to reactor water in service that are not subsequently solution heat treated will be performed in accordance with a written procedure. The procedure will involve finishing the ground surface with successively finer grit sizes to remove the bulk of cold-worked material, with the final grit size no coarser than #120 grit."

The above requirements were originally required for BWRs, but only recommended for PWRs. In their evaluation the staff required these controls be applied to PWRs.

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Potential Impact Number: 23475 **SRP Section Number:** 10.3.6

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

In Section 5.3.1.1 of Chapter 1 of the EPRI Evolutionary Plant FSER the staff concluded the following manual grinding controls will ensure that wrought austenitic stainless steel will perform in service as design:

"- Grinding performed before solution heat treatment requires no additional control except that after heavy grinding, light grinding or flapping will be required.

- Grinding will be restricted except for fit-up, nondestructive testing, or to remove scratches or surface defects.

- All grinding of austenitic stainless steel surfaces exposed to reactor water in service that are not subsequently solution heat treated will be performed in accordance with a written procedure. The procedure will involve finishing the ground surface with successively finer grit sizes to remove the bulk of cold-worked material, with the final grit size no coarser than #120 grit."

The above requirements were originally required for BWRs, but only recommended for PWRs. In their evaluation the staff required these controls be applied to PWRs.

Potential Impact Number: 23477 **SRP Section Number:** 5.4.2.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

ABB-CE has provided a corrosion allowance for a 60-year plant design life and its technical basis as follows: The 60-year minimum corrosion allowance for the materials to exposed process fluids are:

- 0.00015 cm (0.0006 in.) for Inconel 690 - 0.0076 cm (0.003 in.) for austenitic stainless steel - 0.015 cm (0.006 in.) for ferritic stainless steel - 0.15 cm (0.060 in.) for unclad carbon steel

....(which)... will provide sufficient material for 60 years of operation. On this basis, DSER Open Item 5.4.2-7 is resolved.

Potential Impact Number: 23478 **SRP Section Number:** 10.3.6

Document: FINAL SER CE80 CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

ABB-CE also provides in CESSAR-DC the corrosion allowance value for a 60-year plant design life and its technical basis for the determination of that value. Carbon steel is to be used for the main steam and main feedwater systems with a minimum corrosion allowance of 2.5 mm (.1 in.) per year. This corrosion allowance is greater than that used in the design of most current plants and should account for the 60-year design life of a System 80+ plant. For piping sections that are more susceptible to erosion/corrosion degradation, chromium-molybdenum or stainless steel are used per CESSAR-DC, Section 10.3.6.2.G.5. On this basis, DSER Open Item 10.3-6 is resolved.

ABB-CE identified in the CESSAR-DC the specific materials for the steam and feedwater system. Specifically, ABB-CE stated that carbon-manganese and chromium-molybdenum steels are to be used for the main steam and main feedwater systems. Specific grades are as follows:

Main Steam

ASME Class 2 Portion SA 672 Gr. C70, C1. 22 (>24" NPS) SA 106 Gr. B or C (>24" NPS)

B31.1 Portion A 672 Gr. C70, C1. 22 (>24" NPS) A 106 Gr. B or C (24" NPS)

Main Feedwater

ASME Class 2 Portion SA 106 Gr. B or C (24" NPS)

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B31.1 Portion A 672 Gr. C70, C1. 22 or A 106 Gr. C(>24" NPS) A 106 Gr. B or C (24" NPS)

Potential Impact Number: 23479 **SRP Section Number:** 5.4.2.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In the DSER, the staff maintained that the applicant should provide a corrosion allowance for a 60-year plant design life and its technical basis (Open Item 5.4.2-7). In the FSER, the staff verified that acceptable corrosion allowances covering the design life were specified.

Potential Impact Number: 23480 **SRP Section Number:** 10.3.6

Document: DRAFT SER CE80 CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

The applicant should provide a corrosion allowance for a 60-year plant design life and its technical basis (Open Item 10.3-6).

Potential Impact Number: 23481 **SRP Section Number:** 5.4.2.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The staff considers previously accepted standard corrosion allowances to be inadequate. The plant designer should consider all types of corrosion (i.e.,

microbiological, pitting, and crevice) in the design. The staff concluded that the plant designer should control corrosion mechanisms, such as general, pitting, crevice, and microbiological corrosion, in the design of piping systems of primary and secondary systems. The general corrosion allowance should comply with the corrosion allowance specified in Section III of the ASME Code and ANSI/ASME B.31.1, Power Piping. For the specific corrosion allowance (such as that for microbiological corrosion), the plant designer may use industry methodologies such as the EPRI computer code CHECMATE. However, the staff will have to review and approve any industry methodology before it is used in the ALWR design. The staff will evaluate this matter during its review of an individual application for FDA/DC.

Potential Impact Number: 23482 **SRP Section Number:** 10.3.6

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The staff considers previously accepted standard corrosion allowances to be inadequate. The plant designer should consider all types of corrosion (i.e., microbiological, pitting, and crevice) in the design. The staff concluded that the plant designer should control corrosion mechanisms, such as general, pitting, crevice, and microbiological corrosion, in the design of piping systems of primary and secondary systems. The general corrosion allowance should comply with the corrosion allowance specified in Section III of the ASME Code and ANSI/ASME B.31.1, Power Piping. For the specific corrosion allowance (such as that for microbiological corrosion), the plant designer may use industry methodologies such as the EPRI computer code CHECMATE. However, the staff will have to review and approve any industry methodology before it is used in the ALWR design. The staff will evaluate this matter during its review of an individual application for FDA/DC.

Potential Impact Number: 23483 **SRP Section Number:** 4.5.2

Document: C&S: ASTM A262
Title: DETECTING SUSCEPTIBILITY TO INTERGRANULAR ATTACK IN STAINLESS STEEL

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Summary:

Several other SRP Material sections (4.5.1, 5.2.3, 6.1.1) cite ASTM A-262 as endorsed by RG 1.44 for verification of non-sensitization of stainless steel and qualification of welding process employed in production. ASTM A-262 is also endorsed by RG 1.37. This SRP Section cites RG 1.44, and a change has been proposed to cite RG 1.37.

Potential Impact Number: 23492 **SRP Section Number:** 4.4

Document: NRC LETTER From A.C. THADANI to L.A. ENGLAND
Title: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORTS NEDO-31960 AND NEDO-31960 SUPPLEMENT 1, "BWR OWNERS' GROUP LONG-TERM STABILITY SOLUTIONS LICENSING METHODOLOGY" (TAC NO. W75928)

Summary:

The letter documents staff acceptance of Topical Reports, prepared by GE for the BWR Owners Group, as acceptable for referencing in license applications to address long-term resolution of BWR thermal-hydraulic instability concerns.

Potential Impact Number: 23493 **SRP Section Number:** 4.4

Document: FINAL SER ABWR CH 4
Title: REACTOR

Summary:

The NRC, in parallel with work by the BWR Owners' Group (BWROG) stability sub-committee, has for some time been reviewing generic questions relating to BWR stability. This review is still in progress. Questions concerning the need and the methods for improved stability control so as not to exceed core thermal-hydraulic safety limits, and to understand and control, if necessary, adverse affects of power oscillation during anticipated transient without scram (ATWS) events are essentially complete. The ABWR design features indicated above are desirable for improving inherent stability and the system for control rod blocking and insertion to provide exclusion from Region III are consistent with some of the proposed long term solutions for current BWRs. The BWROG has developed several "Long-Term Solutions" (LTS) that have been approved

by the NRC for application to operating reactors. The region exclusion system for the ABWR is similar to the BWROG LTS Option IA. In response to a staff question requesting review of the recent LTS work and the possibility of its application to the ABWR, GE has stated that in addition to the region exclusion system, the BWROG Option III system, the oscillation power range monitor which is based upon the detection of oscillation signals by the LPRMs will be implemented in the ABWR design (See SSAR Section 7.6.1.1.2.2). This LTS methodology has been accepted by the staff review (Letter from A. Thadani, NRC, to L. England, BWROG, "Acceptance for Referencing of Topical Reports NEDO-31960 and NEDO-31960 Supplement 1, `BWR Owners Group Long-Term Stability Solution Licensing Methodology`") and is considered to be acceptable for the ABWR.

Potential Impact Number: 23494 **SRP Section Number:** 4.4

Document: NRC LETTER From A.C. THADANI to L. ENGLISH
Title: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORTS NEDO-32047 AND NEDO-32164, REVISION 0, BWR OWNERS' GROUP EVALUATION OF ATWS RULE ISSUES AND MITIGATIVE ACTIONS (TAC NO. M79766)

Summary:

The letter documents staff findings that NEDO-32047 and NEDO-32164, Revision 0, are acceptable for refencing in license applications to address their concerns regarding BWR thermal-hydraulic instability during ATWS events. The SER contains the following staff conclusions:

(1) On the basis of its review of TRACG qualifications for power oscillation analyses, the staff concludes that TRACG can serve as an adequate tool to estimate qualitatively the global behavior of operating reactors during transients that may result in large power oscillations.

(2) Although large power oscillations may worsen the overheating and severity of fuel damage resulting from an ATWS event, the analyses indicate that core coolability and containment integrity can be maintained. Therefore, the staff concludes that the prescriptive requirements of the ATWS rule remain appropriate.

(3) The BWROG proposed revisions to the EPGs direct immediate action to reduce core inlet subcooling after confirmation of an ATWS event (control rods cannot be inserted), and direct earlier injection of boron in the presence of power oscillations. The

staff review of the details of the proposed revisions to assure that they are appropriate from a risk perspective is continuing. However, these revisions are sufficient for mitigating the consequences of a bounding ATWS event with oscillations.

(4) Reactor vessel water level reduction to reduce core inlet subcooling should continue until the collapsed water level is reduced and maintained at a level about one meter or more below the feedwater spargers to ensure effective termination of power oscillations. In this manner, when feedwater flow is restored, the incoming cold water is heated by mixing with steam. Additional water level reduction is not of significant benefit from a stability point of view. Other criteria must be applied to determine whether level reduction below the top of active fuel (TAF) is warranted.

Potential Impact Number: 23495 **SRP Section Number:** 3.9.6

Document: ADVANCE SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

As discussed in SECY-90-016, the staff determined that ASME Code Section XI requirements do not assure the necessary level of component operability that is desired for evolutionary LWR designs. Accordingly, the staff recommended criteria to the Commission to be used to supplement Section XI of the ASME Code. In its SRM of June 26, 1990, on SECY-90-016, the Commission approved the following staff's position to be used on evolutionary LWRs:

Piping design should incorporate provisions for full flow testing (maximum design flow) of pumps and check valves.

Check valve testing should incorporate the use of advanced non-intrusive techniques to address degradation and performance characteristics.

A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation which cannot be detected through the use of advanced non-intrusive techniques.

Designs should incorporate provisions to test motor operated valves under design basis differential pressure.

APPENDIX II
Potential Impacts

The ASER documents GE commitments as responsive to SECY-90-016, as follows:

- Safety-related pumps and piping configurations can accommodate inservice testing at a flow rate at least as large as the maximum design flow for the pump.
- ABWR safety-related piping systems incorporate provisions for testing to demonstrate the operability of the check valves and power-operated valves (POVs) under design-basis conditions.
- The COL applicant will develop a program to establish the frequency and the extent of disassembly and inspection for all safety-related pumps, check valves, MOVs, and POVs, including the basis for the frequency and the extent of each disassembly.
- Advanced non-intrusive techniques will be used in the implementation of inservice testing program to periodically assess degradation and the performance characteristics of check valves and POVs.
- The COL applicant will determine the optimal frequency of periodic verification of the continuing MOV capability for design basis conditions per GL 89-10 Paragraphs D and J and will include this requirement in the development of the detailed IST program.
- Inservice testing of MOVs will rely on diagnostic techniques that are consistent with the state of the art which will permit an assessment of the performance of the valve under actual loading conditions.

Potential Impact Number: 23496 **SRP Section Number:** 3.9.6

Document: ADVANCE SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The ABWR SSAR states that IST of safety-related pumps and valves will be performed in accordance with the requirements of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, Parts 1, 6, and 10. SSAR Table 1.8-21 also indicates that the applicable code of record for the ABWR is the ASME Code 1989 edition. ASME/ANSI Part 6 of Operations and Maintenance (OM), "Inservice Testing of Pumps," and Part 10, "Inservice Testing of Valves," are referenced in Section XI, ASME Code 1989 Edition. The 1988 Addenda and the 1989 Edition of Section XI are acceptable for the evolutionary LWR IST provided the analysis of leakage rates and corrective action requirements of OM Part 10, paragraph 4.2.2.3 are applied to containment isolation valve testing.

APPENDIX II
Potential Impacts

The staff requested that GE comply with the analysis of leakage rates and corrective action requirements of ASME/ANSI OM Part 10, paragraph 4.2.2.3, for containment isolation valves. In response, GE stated that leakage rate testing of containment isolation valves will be in accordance with ASME Code Section XI. This was not completely acceptable. OM Part 10, which is referenced in ASME Code Section XI, 1989 edition, only requires containment isolation valves to be tested in accordance with Appendix J to 10 CFR Part 50 and does not require that corrective action be based on exceeding individual valve leakage limits. The staff's position is that the analysis of leakage rates and corrective action requirements of paragraph 4.2.2.3 in OM Part 10 are also applicable to containment isolation valves.

Potential Impact Number: 23500 **SRP Section Number:** 3.9.6

Document: ADVANCE SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The ABWR SSAR states that for each MOV with active safety-related function, the design basis and required operating conditions (including testing) under which the MOV will be required to perform are established for the development and implementation of the design, qualification, and preoperational testing. For the design and qualification of MOVs, the SSAR provides commitments that the COL applicant will establish the design and qualification requirements and will provide acceptance criteria for these requirements. The ASER provides a detailed discussion of the SSAR commitments for COL actions for design, qualification (including justification of prototype test data applicability issues addressed in Generic Letter 89-10, Supplement 1), and preoperational testing of MOVs. The staff found that commitments regarding the design, qualification, and preoperational testing for safety-related MOVs provide a reasonable assurance for demonstrating the adequacy of the MOV capability for the design basis conditions.

To follow the 10 CFR Part 52 design certification process, the staff notes that any change to these commitments would involve an unreviewed safety question and, therefore, requires NRC review and approval prior to implementation. Furthermore any requested change to these commitments shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

APPENDIX II
Potential Impacts

In GL 89-10, the staff recommended that MOVs in a safety-related system should either be designed to prevent mispositioning or be required to be subjected to qualification testing to demonstrate capability to recover from mispositioning. The BWR Owners' Group subsequently submitted a backfit appeal on that GL 89-10 recommendation. The staff, with the assistance of Brookhaven National Laboratory, reviewed and reevaluated the issue and determined that the recommendations for MOVs mispositioning is not necessary for BWRs.

Potential Impact Number: 23503 **SRP Section Number:** 9.4.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.63

Summary:

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.

Potential Impact Number: 23504 **SRP Section Number:** 9.4.5

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

Regulatory Position C.3 provides regulatory guidance applicable for necessary support systems for a station blackout event.

Potential Impact Number: 23505 **SRP Section Number:** 3.9.6

Document: ADVANCE SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

APPENDIX II
Potential Impacts

Summary:

In response to the staff's concern regarding the leak-tight integrity of isolation valves, the ABWR SSAR states that the leak-tight integrity of each valve relied on to provide a leak-tight function will be verified. These valves include (1) pressure isolation valves that provide isolation of pressure differential from one part of a system to another or between systems, (2) temperature isolation valves whose leakage may cause unacceptable thermal stress fatigue or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding of pumps, and (3) containment isolation valves that perform a containment isolation function, including valves that are not a part of the 10 CFR Part 50, Appendix J, Type C, testing program but whose leakage may cause loss of water inventory of a suppression pool.

In the ABWR design, several safety systems connected to the RCPB have design pressures below the rated RCS pressure. Also, some systems that are rated at full reactor pressure on the discharge side of pumps have pump suction below RCS pressure. To protect these systems from RCS pressure, two or more isolation valves will be placed in series to form the interface between the high-pressure RCS and the low-pressure system. The staff reviewed the RCS pressure isolation valves list provided in the SSAR and determined that this list contains all of the RCS pressure isolation valves and was acceptable.

The staff's position is that the leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low-pressure systems. GE stated that the periodic leak rate testing of the RCS pressure isolation valves will be performed in accordance with the surveillance requirements of the ABWR TS. GE also stated that the final proposed ABWR TS, to be considered and approved under the design certification program, will reflect the relevant surveillance requirements of the new BWR Standard Technical Specifications. The periodic leak rate testing requirements for ABWR RCS pressure isolation valves are the same as the relevant surveillance requirements of the new BWR Standard TS and are acceptable.

Potential Impact Number: 23511 **SRP Section Number:** 3.9.6

Document: ADVANCE SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

APPENDIX II
Potential Impacts

Issue 87: Failure of High Pressure Coolant Injection Steamline Without Isolation - This issue addresses a postulated break in the HPCI steam supply line and the uncertainty regarding the operability of the HPCI steam supply line isolation valves under those conditions. A similar situation can occur in the RWCU system. The HPCI steam supply line has two containment isolation valves in series (one inside the containment and one outside), both of which are normally open in most plants. An HPCI supply valve located adjacent to the turbine and the turbine stop valve are normally closed. The RWCU system also has two normally open containment isolation valves that must remain open if the system is to function.

At the valve manufacturers' facilities, only the opening characteristics are tested under operating conditions (because of flow limitations). Although the operation of the valves is tested periodically without steam, the capability of the valves to close against the forces created by the steam flow resulting from a downstream line break has not been demonstrated. The valve type is not under General Electric's (BWR vendor) scope of control, but that it is selected by the plant architect-engineer. This issue is addressed in GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated June 28, 1989, with Supplements 1-4. The staff's specific concern is the adequacy of testing, maintenance, and inspection of motor-operated valves (MOVs) so as to assure that they will remain functional when subjected to the design-basis conditions that are to be considered during both normal operation and abnormal events within the design basis of the plant.

The ABWR SSAR provides commitments for the design, qualification, testing, and inspection of MOVs. The staff's evaluation concludes that the commitments as described in the SSAR provide reasonable assurance for demonstrating the adequacy of the MOV isolation capability for the design-basis conditions and are, therefore, acceptable. Furthermore, the SSAR also states that the COL applicant will address the design qualification and testing for MOVs prior to plant startup. Based on this information, the staff concludes that GE has adequately addressed this issue for the ABWR design.

Potential Impact Number: 23512 **SRP Section Number:** 3.9.6

Document: ADVANCE SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

APPENDIX II
Potential Impacts

Issue 105: Interfacing Systems LOCA at LWRs - To protect against an intersystem LOCA (ISLOCA), designers of future ALWR plants should reduce the possibility of a LOCA outside 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 the normal RCS operating pressure. Where not practical to implement this design criteria, applicants must provide justification demonstrating that it is not practicable to reduce the pressure challenge any further for each interfacing system and component that does not meet the RCS URS.

In SECY-93-087, "Policy, Technical, and Licensing issues pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs", the staff stated that systems that have not been designed to full RCS pressure must include the capability for leak testing of the pressure isolation valves.

The staff will continue to require periodic surveillance and leak rate testing of the pressure isolation valves per TS requirements as a part of the ISI program. GE upgraded the design pressure of piping in 14 systems that interface with the RCS and changed the design pressure of two tanks as the result of the ISLOCA review. The staff concluded that using the staff guidance, GE has modified the ABWR systems design "to the extent practicable" and, hence, has adequately addressed this issue.

Potential Impact Number: 23513 **SRP Section Number:** 12.5

Document: RG 8.38 **Revision:** 0
Title: CONTROL OF ACCESS TO HIGH AND VERY HIGH RADIATION AREAS IN NUCLEAR POWER PLANTS

Summary:

Licensees are required by 10 CFR 20.1101 to develop and implement a radiation protection program appropriate to the potential radiation hazards in their facility. The Regulatory Guide positions provide elements which should be included in the plant procedures and practices for access control to be sure that personnel are protected in high and very high radiation areas.

Potential Impact Number: 23514 **SRP Section Number:** 3.9.6

APPENDIX II
Potential Impacts

Document: ADVANCE SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

Issue II.E.6.1: System Design - In-Situ Testing of Valves - Test Adequacy Study - The objective of this issue was to establish the adequacy of requirements for safety-related valve testing. This issue was divided into four parts during its resolution: (1) pressure isolation valves (PIVs), (2) check valves, (3) reevaluation of thermal-overload protection provisions of RG 1.106 for MOVs, and (4) in-situ testing of MOVs.

Relative to in-situ testing of PIVs, the ABWR SSAR requires that all the PIVs be leak tested in accordance with the ABWR TS. This approach is an acceptable way of addressing this part of the issue. Regarding in-situ testing of check valves, the SSAR indicates that the COL applicant is responsible to perform in-situ full flow testing of check valves, in addition to the ASME Section XI inservice testing requirement. Advanced, non-intrusive techniques will be used to assess degradation and performance characteristics of the check valves. In addition, the COL applicant is to develop a program to establish the frequency and extent of disassembly and inspection of check valves. This approach is responsive to the applicable guidelines of SECY-90-016 regarding inservice testing of pumps and valves and is acceptable for addressing this part of the issue. For reevaluation of MOV thermal-overload protection, the SSAR indicates that the ABWR design complies with RG 1.106 and NUREG-1296, respectively. In addition, the SSAR indicates that the guidelines of GL 89-10 will be implemented. Since the staff has determined that the guidelines of RG 1.106 adequately address thermal-overload protection and GL 89-10 addresses control switch settings, which include thermal overload, the staff concludes that GE's approach to this part of the issue is acceptable.

With respect to in-situ testing of MOVs, the SSAR discusses implementation of the guidelines of GL 89-10 in sufficient detail for the staff to conclude that the applicable positions in SECY-90-016 regarding inservice testing of pumps and valves are addressed and GE has taken an acceptable approach to this part of the issue.

Potential Impact Number: 23515 **SRP Section Number:** 9.3.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.63(a)(2)

APPENDIX II
Potential Impacts

Summary:

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.

Potential Impact Number: 23516 **SRP Section Number:** 9.3.4

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

If a system is required for primary coolant charging and makeup, reactor coolant pump seal cooling or injection, decay heat removal, or maintaining appropriate containment integrity specifically to meet the station blackout duration recommended in Regulatory Position 3.1, the following criteria should be met:

1. The system 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

2. 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.

Potential Impact Number: 23517 **SRP Section Number:** 9.2.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

In a comprehensive NRC review and evaluation of operating experience related to service water systems (NUREG-1275, Volume 3), a total of 980 operational events

involving the ESW system were identified, of which, 12 resulted in complete loss of the ESW system. The causes of failure and degradation included: (1) various fouling mechanisms (sediment deposition, biofouling, corrosion and erosion, foreign material and debris intrusion); (2) ice effects; (3) single failures and other design deficiencies; (4) flooding; (5) multiple equipment failures; and (6) personnel and procedural errors.

Based on the potential public risk reduction, this issue was given a high priority ranking. In resolving the issue, the staff found that the concerns involving ESW system reliability were being addressed on a plant-specific basis under various ongoing NRC and industry initiatives such as the Service Water System Operational Performance Inspection Program, Generic Letter 89- 13, the IPE Program, and EPRI research programs. In addition, ESW system reliability concerns were to be addressed by the Maintenance Rule and in the resolution of Issue 23. The staff's technical findings were documented in NUREG/CR-5910 and SEASF-LR-92-022, Revision 1; the regulatory analysis was documented in NUREG-1461. Thus, this issue was RESOLVED and no new requirements were established.

Potential Impact Number: 23518 **SRP Section Number:** 3.9.6

Document: ADVANCE SER CE CH 20
Title: GENERIC ISSUES

Summary:

Issue 105: Interfacing Systems LOCA at LWRs - To protect against an intersystem LOCA (ISLOCA), designers of future ALWR plants should reduce the possibility of a LOCA outside 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 the normal RCS operating pressure. Where not practical to implement this design criteria, applicants must provide justification demonstrating that it is not practicable to reduce the pressure challenge any further for each interfacing system and component that does not meet the RCS URS.

In SECY-93-087, "Policy, Technical, and Licensing issues pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs", the staff stated that systems that have not been designed to full RCS pressure must include the capability for leak testing of the pressure isolation valves.

APPENDIX II
Potential Impacts

The staff's evaluation of the adequacy of the resolution of this issue for the CE System 80+ design will be documented in the FSER.

Potential Impact Number: 23519 **SRP Section Number:** 3.9.6

Document: ADVANCE SER CE CH 20
Title: GENERIC ISSUES

Summary:

Issue II.E.6.1: In Situ Testing of Valves - Issue II.E.6.1 addresses the adequacy of the requirements for safety-related valve testing. This issue was originally divided into the following four parts: (1) testing of pressure isolation valves under Issue 105, ISLOCA at LWR, (2) check valves operability, (3) compliance with the thermal-overload protection provisions of RG 1.106 for motor-operated valves, and (4) operability verification for motor-operated valves per Generic Letter 89-10.

The staff's evaluation of the design, qualification, pre-operational and inservice testing of check valves and motor-operated valves has determined that the commitments of ABB-CE provide a reasonable assurance for verifying the operabilities for check valves and motor-operated valves and, therefore, are acceptable. The details of the staff's evaluations are contained in Section 3.9.6 of the ASER. Moreover, the System 80+ design will comply with the guidance of RG 1.106 regarding the application of thermal overload protection devices that are integral with the motor starter for electric motors on MOVs. Therefore, Issue II.E.6.1 is satisfactorily resolved for the design.

Potential Impact Number: 23520 **SRP Section Number:** 12.5

Document: ADVANCE SER CE CH 12
Title: RADIATION PROTECTION

Summary:

The staff is in the process of revising some existing regulatory guides and developing additional regulatory guides to address some of the new issues contained in the revised 10 CFR Part 20. The new, or revised regulatory guides mentioned in the SER are listed below.

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Potential Impacts

RG 8.7, "Instructions for Recordkeeping and Recording Occupational Radiation Exposure Data" (Revised)
RG 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program" (Revised)
RG 8.25, "Air Sampling in the Workplace"
RG 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses"
RG 8.35, "Planned Special Exposures"
RG 8.36, "Radiation Dose to the Embryo/Fetus"
RG 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants"

In any application for a COL, the COL applicant shall state whether it will follow the guidance contained in these new and revised regulatory guides, or state what alternative guidance will be used.

Potential Impact Number: 23521 **SRP Section Number:** 3.9.6

Document: ADVANCE SER CE CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," the staff concluded that the requirements of Section XI of the ASME Code provide information on the operational readiness of the components, but, in general, do not necessarily verify the capability of the components to perform their intended safety functions. The staff concluded that the ASME Code does not assure the necessary level of component operability that is desired for the evolutionary LWR designs.

Accordingly, in SECY-90-016, the staff recommended criteria to the Commission to be used to supplement Section XI of the ASME Code. In addition, the staff agreed with the recommendation of the ACRS that the guidelines of Generic Letter (GL) 89-10 "Safety-Related Motor Operated Valve Testing and Surveillance," be applied to evolutionary LWR designs. The ACRS also recommended that the staff resolve the check valve testing and surveillance issue and that consideration be given to industry-proposed alternative ways of meeting IST and surveillance requirements. In

its SRM of June 26, 1990, on SECY-90-016, the Commission approved the following staff's position, as an applicable regulation pursuant to SECY- 92-287, to be used on evolutionary LWRs:

Piping design should incorporate provisions for full flow testing (maximum design flow) of pumps and check valves.

Check valve testing should incorporate the use of advanced non-intrusive techniques to address degradation and performance characteristics.

A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation which cannot be detected through the use of advanced non-intrusive techniques.

Designs should incorporate provisions to test motor operated valves under design basis differential pressure.

The ASER documents ABB/CE commitments as responsive to SECY-90-016, as follows:

- Safety-related pumps and piping configurations can accommodate inservice testing at a flow rate at least as large as the maximum design flow for the pump.
- A program will be developed by the COL applicant to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety-related pumps, including the basis for the frequency and the extent of each assembly.
- Periodic testing per GL 89-10 Paragraphs D and J will be conducted under adequate differential pressure and flow conditions which allow a justifiable demonstration of continuing MOV capability for design basis conditions.
- Inservice testing of MOVs will rely on diagnostic techniques that are consistent with the state of the art and that will permit an assessment of the performance of the valve under actual loading conditions.
- All safety-related piping systems incorporate provisions for testing to demonstrate the operability of power-operated valves (POVs) and check valves under design-basis conditions.
- IST will incorporate the use of advanced non-intrusive techniques to periodically assess degradation and the performance characteristics of POVs and check valves.

Potential Impact Number: 23522 **SRP Section Number:** 3.9.6

APPENDIX II
Potential Impacts

Document: ADVANCE SER CE CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The development of a complete plant specific IST program shall be the responsibility of the COL applicant. The IST program will include the tests performed on each pump and valve and the Code requirement met by each test; test parameters and frequency of the tests; the normal, safety, and fail-safe position of each valve; component type for each pump and valve; and P&ID coordinates for each pump and valve. In addition, any requests for relief shall be submitted by the COL applicant and will be reviewed by the NRC staff, on the basis of the Code edition referenced in 10 CFR 50.55a(f), the System 80+ design, and the inservice testing methods available at the time of the COL application. In the CESSAR-DC, ABB-CE stated that the COL applicant will provide a detailed pump and valve IST program. This is acceptable.

The staff requested ABB-CE to include provisions in its piping system design to accommodate the applicable ASME Code IST requirements. The CESSAR-DC stated that IST plan is developed in accordance with the requirements of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, Parts 1,6, and 10. ASME/ANSI Part 6 of Operations and Maintenance (OM), "Inservice Testing of Pumps," and Part 10, "Inservice Testing of Valves," were referenced in Section XI, ASME Code 1989 Edition. The 1988 Addenda and the 1989 Edition of Section XI are acceptable for the evolutionary LWR IST provided the analysis of leakage rates and corrective action requirements of OM Part 10, paragraph 4.2.2.3 are applied to CIV testing.

CIVs with their required testing are presented in the CESSAR-DC. Those valves for which the Appendix J, Type C test is specified in the CESSAR-DC will also be tested in accordance with OM-10, Subsections 4.2.2.3(e) and 4.2.2.3(f). The staff finds these commitments to be acceptable because these meet the staff's position for the leak-rate testing of CIVs.

Potential Impact Number: 23529 **SRP Section Number:** 3.9.6

Document: ADVANCE SER CE CH 3

APPENDIX II
Potential Impacts

Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The staff noted that the reverse-flow testing includes the leak rate testing for safety-related check valves designated Category A/C as well as the quarterly reverse-flow closure verification for safety-related check valves performing a safety function in the closed position whether categorized as C or A/C. Verification that a Category C valve is in the closed position can be done by visual observation, by an electrical signal initiated by a position-indicating device, by observation of appropriate pressure indication in the system by leak testing, or by other positive means. Furthermore, valve closure verification that is impractical to perform during power operation may be deferred to cold shutdown. If it is also impractical to perform at cold shutdown, it may be deferred to refueling. The CESSAR-DC IST plan was revised to reflect the above staff's position.

Potential Impact Number: 23530 **SRP Section Number:** 3.9.6

Document: ADVANCE SER CE CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The CESSAR-DC states that the leak-tight integrity of each valve relied upon to provide a leak-tight function will be verified. These valves include (1) pressure isolation valves that provide isolation of pressure differential from one part of a system to another or between systems, (2) temperature isolation valves whose leakage may cause unacceptable thermal stress fatigue or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding of pumps, and (3) CIVs that provide isolation capability for the piping system penetrating containment.

CIVs with their required testing are presented in the CESSAR-DC. Those valves for which the Appendix J, Type C test is specified in the CESSAR-DC will also be tested in accordance with OM-10, Subsections 4.2.2.3(e) and 4.2.2.3(f). The staff finds these

commitments to be acceptable because these meet the staff's position for the leak-rate testing of CIVs.

Several safety systems connected to the RCPB have design pressures below the rated RCS pressure. Also, some systems rated at full reactor pressure on the discharge side of pumps have pump suction below RCS pressure. To protect these systems from RCS pressure, two or more isolation valves will be placed in series to form the interface between the high-pressure RCS and the low-pressure system. The CESSAR-DC provides a list of RCS pressure isolation valves and states that those pressure isolation valves will be leak-rate tested in accordance with the CESSAR-DC list and the surveillance requirements specified in the TS. The staff finds the list of RCS pressure isolation valves to be acceptable. Furthermore, the periodic leak rate testing requirements for System 80+ RCS pressure isolation valves as provided in the CESSAR-DC are the same as the relevant surveillance requirements of the new ABB-CE standard TS and are acceptable.

Potential Impact Number: 23533 **SRP Section Number:** 3.9.6

Document: ADVANCE SER CE CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

This report presents the review results of the ABB-CE System 80+ IST program documented in the CESSAR-DC.

This review was performed to determine if ABB-CE complies with the regulatory and ASME Section XI, 1989 Edition, requirements, and included:

- Verification that all pumps and valves classified as safety-related in the CESSAR-DC are included in the IST plan, as required by Section XI. This includes both passive and active components and the verification that all CIVs are included in the IST plan.

- A detailed review of justifications for deferral of testing to the cold shutdown or refueling condition and other proposed Code testing exceptions. As discussed in GL 91-18, it is not the intent of IST to cause unwarranted plant shutdowns or to unnecessarily challenge other safety systems.

- A review of the testing specified in the plan to ensure that all safety-related functions are tested.

- A review of the typical inservice testing connections/configurations.

The review of the IST program was performed utilizing SRP Section 3.9.6; GL No. 89-04, "Guidance on Developing Acceptable IST Programs;" and the minutes of the public meeting on GL 89-04, dated October 25, 1989.

Potential Impact Number: 23534 **SRP Section Number:** 15.6.5

Document: ADVANCE SER CE CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

Existing experimental evidence and recent analysis results show that an inherent mechanism for boron dilution in the PWR RCP loop seals could exist for events (including small break loss of coolant accidents, SBLOCAs) that involve heat removal by reflux cooling. The deborated water in the RCP loop seals could be transported to the core through natural circulation processes or startup of RCPs. Injection of the deborated water into the core would be a significant reactivity addition that could possibly 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.

Subsequently, ABB-CE presented additional information to the staff regarding this issue at a meeting held on February 10, 1994. Information was provided on the use of the FLUENT code for mixing calculations and on the actual flow skirt and lower internals configuration under the RCS conditions cited above. Specific conservatism regarding such things as jet stream effects on mixing were clarified. The effect of time in cycle was discussed and showed that only during 1/3 of the fuel cycle was a return to critical possible even with a pure slug. Also, the volume of condensate formed was related to the number of SI pumps available. Therefore, cases of larger condensate production were found to be of lower probability since it is more probable that all SI pumps would be available. In addition, ABB-CE has further modified the System 80+ EOGs to require technical support center approval prior to RCP restart and has re-ordered the EOG steps to emphasize verification of natural circulation prior to consideration of RCP restart. Finally, ABB-CE indicated that should a prompt critical excursion which

breaches the vessel occur, the containment is expected to remain intact because of the availability of containment sprays and cavity flooding.

The staff considers this issue to be technically resolved. It will remain a confirmatory item, however, pending formal documentation of the information presented at the meeting, incorporation into the CESSAR-DC, and update of the System 80+ EOGs. This issue is identified as part of FSER Confirmatory Item 1.1-1.

Potential Impact Number: 23544 **SRP Section Number:** 12.5

Document: C&S: ANS 3.1
Title: Selection, Qualification and Training of Personnel for Nuclear Power Plants

Summary:

ANSI /ANS 3.1 - 1978 is cited in SRP section 12.5 as a guidance document under the Regulatory Guide and other document Acceptance Criteria list at item number 23. The 1993 version of this standard is current.

Potential Impact Number: 23545 **SRP Section Number:** 12.5

Document: C&S: ANSI N13.7
Title: RADIATION PROTECTION - PHOTOGRAPHIC FILM DOSIMETERS - CRITERIA FOR PERFORMANCE

Summary:

ANSI N13.7 - 1972 is cited in SRP section 12.5 as a guidance document under the Regulatory Guide and other document Acceptance Criteria list at item number 27. The 1989 reaffirmation of the 1983 version of this standard is current.

Potential Impact Number: 23546 **SRP Section Number:** 12.5

Document: C&S: ANSI N42.3-69
Title: Test Procedure for Geiger-Mueller Counters

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Summary:

ANSI N42.3 - 1969 is cited in SRP section 12.5 as a guidance document under the Regulatory Guide and other document Acceptance Criteria list at item number 28. This standard has been redesignated as ANSI/IEEE 309. The 1991 reaffirmation of the 1970 version is current.

Potential Impact Number: 23547 **SRP Section Number:** 3.9.6

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The staff disagrees with EPRI's position that a commitment to check valve testing methods requires a detailed design analysis and concludes that a commitment to check valve testing methods should be part of the Evolutionary Requirements Document. In addition, a requirement should be added to the list of guidelines from EPRI NP-5479, "Application Guidelines for Check Valves in Nuclear Power Plants," to state that, in the selection and application of valves, the plant designer should also consider parts clearance, disc stability, and wear relative to actual operational flow conditions. Pending such changes, the staff will review individual applications for FDA/DC in accordance with the above positions.

The staff maintains that testing method and testability are important parts of reliability assurance and that a commitment to the staff's position on full-flow testing of check valves should be part of the Evolutionary Requirements Document. Pending such a commitment, the staff will review individual applications for FDA/DC in accordance with the above position.

EPRI states that check valve and system design will include provisions for the use of nonintrusive diagnostic techniques. This commitment is consistent with the staff's request and is, therefore, acceptable.

Potential Impact Number: 23548 **SRP Section Number:** 3.9.6

Document: FINAL SER EPRI CH 1

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Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

It is the staff's position that the pumps for ALWR designs should be provided with instrumentation to verify that the net positive suction head (NPSH) is greater than or equal to the NPSH required during all modes of pump operation. EPRI indicates that its provision to test pumps in accordance with ASME/ANSI OM-6 will ensure that sufficient instrumentation and test connections are provided to monitor pump performance and trend degradation. However, OM-6 does not address the above staff position. Therefore, the staff requires a more explicit commitment to provide instrumentation to measure suction pressure. The staff will review individual applications for FDA/DC in accordance with the above position.

EPRI states that system design will provide the capability for pump flow testing on a periodic basis at a flow rate that adequately verifies operability and will not result in pump degradation and as a minimum, the capability will be approximately 60 percent of the design flow rate. However, 60 percent of the design flow rate is not acceptable. Therefore, the staff will review individual applications for FDA/DC in accordance with its position that the capability to test at 100 percent of the design flow rate should be provided.

EPRI states that system designs will be such that the pumps will not operate below the minimum flow required for pump protection for all operating modes. If minimum recirculation flow lines are required, these lines will be sized to ensure that degradation will not occur as a result of continuous operation on bypass. The design of the minimum flow lines, if installed, will permit periodic testing to verify the flow is in accordance with design. This commitment addresses the staff's concern, conforms to the applicable staff guideline, and is, therefore, acceptable.

The staff requires, as a minimum, a commitment to develop a program that will establish the frequency and the extent of disassembly and inspection of safety-related pumps, including the basis for the frequency and extent of each disassembly. Pending such a commitment, the staff will review individual applications for FDA/DC in accordance with the above position.

Potential Impact Number: 23549 **SRP Section Number:** 3.9.6

Document: FINAL SER EPRI CH 5

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Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

In SECY-90-016, the staff concluded that the following aspects of pump and valve testing and inspection are necessary to provide an adequate level of assurance of operability and should be applied to all safety-related pumps and valves and not be limited to ASME Code Classes 1, 2, and 3 components.

- Piping design should incorporate provisions for full-flow testing (maximum design flow) of pumps and check valves.

- Designs should incorporate provisions to test motor-operated valves under design-basis differential pressure.

- Check valve testing should incorporate the use of advanced non-intrusive techniques to address degradation and performance characteristics.

- A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced non-intrusive techniques. The staff notes that current state-of-the-art non-intrusive techniques are insufficient to preclude disassembly and inspection of any pumps or valves. Therefore, with the current non-intrusive techniques, it will be necessary to determine a frequency for disassembly of all these components.

The staff also informed EPRI of staff positions relating to inservice testing of check valves, valves in the emergency diesel generator subsystem, pressure isolation valves, solenoid-operated valves, excess flow check valves, and control rod drive valves.

Potential Impact Number: 23550 **SRP Section Number:** 3.9.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.55a, (b)(2)(vii)

Summary:

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(b) The ASME Boiler and Pressure Vessel Code, which is referenced in the following paragraphs, was approved for incorporation by reference by the Director of the Federal Register on January 1, 1981. A notice of any changes made to the material incorporated by reference will be published in the Federal Register....

(2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Section XI, Division 1, and include addenda through the 1988 Addenda and editions through the 1989 Edition, subject to the following limitations and modifications:....

(vii) Inservice testing of containment isolation valves. When using Subsection IWV in the 1988 Addenda or the 1989 Edition of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, leakage rates for Category A containment isolation valves that do not provide a reactor coolant system pressure isolation function must be analyzed in accordance with paragraph 4.2.2.3(e) of part 10, and corrective actions for these valves must be made in accordance with paragraph 4.2.2.3(f) of part 10 of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987.

(viii) Section XI References to OM Part 4, OM Part 6 and OM Part 10 (Table IWA-1600-1). When using Table IWA-1600-1, "Referenced Standards and Specifications" in the Section XI, Division 1, 1987 Addenda, 1988 Addenda, or 1989 Edition, the specified "Revision Date or Indicator" for ASME/ANSI OM Part 4, ASME/ANSI part 6, and ASME/ANSI part 10 shall be the OMa-1988 Addenda to the OM-1987 Edition.

Potential Impact Number: 23551 **SRP Section Number:** 3.9.6

Document: RG 1.147 **Revision:** 9 , 4/1/92
Title: INSERVICE INSPECTION CODE CASE ACCEPTABILITY ASME SECTION XI DIVISION 1

Summary:

The Section XI ASME Code Cases listed below are acceptable to the NRC staff for application in the inservice inspection of components and their supports for water-cooled nuclear power plants. Their use is acceptable within the limitations stated in the " Inquiry" and " Reply" sections of each individual Code Case, within the limitations of such NRC or other requirements as may exist, and within the additional limitations recommended by the NRC staff and given with the individual Code Case in

the list....N-465, Alternative Rules for Pump Testing, Section XI, Division 1;....N-472, Use of Digital Readout and Digital Measurement Devices for Performing Pump Vibration Testing, Section XI, Division 1.

Potential Impact Number: 23555 **SRP Section Number:** 5.2.4

Document: NRC GENERIC LETTER 88-05
Title: BORIC ACID CORROSION OF CARBON STEEL REACTOR PRESSURE BOUNDARY COMPONENTS IN PWR PLANTS

Summary:

This Generic Letter requests 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 which constitutes a new staff position. Previous staff positions have not considered the corrosion of external surfaces of the reactor coolant pressure boundary.

Potential Impact Number: 23559 **SRP Section Number:** 5.4.2.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

ABB-CE restricts the use of SA 540 Grade B23 or B24 bolts in the RCPB to Class 3 material which has a maximum ultimate tensile strength of 1191 MPa (170 ksi). The use of this class of material is in accordance with the requirements of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." Laboratory and field experience with these materials has indicated resistance to stress corrosion cracking under anticipated service conditions. On this basis, Open Item 5.4.2-6 is resolved.

Potential Impact Number: 23560 **SRP Section Number:** 5.4.2.1

Document: NUREG 0933

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Title: A Prioritization of Generic Safety Issues

Summary:

There have been numerous reports of failed bolts or closure studs. Many of these bolts are related to the primary pressure boundary. Degradation or failure of these studs and bolts reduce the integrity of the reactor coolant pressure boundary. Visual examination is currently the only reliable detection method. A major accident such as a LOCA could occur due to undetected bolting failure.

This issue was resolved and no new requirements were established. However, in order to improve the review of future plants and significant modifications to operating plants, the staff recommended that a new SRP Section be developed to codify existing guidance and industry recommendations.

Potential Impact Number: 23561 **SRP Section Number:** 5.4.2.1

Document: NRC GENERIC LETTER 91-17 Revision Date: 10/17/91
Title: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

Generic letter 91-17 provides information on the U.S. Nuclear Regulatory Commission's resolution of Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants." The NRC has resolved this issue without developing any new requirements. The bases for resolution are further documented in the enclosed NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants".

NUREG-1339 recommends that a new section of the SRP be prepared to provide guidance to the staff for the review of future plants. It is recognized that NUREG-1339 makes an exception for closure studs which are already addressed in SRP sections 5.3.1 and 5.2.3, however, the staff noted the absence of an SRP section on general reviews of bolting and recommended that one be prepared and issued.

Potential Impact Number: 23562 **SRP Section Number:** 5.4.2.1

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Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The staff concludes that the implementation of the requirements in the EPRI Evolutionary Requirements Document Chapter 1, Section 5.3.3 will ensure that metallic fasteners will perform in service as required. The requirements include threaded fasteners used to maintain pressure boundary integrity in the reactor coolant and related systems and in the steam, feedwater, and condensate systems and the threaded fasteners used inside those systems and in pipe and component structural mountings for those systems.

Potential Impact Number: 23563 **SRP Section Number:** 5.4.2.1

Document: NUREG 1339
Title: RESOLUTION OF GENERIC SAFETY ISSUE 29: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

In NUREG-1339, the staff noted the absence of an SRP section on general review of bolting and recommended that one be prepared and issued. Such a section, entitled, for example, "Safety-Related Bolting" would expand the limited coverage on fasteners now included in the SRP and provide a systematic method for implementation of the staff position regarding the basis for resolution of GSI 29.

Potential Impact Number: 23564 **SRP Section Number:** 10.3.6

Document: FINAL SER CE80 CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

The inspection programs described in the CESSAR-DC, Section 10.3.4 will provide further assurance that susceptible piping will not be excessively degraded by corrosion/erosion before it is detected. ABB-CE's inspection programs rely on the EPRI

NP-3944, "Erosion/corrosion in Nuclear Plant Steam Piping: Causes and Inspection Guidelines," for guidance, which has been accepted by the staff. On this basis, DSER Open Item 10.3-7 is resolved.

Potential Impact Number: 23565 **SRP Section Number:** 5.4.2.1

Document: NUREG 0844
Title: NRC INTEGRATED PROGRAM FOR RESOLUTION OF UNRESOLVED SAFETY ISSUES A-3, A-4 AND A-5 REGARDING STEAM GENERATOR TUBE INTEGRITY

Summary:

47248:105 . One of the staff recommended actions in NUREG-0844 is that licensees and applicants should have a secondary water chemistry program to minimize steam generator tube degradation. The specific plant program should incorporate the secondary water chemistry guidelines in EPRI-NP-2704, 'PWR Secondary Water Chemistry Guidelines,' October 1982, and should address measures taken to minimize steam generator corrosion, including materials selection, chemistry limits, and control methods. In addition, the specific plant procedures should include progressively more stringent corrective actions for out-of-specification water chemistry conditions.

Potential Impact Number: 23566 **SRP Section Number:** 5.4.2.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In response to Q281-37, ABB-CE stated that the secondary water chemistry complies with EPRI Report NP-6239, "PWR Secondary Water Chemistry Guidelines," Revision 2. The staff found several inconsistencies between the CESSAR and the EPRI secondary water chemistry guidelines. Technical resolution of these inconsistencies is noted and accepted in the FSER.

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Analyst Summary: The staff found compliance with EPRI NP-6239, coupled with the provided technical basis for the noted inconsistencies to be an acceptable basis for secondary water chemistry control.

Potential Impact Number: 23567 **SRP Section Number:** 5.4.2.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

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 and its subsequent revisions, (secondary water) and as supplemented by Table 1.4 of the FSER are acceptable.

Table 1.4 provides the following limits for the makeup water storage tank: conductivity at 25 degrees (IS/cm)-0.2; oxygen (ppm) maximum-0.100; chloride (ppm) maximum-0.15; fluoride (ppm) maximum-0.15; suspended solids (ppm) maximum*-1.0.

Table 1.4 provides the following limits for makeup water to the steam generators: pH (minimum)-7.0; pH (maximum)-7.5; conductivity at 25 degrees (IS/cm)-0.1; sodium (ppb)-<3.0; silica (ppb)-10.

*Concentration of solids is determined by filtration through a pore size of 0.45 micron filter with a line.

Potential Impact Number: 23568 **SRP Section Number:** 3.9.4

Document: C&S: ANSI B16.5
Title: Pipe Flanges and Flanged Fittings

Summary:

ANSI B16.5 is referenced in SRP Section 3.9.4 as an acceptable standard for quality group D piping systems.

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Potential Impact Number: 23569 **SRP Section Number:** 3.9.4

Document: C&S: ANSI B16.9
Title: Factory-Made Wrought Steel Buttwelding Fittings

Summary:

ANSI B16.9 is referenced in SRP Section 3.9.4 as an acceptable standard for quality group D piping systems.

Potential Impact Number: 23570 **SRP Section Number:** 3.9.4

Document: C&S: ANSI B16.11
Title: Forged Steel Fittings, Socket-Welding and Threaded

Summary:

ANSI B16.11 is referenced in SRP Section 3.9.4 as an acceptable standard for quality group D piping systems.

Potential Impact Number: 23571 **SRP Section Number:** 3.9.4

Document: C&S: ANSI B16.25
Title: Buttwelding Ends

Summary:

ANSI B16.25 is referenced in SRP Section 3.9.4 as an acceptable standard for quality group D piping systems

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Potential Impact Number: 23572 **SRP Section Number:** 3.9.4

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Document: C&S: ANSI B31.1
Title: Power Piping

Summary:

ANSI B31.1 is referenced in SRP Section 3.9.4 as an acceptable standard for quality group D piping systems.

Potential Impact Number: 23573 **SRP Section Number:** 3.9.4

Document: C&S: ANSI B16.34
Title: VALVES - FLANGED, THREADED AND WELDING END

Summary:

ANSI B16.34 is referenced in SRP Section 3.9.4 as an acceptable standard for quality group D piping systems.

Potential Impact Number: 23574 **SRP Section Number:** 5.3.2

Document: C&S: ASTM E185-73
Title: Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels

Summary:

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. The current version of this standard is ASTM E-185-82.

Potential Impact Number: 23575 **SRP Section Number:** 5.3.1

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Document: C&S: ASTM E185
Title: Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels

Summary:

ASTM E-185, "Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear 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 current version of this standard is ASTM E-185-82.

Potential Impact Number: 23576 **SRP Section Number:** 10.3.6

Document: RG 1.31 **Revision:** 3
Title: CONTROL OF FERRITE CONTENT IN STAINLESS STEEL WELD METAL

Summary:

This guide describes a method acceptable to the NRC staff for implementing NRC requirements/regulations with regard to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems in light-water-cooled nuclear power plants.

Potential Impact Number: 23577 **SRP Section Number:** 10.3.6

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

SRP Section 5.4.2.1 does not presently cover the use of austenitic stainless steels. However, the staff reviewed the use of austenitic stainless steel in Section 5.4.2 (which covered the review of SRP Section 5.4.2.1) of this ASER. Specifically;

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"As discussed in Sections 4.5.1, 4.5.2, and 5.2.3 of this report, ABB-CE limited the carbon content of austenitic stainless steels to .065 percent compared to the usual specification limit of .08 percent and imposed controls on welding, fabrication, and water chemistry during start-up and operation to mitigate the occurrence of IGSCC in these stainless steels. These mitigating actions have proven successful in operating reactors for more than 15 years and provide reasonable assurance of control of IGSCC, and therefore, the integrity of these components. On this basis, Open Item 5.4.2-10 is resolved."

Potential Impact Number: 23579 **SRP Section Number:** 10.3.6

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In the FSER, the staff noted that the ferrite content limits for austenitic stainless steel castings and weld metal given in the CESSAR-DC were broader than those in industry guidelines (EPRI URD) and staff guidance (NUREG-CR/4513). ABB-CE eventually limited the ferrite content of austenitic stainless steel castings with normal operating temperatures above 500 F (260 C) to a maximum of 20 percent. ABB-CE also modified the upper ferrite limit to 15 percent for austenitic stainless steel weld metal. The staff believes the lower ferrite content limits for austenitic stainless steel castings and weld metal will provide reasonable assurance that components of these materials maintain adequate fracture toughness for their 60-year life.

Potential Impact Number: 23580 **SRP Section Number:** 8.1

Document: DRAFT FINAL SER ABWR CH 8
Title: ELECTRIC POWER SYSTEMS

Summary:

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. Additional guidance not presently in SRP Chapter 8 is provided regarding approved staff policy (Secy-93-087) regarding the use of IEEE standards not reviewed by NRC.

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Potential Impact Number: 23583 **SRP Section Number:** 9.2.5

Document: FINAL SER EPRI CH 8
Title: EPRI URD-EPD: PLANT COOLING WATER SYSTEMS

Summary:

In the DSER for Chapter 8, the staff stated that EPRI had committed to require that ALWR plant designs referencing the Evolutionary Requirements Document comply with the 'Federal Guidelines on Dam Safety' when the reference pond is formed by a dam or a system of dikes or levees. The staff identified this as a confirmatory issue. The staff has verified that Section 7.3.2.2.1 now requires that the UHS facility be seismic Category I and, where applicable, be designed in accordance with the 'Federal Guidelines on Dam Safety.'

Potential Impact Number: 23584 **SRP Section Number:** 11.2

Document: ADVANCE SER CE CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

Using primary coolant concentrations corresponding to one-percent failed fuel or expected values (as given by GALE code run) whichever is greater for fission products, and expected values for corrosion products, annual liquid effluent data as given by GALE code run, and minimum liquid effluent dilution flow of 2.8 m3/sec, ABB-CE has demonstrated that the sum of the ratios of the annual average liquid effluent concentrations of radionuclides in any unrestricted area to the liquid effluent concentration limits for the respective radionuclides given in 10 CFR Part 20, Sections 20.1001-20.2402, Appendix B, Table 2, Column 2, is well below 1.0 (CESSAR-DC Table 11.2-5 value is 0.091). The staff has independently verified that the annual average liquid effluent concentrations in an unrestricted area for the CESSAR-DC plant will be well below the limits specified in the subject table column. Therefore, the staff finds that the CESSAR-DC design will comply with 10 CFR Part 20, Section 20.1302.

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Potential Impact Number: 23585 **SRP Section Number:** 11.2

Document: ADVANCE SER ABWR CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

GE has met the requirements of 10 CFR Part 20 with the ABWR design that will have the minimum discharge flow rate of 340 m3/hr (1500 gpm) and for which liquid waste can only be discharged from either the HSD sample tank or the LCW sample tank. The staff has considered the potential consequences resulting from reactor operation and has determined that the concentrations of radioactive materials in liquid effluent averaged over a year, as permitted by 10 CFR 20.1302 for the above case will be well below the limits in 10 CFR Part 20, Sections 20.1001 through 20.2421, Appendix B, Table II, Column 2. Instantaneous discharge concentrations of the radionuclides in liquid effluents to an unrestricted area also will be within these limits because GE has indicated (by a submittal dated November 5, 1990) that the discharge rate via the single discharge line will be administratively controlled to conform to these limits. The staff will review the administrative controls to limit the instantaneous discharge concentrations of the radionuclides in liquid effluents to an unrestricted area to within 10 times the limits in 10 CFR Part 20, Sections 20.1001 through 20.2402, Appendix B, Table II, Column 2, on a plant-specific basis for the COL applicants. Therefore, this was identified as DFSER COL Action Item 11.2.2-2. By SSAR Amendment 27, GE identified COL License Information (Section 11.2.5, Item 5) which states the COL applicant will provide administrative controls to ensure the limits mentioned above. GE has also included this action item in the final certified SSAR. The adequacy and acceptability of the final certified SSAR is evaluated in Chapter 1 of this report.

Potential Impact Number: 23587 **SRP Section Number:** 11.2

Document: ADVANCE SER ABWR CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

IE Bulletin 80-05 identified an issue concerning the release of radioactive material or other detrimental effects as a result of low vacuum conditions causing tank buckling. The low vacuum condition is created by cooling hot water in the low-pressure tank. Per fax dated May 21, 1992, GE stated that several low-pressure tanks that could contain

primary system water have vents to prevent the development of a low vacuum condition. In the DFSER, the staff stated that the above information resolves the tank failure concern in IE Bulletin 80-05, subject to documentation of the information in the SSAR. Therefore, the staff identified the resolution of the issue as DFSER Confirmatory Item 11.2.2-1. By SSAR Amendment 27, GE included the above information (Section 11.2.4), which is acceptable to the staff. GE has also included this information in the final certified SSAR. The adequacy and acceptability of the final certified SSAR is evaluated in Chapter 1 of this report. On the basis of this evaluation this item is resolved.

Potential Impact Number: 23588 **SRP Section Number:** 11.2

Document: ADVANCE SER CE CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

ABB-CE has also reviewed the applicability of IE Bulletin 80-05, "Vacuum Conditions Resulting in Damage to Chemical Volume Control System (CVCS) Holdup Tanks (Sometimes Called `Clean Waste Receiver Tanks')" to CESSAR-DC standard design. The subject bulletin addresses the issue concerning the release of radioactive material or other adverse effects as a result of low-vacuum conditions causing tank buckling. The low-vacuum condition is created by cooling hot water in a low pressure tank. Based on its review, ABB-CE has included its findings in CESSAR-DC Subsections 11.2.2.2.1 and 11.2.2.2.2. The subject subsections state that all tanks are provided with either: (1) tank vents sized adequately to prevent the collapse of the tank during draindown; or (2) a vacuum breaker, as necessary. The staff finds the above features adequately address the concern identified in the subject bulletin and, are, therefore acceptable. The staff finds the above features also comply with GDC 61.

Potential Impact Number: 23605 **SRP Section Number:** 11.3

Document: ADVANCE SER CE CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

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ABB-CE has demonstrated that the annual average gaseous effluent concentration of radionuclides in exclusion area boundary (EAB) (CESSAR-DC Table 11.3-6) will be below the gaseous effluent concentration limits listed in 10 CFR Part 20, Sections 20.1001-20.2402, Appendix B, Table 2, Column 1. Table 11.3-6 shows that the sum of the ratios of annual average gaseous effluent concentrations of radionuclides at the EAB to their respective concentration limits given in the subject 10 CFR Part 20 table is less than 1. For the above reasons, the staff considers that the DSER Open Item 11.3-4 is resolved and that the CESSAR-DC GWMS design complies with 10 CFR Part 20, Section 20.1302.

Potential Impact Number: 23606 **SRP Section Number:** 11.3

Document: ADVANCE SER ABWR CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

The ABWR design meets the requirements of 10 CFR Part 20 because the staff has considered the potential consequences resulting from reactor operation with a postulated fission product release rate consistent with an offgas noble gas release rate of 3.7 x 106 Bq/MWt-sec (100 Ci/MWt-sec) at 30 minutes decay for a BWR and estimated that, under these conditions, the concentration of radionuclides in gaseous effluents in unrestricted areas with a value of X/Q that is equal to or less than 9.8 x 10-6 sec/m3 will be below the concentration values in 10 CFR Part 20, Sections 20.1601 through 20.2402, Appendix B, Table II, Column 1 for KR-89 and XE-137 whose specific concentration values are not explicitly given in the above table, the staff used relevant concentration values for these radionuclides, based on ratios of wholebody dose factors of noble gas radionuclides given in RG 1.109 (Rev. 1).

Potential Impact Number: 23618 **SRP Section Number:** 11.4

Document: ADVANCE SER CE CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

The spent resin tanks and the shipping containers are sampled and surveyed to verify that the dewatering or solidification (if required) and packaging are complete and meet the guidelines of BTP ETSB 11-3, which provides the design guidance for SWMS. Additionally, the wet solid wastes will be processed and disposed in accordance with 10 CFR Part 61 requirements. To ensure the above compliance, ABB-CE has identified a COL interface (CESSAR-DC, Sec- tion 11.4.1.1, Item F) which calls for the owner operator to process and classify the packaged waste in accordance with 10 CFR Part 61 requirements. Specifically, the COL interface calls for the owner operator to develop operating procedures (i.e., boundary conditions for a set of process parameters such as settling time, drain time, drying time, etc.) for processing the wet solid wastes to assure that 10 CFR Part 61 requirements are met. The staff recognizes that such development of procedures (i.e., process control program (PCP)) to ensure that the SWMS will operate as intended is within the scope of COL applicant. The staff will therefore review PCP, including dewatering or solidification, (if performed), on a plant-specific basis against BTP ETSB 11-3 guidelines and 10 CFR Part 61 requirements.

Potential Impact Number: 23619 **SRP Section Number:** 11.4

Document: NRC MEMORANDUM 02/14/83 from T.C. Johnson to R.E. Browning
Title: BTP ON WASTE FORM

Summary:

A Branch Technical Position was enclosed with a 2-14-83 Memorandum (WMLL: 204.1.5) from the Low-Level Waste Licensing Branch to the Division of Waste Management. This BTP reflected the current NRC staff position on acceptable means for meeting the 10CFR Part 61 waste solidification requirements.

Potential Impact Number: 23620 **SRP Section Number:** 11.4

Document: ADVANCE SER CE CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

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The staff considered the capability of the system in conjunction with LWMS and GWMS to maintain any liquid or gaseous effluents in unrestricted areas, arising from the system operation, below the limits in 10 CFR Part 20, Section 20.1302.

The staff concludes that the CESSAR-DC SWMS complies with 10 CFR Part 20, Section 20.1302 and GDC 60, 63, and 64.

Additionally, the staff concludes that the SWMS design will comply with 10 CFR Part 61, and 10 CFR Part 71 requirements and DOT regulations; however, since demonstration of such compliance falls within the scope of the COL applicant, the staff will review compliance of SWMS design with 10 CFR Part 61 and 10 CFR Part 71 on a plant-specific basis. For the above reasons, the staff concludes that CESSAR-DC SWMS meets the acceptance criteria of SRP Section 11.4 and is, therefore, acceptable.

Potential Impact Number: 23628 **SRP Section Number:** 11.5

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION
Location: Subpart D; 20.1302

Summary:

20.1302 Compliance with dose limits for individual members of the public.
(a) The licensee shall make or cause to be made, as appropriate, surveys of radiation levels in unrestricted and controlled areas and radioactive materials in effluents released to unrestricted and controlled areas to demonstrate compliance with the dose limits for individual members of the public in 20.1301.
(b) A licensee shall show compliance with the annual dose limit in 20.1301
(1) Demonstrating by measurement or calculation that the total effective dose equivalent to the individual likely to receive the highest dose from the licensed operation does not exceed the annual dose limit; or
(2) Demonstrating that--
(i) The annual average concentrations of radioactive material released in gaseous and liquid effluents at the boundary of the unrestricted area do not exceed the values specified in table 2 of appendix B to 20.1001-20.2401; and
(ii) If an individual were continually present in an unrestricted area, the dose from external sources would not exceed 0.002 rem (0.02 mSv) in an hour and 0.05 rem (0.5 mSv) in a year.

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Potential Impact Number: 23630 **SRP Section Number:** 11.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Part 50.36a

Summary:

(a) In order to keep releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, as low as is reasonably achievable, each license authorizing operation of a nuclear power reactor will include technical specifications that, In addition to requiring compliance with applicable provisions of 20.1301 of this chapter, require:
(1) That operating procedures developed pursuant to Section 50.34a(c) for the control of effluents be established and followed and that equipment installed in the radioactive waste system, pursuant to Section 50.34(a), be maintained and used.
(2) Each licensee shall submit a report to the Commission annually that specifies the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous 12 months of operation, including any other information as may be required by the Commission to estimate maximum potential annual radiation doses to the public resulting from effluent releases.

(b) In establishing and implementing the operating procedures described in paragraph (a) of this section, the licensee shall be guided by the following considerations: Experience with the design, construction, and operation of nuclear power reactors indicates that compliance with the technical specifications described in this section will keep average annual releases of radioactive material in effluents and their resultant committed effective dose equivalents at small percentages of the dose limits specified in 20.1301 and in the operating license. At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual operating conditions which may temporarily result in releases higher than such small percentages, but still within the dose values specified in 20.1301 of this chapter and in the operating license.

The guides set out in appendix I provide numerical guidance on limiting conditions for operation for light-water-cooled nuclear power reactors to meet the requirement that radioactive materials in effluents released to unrestricted areas be kept as low as is reasonably achievable.

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Potential Impact Number: 23631 **SRP Section Number:** 11.5

Document: ADVANCE SER CE CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

To comply with 10 CFR Part 50, Appendix I, numerical objectives for the offsite doses due to effluents during normal plant operation including AOOs, plants will be required to limit annual and 3-month offsite doses. Also, the plants will be required to use treatment systems (e.g., waste gas treatment by delay beds; demineralizers to treat liquid radwastes; ventilation exhaust filtration systems), if the monthly dose is likely to exceed about 25 percent of the Appendix I annual dose guidelines prorated for one month. These requirements will be specified in a plant-controlled document and will be implemented.

Additionally, plants will be required to limit instantaneous discharge concentrations from the process gas subsystem and LWMS to comply with 10 CFR Part 20, Section 1302. Plants will be required to provide the associated set points for the applicable radiation monitors in the plant-specific ODCMs. These requirements, in conjunction with the automatic control (i.e., termination of the discharge) features of applicable effluent monitors, will ensure CESSAR-DC effluent monitors are designed in compliance with 10 CFR Part 20, Section 20.1302. The staff finds this acceptable. The staff will review the plant-specific radiological effluent technical specifications (RETS) that will be provided in a plant-controlled document as well as setpoints in the plant-specific ODCM on plant-specific basis.

As stated in the above discussion, the staff will review the RETS, ODCM and the operation and maintenance manual (the subject manual for demonstrating compliance with the TMI requirements referred to above), and specific compliance with the guidelines of ANSI N13.1-1969, RGs 1.21 and 4.15, on a plant-specific basis.

Potential Impact Number: 23633 **SRP Section Number:** 11.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix I, section IV, Paragraph B.1

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Summary:

The licensee shall establish an appropriate surveillance and monitoring program to provide data on quantities of radioactive material released in liquid and gaseous effluents to assure that the provisions of paragraph A of this section are met;.

Potential Impact Number: 23634 **SRP Section Number:** 11.5

Document: NUREG 1301
Title: OFFSITE DOSE CALCULATION MANUAL GUIDANCE: STANDARD RADIOLOGICAL EFFLUENT CONTROL FOR PRESSURIZED WATER REACTORS

Summary:

NUREG-1301 was forwarded as guidance to licensees who elected to implement Generic Letter 80-01.

NUREG-1301 contains guidance which may be used by licensees who choose to implement the provision of Generic Letter 89-01, which allows Radiological Effluent Technical Specifications (RETS) to be removed from the main body of the Technical Specifications and placed in the Offsite Dose Calculation Manual (ODCM). Guidance is provided for standard effluent controls definitions, controls for effluent monitoring instrumentation, controls for effluent releases, controls for radiological environmental monitoring, and the basis for controls. Guidance of the formulation of RETS has been available in draft form (NUREG-0472 and -0473) for a number of years; the current effort simply recasts those RETS into standard radiological effluent controls for application to the ODCM. Also included for completeness are: (1) radiological environmental monitoring program guidance previously which had been available as a Branch Technical Position (Rev. 1, November 1979); (2) existing ODCM guidance; and (3) a reproduction of Generic Letter 89-01.

Potential Impact Number: 23635 **SRP Section Number:** 11.5

Document: NUREG 1302
Title: OFFSITE DOSE CALCULATION MANUAL GUIDANCE: STANDARD RADIOLOGICAL EFFLUENT CONTROLS FOR BOILING WATER REACTORS

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Summary:

Generic Letter 89-01 Sup. 1 forwarded NUREG-1301 as guidance to PWR licensees who elected to implement Generic Letter 80-01. NUREG-1302 provides similar guidance for BWRs.

NUREG-1302 contains guidance which may be used by licensees who choose to implement the provision of Generic Letter 89-01, which allows Radiological Effluent Technical Specifications (RETS) to be removed from the main body of the Technical Specifications and placed in the Offsite Dose Calculation Manual (ODCM). Guidance is provided for standard effluent controls definitions, controls for effluent monitoring instrumentation, controls for effluent releases, controls for radiological environmental monitoring, and the basis for controls. Guidance of the formulation of RETS has been available in draft form (NUREG-0472 and -0473) for a number of years; the current effort simply recasts those RETS into standard radiological effluent controls for application to the ODCM. Also included for completeness are: (1) radiological environmental monitoring program guidance previously which had been available as a Branch Technical Position (Rev. 1, November 1979); (2) existing ODCM guidance; and (3) a reproduction of Generic Letter 89-01.

Potential Impact Number: 23637 **SRP Section Number:** 11.5

Document: NRC GENERIC LETTER 89-01 Sup 1
Title: NUREG 1301 OFFSITE DOSE CALCULATION MANUAL GUIDANCE: STANDARD RADIOLOGICAL EFFLUENT CONTROLS FOR PRESSURIZED WATER REACTORS

Summary:

Enclosed, for your information, is a copy of NUREG-1301, Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent
Controls for Pressurized Water Reactors. This NUREG report does not impose any new requirements or staff positions; its use is therefore totally voluntary. The subject NUREG supersedes prior draft guidance for RETS (NUREG-0472) for those licensees who elect to implement Generic Letter 89-01. Licensees not electing to implement this Generic Letter should continue to follow the prior draft RETS guidance.

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Potential Impact Number: 23640 **SRP Section Number:** 16.0

Document: ADVANCE SER ABWR CH 16
Title: TECHNICAL SPECIFICATIONS

Summary:

The staff review of the advanced boiling water reactor (ABWR) technical specifications (TSs) was closely coupled to the development of the improved Standard Technical Specifications (STS) under the TSs Improvement Program in accordance with the Commission Policy Statement on TSs Improvement. Since the ABWR design evolved primarily from the BWR/6 design, most of the ABWR TSs were modeled after NUREG-1434, "Standard Technical Specifications General Electric Plants, BWR/6." However, the ABWR containment most closely compared to plant containments covered by NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4." Therefore, the ABWR containment TSs were modeled after NUREG-1433. These improved STS contain the benefits of the accumulated operating experience from currently operating light water reactors.

As part of the TSs Improvement Program, the staff concluded that portions of STS Section 5.0, Administrative Controls could be relocated to licensee controlled documents. This improvement was incorporated into the ABWR TSs. COL applicants will have to assure that the portions of Section 5.0 relocated to licensee controlled documents are controlled in accordance with an administrative control system acceptable to the staff.

Based on the staff review of the ABWR TSs, as discussed above, the staff concludes that the ABWR TSs are consistent with the regulatory guidance contained in the BWR/6 and BWR/4 STS and contain design-specific parameters and additional TSs requirements considered appropriate by the staff. Therefore the staff concludes that the ABWR TSs satisfy Sections 50.34 and 50.36 of 10 CFR Part 50 and are acceptable contingent upon the staff independent audit of and the GE certification of the final ABWR TSs.

Potential Impact Number: 23641 **SRP Section Number:** 16.0

Document: ADVANCE SER CE CH 16
Title: TECHNICAL SPECIFICATIONS

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Summary:

The staff review of the System 80+ Technical Specifications (TSs) was closely coupled to the development of the improved Standard Technical Specifications (STS) under the TSs Improvement Program in accordance with the Commission Policy Statement on TSs Improvement 58 FR 39132 (SECY-93-067). Since the System 80+ design evolved primarily from the System 80 design, most of the System 80+ TSs were modeled after NUREG-1432, "Standard Technical Specifications for Combustion Engineering Plants" (January 1991). These improved STS contain the benefits of the accumulated operating experience from currently operating light-water reactors.

The System 80+ design, however, has some major changes from the System 80 design. The System 80+ TSs address these changes and the issue of shutdown risk considerations. On this basis, the staff's review of the System 80+ TSs concentrated on the differences from the System 80 STS (NUREG-1432).

On the basis of its review of the System 80+ TSs, the staff concludes that the System 80+ TSs will be consistent with the regulatory guidance contained in the System 80 STS (NUREG-1432) and will contain design-specific parameters and additional TS requirements considered appropriate by the staff. Therefore, the staff concludes that the System 80+ TSs will satisfy 10 CFR 50.34 and 10 CFR 50.36, and will be acceptable, contingent upon (1) resolution and incorporation of the staff proof and review comments, particularly those related to development of the new TS requirements for low power and shutdown conditions, (2) satisfactory completion of the independent audit, and (3) ABB-CE certification of the final System 80+ TSs.

Potential Impact Number: 23642 **SRP Section Number:** 16.0

Document: NRC POLICY STATEMENT 58 FR 39132 Revision Date: 7/22/93
Title: FINAL POLICY STATEMENT ON TECHNICAL SPECIFICATIONS IMPROVEMENTS FOR NUCLEAR POWER PLANTS

Summary:

This statement presents the policy of the Nuclear Regulatory Commission (NRC) with respect to the scope and purpose of Technical Specifications for nuclear power plants as required by 10 CFR 50.36. It establishes a specific set of objective criteria as guidance for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. It encourages licensees to implement a

voluntary program to update their Technical Specifications to be consistent with improved vendor-specific Standard Technical Specifications (STS) issued by the NRC in September 1992. The improved STS were published as the following NRC Reports:

NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants",
NUREG-1431, "Standard Technical Specifications,
Westinghouse Plants",
NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants",
NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4",
and
NUREG-1434, "Standard Technical Specifications, General Electric Plants, BWR/6".

These improved STS were the result of extensive technical meetings and discussions among the NRC staff, industry owners groups, vendors, and the Nuclear Management and Resources Council (NUMARC). The improved STS were developed based on the criteria in the interim Policy Statement published in February 1987. The Policy Statement now reflects modifications resulting from public comments on the interim Policy Statement and from the experience gained in developing the improved STS. Implementation of the Policy Statement through implementation of the improved STS is expected to produce an improvement in the safety of nuclear power plants through the use of more operator-oriented Technical Specifications, improved Technical Specification Bases, reduced action statement induced plant transients, and more efficient use of NRC and industry resources. The Policy Statement is not a regulation and does not establish binding requirements or limit the scope of safety issues for case-specific adjudication.

Potential Impact Number: 23646 **SRP Section Number:** 5.3.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Part 50.60

Summary:

Section 50.60 Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation.

(a) Except as provided in paragraph (b) of this section, all lightwater nuclear power reactors must meet the fracture toughness and material surveillance program

requirements for the reactor coolant pressure boundary set forth in Appendices G and H to this part.

(b) Proposed alternatives to the described requirements in Appendices G and H of this part or portions thereof may be used when an exemption is granted by the Commission under Section 50.12.

Potential Impact Number: 23656 **SRP Section Number:** 8.2

Document: FINAL SER CE80 CH 8
Title: ELECTRIC POWER SYSTEMS

Summary:

The staff required ABB-CE to address the following EPRI guidelines regarding plant grounding and lightning protection:

The plant grounding grid, consisting of bare copper cables, shall limit the step and touch potentials to safe values under all fault conditions.

The grounding system shall have bare copper risers for all electrical underground ducts and equipment, and for connections to the grounding systems within buildings.

The design and analysis of the grounding system shall follow the procedures and recommendations of the latest revision of IEEE 665.

Each building shall have grounding systems connected to the plant grounding grid. As a minimum, every other steel column of the building perimeter will be connected directly to the grounding grid.

The main generator shall be grounded with a neutral grounding device having an impedance that shall limit the maximum phase current under short circuit conditions to a value not greater than that for a three- phase fault at its terminals.

Suitable grounding device(s) shall ensure proper grounding of the isophase buses when the generator is disconnected from the isophase buses.

The onsite medium voltage ac distribution system shall be resistance grounded at the neutral point of the low-voltage windings of the unit auxiliary and reserve transformers.

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The neutral point of the generator windings of the onsite standby power supply units, Class 1E and non-Class 1E, shall be grounded through distribution-type transformers and loading resistors sized for continuous operation with a ground fault.

The neutral point of the low-voltage ac distribution systems shall be either solidly or impedance grounded, as necessary to properly coordinate ground fault protection.

The dc systems shall be ungrounded.

Each major piece of equipment, metal structure, or metallic tank shall have two ground connections diagonally opposite each other.

The ground bus of all switchgear assemblies, MCCs, and control cabinets shall be connected to the plant ground grid through at least two parallel paths.

One bare copper cable shall be installed with each underground electrical duct run, and all metallic hardware in each personnel hatch shall be connected to this cable.

Plant instrumentation shall be grounded through separate radial grounding systems, consisting of isolated instrumentation ground buses and insulated cables. The instrumentation grounding systems shall be connected to the plant grounding grid at one point only and shall be insulated from all other grounding circuits.

*The plant analog and digital instrumentation systems shall have separate instrumentation grounding systems.

All major plant structures, including the containment enclosure building, shall be protected from lightning. The lightning protection shall be in accordance with the National Fire Protection Association's Lightning Protection Code, NFPA-78, and the Nuclear Energy Property Insurance Association's Basic Fire Protection for Nuclear Power Plants.

Each phase of all tie lines connecting the plant electrical systems to the switching station(s) and offsite transmission system shall be protected by lightning arresters. These arresters will be connected to the high voltage terminals of the main stepup and reserve transformers.

Plant instrumentation and monitoring equipment located outdoors or connected to outdoor cables shall have built-in surge-suppression devices to protect the equipment from lightning-induced surges.

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ABB-CE stated that separate ground buses for digital and analog instrumentation system are not necessary or practical because many vendor-supplied systems contain both analog and digital components in the same cabinets and they do not supply separate ground buses for these components. Further, separate buses would require a significant amount of cabling and more congested cabinets. Also, IEEE 1050-1989, "IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations," does not require separate ground buses for analog and digital systems. This justification is acceptable. Further, the inclusion of each of the remaining EPRI plant grounding guidelines will ensure that plant structures, systems, and equipment will be appropriately grounded and protected from lightning. Therefore, this design is consistent with the protection and independence requirements of GDC 17 and is acceptable.

Potential Impact Number: 23657 **SRP Section Number:** 8.2

Document: ADVANCE SER ABWR CH 8
Title: ELECTRIC POWER SYSTEMS

Summary:

Discusses the application, for the ABWR, of the staff position described in SECY 93-087, Section II.B., on independence of safety systems during operation or failure of non-class 1E loads.

Potential Impact Number: 23658 **SRP Section Number:** 8.2

Document: FINAL SER ABWR CH 8
Title: ELECTRIC POWER SYSTEMS

Summary:

In response to ACRS and staff concerns, GE has committed to meet the following Electric Power Research Institute (EPRI) plant grounding guidelines:

A station grounding grid, consisting of bare copper cables, will be provided that will limit step-and-touch potentials to safe values under all fault conditions.

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Bare copper risers will be furnished for all underground electrical ducts and equipment, and for connections to the grounding systems within buildings.

The design and analysis of the grounding system will follow the procedures and recommendations specified by the latest revision of IEEE 665, "Guide for Generation Station Grounding."

Each building will be equipped with grounding systems connected to the station grounding grid. As a minimum, every other steel column of each building perimeter will connect directly to the grounding grid.

The plant's main generator will be grounded with a neutral grounding device. The impedance of that device will limit the maximum phase current under short-circuit conditions to a value not greater than that for a three-phase fault at its terminals.

Provisions will be included to ensure proper grounding of the isophase buses when the generator is disconnected.

The onsite, medium-voltage ac-distribution system will be resistance grounded at the neutral point of the low-voltage windings of the unit auxiliary and reserve transformers.

Grounding of the neutral point of the generator windings of the onsite standby power supply units (Class 1E diesel generators and non-Class 1E combustion turbine generator (CTG)) will be through distribution-type transformers and loading resistors, sized for continuous operation in the event of a ground fault.

The neutral point of the low-voltage ac-distribution systems will be either solidly or impedance grounded, as necessary, to ensure proper coordination of ground fault protection.

The dc systems will be left ungrounded.

Each major piece of equipment, metal structure, or metallic tank will be equipped with two ground connections diagonally opposite each other.

The ground bus of all switchgear assemblies, MCCs, and control cabinets will be connected to the station ground grid through at least two parallel paths.

One bare copper cable will be installed with each underground electrical duct run, and all metallic hardware in each manhole will be connected to this cable.

Plant instrumentation will be grounded through separate radial grounding systems consisting of isolated instrumentation ground buses and insulated cables. The instrumentation grounding systems will be connected to the station grounding grid at only one point and will be insulated from all other grounding circuits.

Separate instrumentation grounding systems shall be provided for plant analog and digital instrumentation systems.

A lightning protection system will be provided for each major plant structure, including the containment enclosure building. The design and installation of these systems will comply with the National Fire Protection Association's Lightning Protection Code, NFPA-78, and the Nuclear Energy Property Insurance Association's "Basic Fire Protection for Nuclear Power Plants" document.

Lightning arresters will be provided in each phase of all tie lines connecting the plant electrical systems to the switching station(s) and offsite transmission system. These arresters will be connected to the high-voltage terminals of the main step-up and reserve transformers.

Plant instrumentation and monitoring equipment located outdoors or connected to cabling that runs outdoors will be equipped with built-in surge suppression devices to protect the equipment from lightning- induced surges.

GDC 17 of 10 CFR Part 50, Appendix A, requires provisions be included in the design to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. The staff concludes that a design for plant structures, systems, and equipment which meets the above described commitments will be appropriately grounded and protected from lightning. The design, therefore, meets the above defined requirements of GDC 17 and is acceptable.

Potential Impact Number: 23659 **SRP Section Number:** 8.2

Document: FINAL SER EPRI CH 11
Title: EPRI URD-EPD: ELECTRIC POWER SYSTEMS

Summary:

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Provides additional background information for the staff position described in SECY 91-078 on an alternate power source for non-safety loads.

Potential Impact Number: 23660 **SRP Section Number:** 8.2

Document: FINAL SER EPRI CH 11
Title: EPRI URD-EPD: ELECTRIC POWER SYSTEMS

Summary:

Provides additional background information for the staff position described in SECY 91-078 on connection of offsite power sources to safety buses.

Potential Impact Number: 23661 **SRP Section Number:** 8.2

Document: C&S: ANSI C37.06
Title: Switchgear-AC High-Voltage Circuit Breakers Rated on a Symmetrical Current Basis-Preferred Ratings and Related Required Capabilities

Summary:

ANSI C37.06 is referenced and discussed in Appendix A to SRP Section 8.2

Potential Impact Number: 23662 **SRP Section Number:** 8.2

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

The staff believes that the preferred method of demonstrating compliance with 10 CFR 50.63 is through installation of a spare (full capacity) alternate ac power source of diverse design that is consistent with the guidance in Regulatory Guide 1.155 and is

capable of powering at least one complete set of normal safe shutdown loads. The staff recommended that the Commission approve imposition of an alternate ac source for evolutionary ALWRs.

Potential Impact Number: 23663 **SRP Section Number:** 10.3

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Section 3.5.2 of Chapter 1 requires the PWR plant to be capable of operating at reduced power with a secured coolant pump to enhance the availability of the plant and to reduce reactor trips. Appendix B to Chapter 1 indicates EPRI's commitment to comply with Generic Letter (GL) 86-09, Technical Resolution of Generic Issue No. B-59 - (N-1) Loop Operation in BWRs and PWRs. GL 86-09 states that (N-1) loop operation is acceptable provided acceptable evaluation results are shown for certain plant-specific design characteristics, such as the impact of the down loop on instrumentation and control systems, human factors, operational systems, safety systems, status of valves, core-flow distribution, and potential for cold water reactivity insertion. Since these characteristics are highly dependent on the specific design of the plant, acceptability of operation with one secured reactor coolant pump is subject to plant-specific evaluation to address the concerns delineated in GL 86-09. The staff will evaluate that analysis during its review of an individual application for FDA/DC.

Potential Impact Number: 23664 **SRP Section Number:** 10.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

The purpose of this task is to develop a set of acceptance criteria, review guidelines, and TS changes for the (N-1) loop authorization requests. This set of criteria, guidelines, and TS changes will encompass accident scenarios (both LOCAs and non-LOCAs) to be analyzed by the licensees, computer models acceptable to NRC for these analyses and acceptable input parameters in terms of reactor operating

conditions (such as allowance for uncertainties in power level and fluid measurement). This has already been accomplished for PWRs by virtue of the completion of the SER for BV-1. In addition, the BWR analyses have been reviewed and accepted by the staff and the appropriate generic TS changes to allow SLO for BWRs have been identified.

It is concluded that this is a Regulatory Impact issue which has been resolved for BWRs and PWRs. For PWRs, the issue has been resolved on the basis of the BV-1 SER (MPA E-05) and, for BWRs, the issue has been resolved on the basis of Item B-19, the plant-specific tests at Browns Ferry, and the review of licensee submittals under MPA E-04.

Potential Impact Number: 23665 **SRP Section Number:** 10.3

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

In Section 2.3.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI proposes to eliminate the BWR main steam isolation valve leakage control system and provide an alternative leakage pathway (i.e., the main steamline and the condenser) to the main condenser downstream of the isolation valves in the event of a loss-of-coolant accident (LOCA). This issue is also related to Section 3.3.2 in Appendix B to Chapter 1, Section 3.4.1.5 in Chapter 2, Sections 5.3.3 and 5.4.1.5 in Chapter 3, Section 1.2.3 in Chapter 5, and Section 3 in Chapter 13 of the Evolutionary Requirements Document. In its letters dated May 17 and August 29, 1991, the staff stated its positions on the quality group and seismic classifications of the BWR main/extraction steam system, main steamline, main turbine system, condenser, and related non-seismic structures that resulted from the proposed elimination of the main steam isolation valve leakage control system. The objective of these staff positions is to establish adequate measures to ensure that the above systems and structures will maintain their structural integrity during and following an SSE.

The proposed approach developed by EPRI and the BWROG would allow higher leakage limits and would take credit for the main steam piping and condenser to plate out the fission products following core damage. In this way, the main steam piping (and its associated piping) and the condenser would be used to mitigate the radiological consequences of an accident that could result in potential offsite exposures comparable to the dose reference values in 10 CFR Part 100.

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Potential Impact Number: 23666 **SRP Section Number:** 10.3

Document: FINAL SER ABWR CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

In addition to meeting the acceptance criteria of SRP Section 10.3, the MS system for the ABWR must also be capable of mitigating the radiological consequences of an accident that could result in potential offsite exposures comparable to the dose reference values specified in 10 CFR Part 100. Most of the currently licensed boiling water reactors rely on the MSIV leakage control system to mitigate the radiological consequences of MSIV leakage following a design-basis loss-of-coolant accident and to stay within 10 CFR Part 100 limits if the MSIV leakage rate exceeded the technical specification limit of 0.3 m3/hr (11.5 ft3/hr). The ABWR will not have an MSIV leakage control system and, therefore, will rely on the MS system coupled with the MC to contain MSIV leakage, relying on plateout and holdup of radioactive iodine and to limit the radiological consequences to within 10 CFR Part 100 limits. In the FSER, the staff stated that in order to take credit for the MSSS and MC for containment and holdup of MSIV leakage, the MSSS and the MC and connections from the MSLs to the condenser must be capable of maintaining their integrity during and following an SSE. Subsequently, GE provided Subsection 3.2.5.3 to the SSAR which clarified the seismic requirements for the MSL leakage paths. Section 3.2 of this report contains a detailed evaluation of the seismic analysis requirements for the MSSS and the MC. To process the MSIV leakage through the MC, a leakage path must be assured either through the MS drain line to the condenser or through the turbine bypass system (TBS) to the condenser. In the FSER, the staff stated that for whichever of these two paths is chosen, reliable power sources must be available so that a control operator can establish the flow path assuming a single-active failure. The staff stated they would review this on a case-by- case basis for each ABWR COL applicant.

In the FSER, the staff stated that the amount of allowable MSIV leakage will also be reviewed for each ABWR COL applicant. This was identified as DFSER COL Action Item 10.3.1-1. Subsequently, the staff has reviewed Amendment 31 of the SSAR and concluded that the modifications to SSAR Subsections 10.3.2 and 10.3.7 will provide adequate assurance that the COL applicant referencing the ABWR design will provide the amount of allowable MSIV leakage for review by the staff. GE has also included this action item in the SSAR.

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Potential Impact Number: 23667 **SRP Section Number:** 16.0

Document: NUREG 1430
Title: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS

Summary:

In their Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors (58 FR 39132), the Commission encouraged licensees to implement a voluntary program to update their Technical Specifications to be consistent with improved vendor-specific Standard Technical Specifications (STS) issued by the NRC in September, 1992. This potential impact is related to the STS for B&W reactors (microfiche location of volumes - 63882:002, 63881:020, and 63880:001).

Potential Impact Number: 23668 **SRP Section Number:** 16.0

Document: NUREG 1431
Title: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS

Summary:

In their Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors (58 FR 39132), the Commission encouraged licensees to implement a voluntary program to update their Technical Specifications to be consistent with improved vendor-specific Standard Technical Specifications (STS) issued by the NRC in September, 1992. This potential impact is related to the STS for Westinghouse reactors (microfiche location of volumes - 63886:001, 63885:001, and 63883:001).

Potential Impact Number: 23669 **SRP Section Number:** 16.0

Document: NUREG 1432
Title: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS

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Summary:

In their Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors (58 FR 39132), the Commission encouraged licensees to implement a voluntary program to update their Technical Specifications to be consistent with improved vendor-specific Standard Technical Specifications (STS) issued by the NRC in September, 1992. This potential impact is related to the STS for Combustion Engineering reactors (microfiche location of volumes - 63961:271, 63965:246, and 63964:164).

Potential Impact Number: 23670 **SRP Section Number:** 16.0

Document: NUREG 1433
Title: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/4

Summary:

In their Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors (58 FR 39132), the Commission encouraged licensees to implement a voluntary program to update their Technical Specifications to be consistent with improved vendor-specific Standard Technical Specifications (STS) issued by the NRC in September, 1992. This potential impact is related to the STS for General Electric BWR/4 reactors (microfiche location of volumes - 63895:001, 63909:001, and 63910:001).

Potential Impact Number: 23671 **SRP Section Number:** 16.0

Document: NUREG 1434
Title: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6

Summary:

In their Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors (58 FR 39132), the Commission encouraged licensees to implement a voluntary program to update their Technical Specifications to be consistent with

improved vendor-specific Standard Technical Specifications (STS) issued by the NRC in September, 1992. This potential impact is related to the STS for General Electric BWR/6 reactors (microfiche location of volumes - 63977:234, 63979:001, and 63980:002).

Potential Impact Number: 23673 **SRP Section Number:** 14.2

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

Issue I.C.7: Operating Procedures - NSSS Vendor Review of Procedures

This issue requires that applicants for near-term operating licenses (NTOLs) obtain the NSSS vendor's review of the low-power and power-ascension test and emergency procedures to further verify the adequacy of the procedures.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

Potential Impact Number: 23674 **SRP Section Number:** 14.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

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EPRI proposed resolution of Issue B-56, "Diesel Reliability": The NRC is currently in the process of upgrading Regulatory Guide 1.9 to incorporate improvements in diesel reliability programs. Regulatory Guide 1.108 will be withdrawn, and revisions to SRP sections and Technical Specifications are being prepared. NUMARC [Nuclear Management and Resources Council] plans to revise Appendix D of NUMARC-87-00 to assure consistency with the emergency diesel generator reliability program described in Regulatory Guide 1.9.

The ALWR is committed to improving the reliability of their EDG units over current experience and have required a minimum target reliability of 0.98. This will be achieved by only using EDG units, including auxiliary and support and control systems, that have demonstrated this reliability or, for new designs, are qualified per the latest revisions of ANSI/IEEE Standard 387 and IEEE Standard 323. Additionally, the reliability of the EDGs will be improved by a design that allows longer start times, thus eliminating problems which adversely affect reliability. Finally, the EDGs will be backed up by combustion turbines which must also have a target reliability of 0.98.

Staff Evaluation: EPRI identifies a minimum EDG target reliability of 0.98 and a backup from combustion turbines having a target reliability of 0.98. However, targets alone do not ensure maintenance of acceptable reliability levels.

In the course of resolving Issue B-56, the Commission recently directed the staff to amend 10 CFR 50.63 to require monthly EDG surveillance testing and reporting of failures to start and load runs against criteria developed to identify potential degradation of EDG performance. This performance-based rule (when issued) would apply to LWRs. Comparable performance monitoring criteria would apply to ALWRs. Since the Evolutionary Requirements Document does not provide sufficient information to enable the staff to fully evaluate the design requirements related to this generic issue, the staff will evaluate the design features for acceptability when it reviews individual applications for FDA/DC. The staff will expect the FDA/DC applicant to address the resolution of Issue B-56 for staff evaluation.

Potential Impact Number: 23682 **SRP Section Number:** 15.6.5.A

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

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EPRI originally took the position that a decontamination factor for radionuclide scrubbing by BWR suppression pools should be credited on the basis of evaluations of accident sequences using the MAAP code and decontamination factors determined by SUPRA, SPARC, or other technically defensible methods.

RG 1.3 (Position C.1.f) allows no credit for retention of iodine by a BWR suppression pool. However, as discussed in Section C.2.3 of the DSER for Chapter 5, the staff has issued a revised SRP Section 6.5.5, "Pressure Suppression Pools as Fission Product Cleanup Systems," dated December 1988. The essential feature of this revised section is that it recognizes that suppression pools are capable of scrubbing non-noble gas fission products and it would be an undue conservatism to ignore this capability. On the basis of the revised SRP Section 6.5.5, the staff concluded that credit may be given for suppression pool fission product removal, provided suppression pool decontamination factors are evaluated in accordance with the methodology prescribed in the revised SRP section.

Potential Impact Number: 23689 **SRP Section Number:** 12.1

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION

Summary:

10 CFR Part 20 was revised with the following sections and appendices being removed: sections 20.1 to 20.601, appendices A,B,C, and D for sections 20.1 to 20.601, and 20.1008

Potential Impact Number: 23690 **SRP Section Number:** 12.3 - 12.4

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION

Summary:

10 CFR Part 20 was revised with the following sections and appendices being removed: sections 20.1 to 20.601, appendices A,B,C, and D for sections 20.1 to 20.601, and 20.1008.

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The federal register notices associated with these revisions to the CFRs are: 56 FR 61352, 57 FR 38588, 57 FR 57877, 58 FR 67657, and 59 FR 1900.

Potential Impact Number: 23691 **SRP Section Number:** 12.5

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION

Summary:

10 CFR Part 20 was revised with the following sections and appendices being removed: sections 20.1 to 20.601, appendices A,B,C, and D for sections 20.1 to 20.601, and 20.1008.

The federal register notices associated with these revisions to the CFRs are: 56 FR 61352, 57 FR 38588, 57 FR 57877, 58 FR 67657, and 59 FR 1900.

Potential Impact Number: 23692 **SRP Section Number:** 9.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.34(f)(2)(viii)

Summary:

In this revision of the CFRs the limit for radiation exposure to the extremities of 75 rem was changed. The limit due to operation of the post-accident sampling systems is now 50 rem to the extremities.

Potential Impact Number: 23694 **SRP Section Number:** 15.6.5.B

Document: ADVANCE SER CE CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

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Summary:

In a letter on October 19, 1992, the staff noted that, with the issuance of (draft) NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" for public comment, the staff believes that future advanced light water reactors (ALWRs) should utilize the revised source term since it reflects the most current U.S. Nuclear Regulatory Commission's understanding of fission product behavior following severe reactor accidents.

The staff has determined that in its evaluation of the evolutionary designs, the current insights from source term research as described in draft NUREG-1465 regarding fission product releases into the containment would be utilized.

In draft NUREG-1465, 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 (I2) and hydrogen iodide (HI). Once within the containment, highly soluble cesium iodide will readily dissolve in water pools forming iodide (I-) in solution and deposit onto the interior surfaces. The staff also stated in NUREG-1465 that the radiation-induced conversion of iodide (I-) in water into elemental iodine (I2) is strongly 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. The staff agrees with this requirement.

Potential Impact Number: 23695 **SRP Section Number:** 15.6.5.B

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Revised SRP Section 6.5.2 states that for sump solutions having a pH less than 7, molecular iodine should be conservatively assumed to evolve into the containment

atmosphere. In Section C.2.1 of the DSER for Chapter 5, the staff stated that EPRI should include a requirement that the ALWR fission product cleanup analyses reflect the effect of reduced pH consistent with the methodology in revised SRP Section 6.5.2 and identified this as an open issue. The staff will evaluate this matter during its review of an individual application for FDA/DC to ensure that the effect of reduced pH is properly reflected in the cleanup analysis.

The staff notes that there are large uncertainties with regard to gaseous iodine production due to pH changes in the in-containment refueling water storage tank, hydrogen burns, and irradiation of other chemical forms of iodine.

Potential Impact Number: 23696 **SRP Section Number:** 6.5.2

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

The staff has determined that in its evaluation of the evolutionary designs, the current insights from source-term research as described in draft NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," June 1992, regarding fission-product releases into the containment would be utilized. ABB-CE has implemented the new source term technology summarized in Draft NUREG-1465, and not the old TID 14844 source term cited in the regulations. The NRC staff has encouraged the development and implementation of the new source term technology and, concurs with ABB-CE's approach.

In implementation of the revised accident source term for evolutionary reactor designs, the staff approached the removal of airborne fission-products in particulate form by spray in an entirely different way from that of ABB-CE. The staff developed a mechanistic and simplistic model that can be used to estimate aerosol removal by sprays without the necessity of using detailed systems codes such as NAUA-4 or CONTAIN. It is described in detail in NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays", (June 1993).

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

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as documented in Chapter 15, Appendix A, Section 15.A.10 of the ABB-CE System 80+ FSER.

Potential Impact Number: 23697 **SRP Section Number:** 8.2

Document: SECY 91-078
Title: CHAPTER 11 OF THE ELECTRIC POWER RESEARCH INSTITUTE'S (EPRI'S) REQUIREMENTS DOCUMENT AND ADDITIONAL EVOLUTIONARY LIGHT WATER REACTOR (LWR) CERTIFICATION ISSUES

Summary:

Describes new staff position on the need for an alternate power source to nonsafety loads such as reactor coolant pumps and main feedwater pumps.

Potential Impact Number: 23698 **SRP Section Number:** 8.2

Document: SECY 91-078
Title: CHAPTER 11 OF THE ELECTRIC POWER RESEARCH INSTITUTE'S (EPRI'S) REQUIREMENTS DOCUMENT AND ADDITIONAL EVOLUTIONARY LIGHT WATER REACTOR (LWR) CERTIFICATION ISSUES

Summary:

Describes new staff position on connection of offsite power sources to the safety buses.

Potential Impact Number: 23701 **SRP Section Number:** 9.5.1

Document: NRC GENERIC LETTER 86-10 SUP 1
Title: FIRE ENDURANCE TEST ACCEPTANCE CRITERIA FOR FIRE BARRIER SYSTEMS USED TO SEPARATE REDUNDANT SAFE SHUTDOWN TRAINS WITHIN THE SAME FIRE AREA

Summary:

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The NRC issued this supplement to disseminate the review guidance contained in Enclosure 1, "Fire Endurance Test Acceptance Criteria for Fire Barriers Used to Separate Redundant Safe Shutdown Trains Within the Same Fire Area." This guidance will be used by the staff to review and evaluate the adequacy of fire endurance tests and fire barrier systems proposed by licensees or applicants in the future to satisfy existing NRC fire protection rules and regulations. This guidance refines and clarifies the fire barrier testing acceptance criteria specified by generic letter 86-10, for application in that specific (future review) context.

Potential Impact Number: 23704 **SRP Section Number:** 15.6.1

Document: NRC GENERIC LETTER 83-10A
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

We have completed our evaluation of your analyses of LOFT Test L3-6 and conclude that the evaluations acceptably predict the test results. Therefore, we find the currently approved B&W evaluation model for small break LOCAs in continued conformance with Appendix K to 10 CFR 50 for the case of limited RCP operation after reactor trip and for the range of licensed B&W reactor designs.

We have reviewed industry analyses and performed our own analyses to determine whether RCP trip is necessary during LOCAs, and evaluated the desirability of continued RCP operation during non-LOCA transients and accidents, including steam generator tube ruptures. We have 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, some of the calculations show that for certain small break LOCAs, primarily those with only one of the two High Pressure Safety Injection (HPSI) Pumps assumed available, continued operation of the RCPs or continued operation of the RCPs followed by delayed RCP trip could lead to core damage.

In summary, we have concluded that the need for RCP trip following a transient or accident should be determined by each licensee on a case-by-case basis, considering the Owners Group input. However, the staff must 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 licensees to comply with the Commission's rules and regulations.

We recognize that possible differences exist between the requirements of 10 CFR 50.46, which assure ample core cooling capacity, and the approaches described in the enclosure which are based upon assuring proper operator/ system response under conditions that may be faced during accidents and transients. Accordingly, in such cases, we will consider a request for exemption from specific requirements of 10 CFR 50.46 pursuant to 10 CFR 50.12.

For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual licensees, we strongly urge that licensees work collectively with owners of similar plants (i.e., owners Group) and propose setpoints and methods consistent with other licensees.

However, once the technical bases for the decision are established, we intend to conduct inspections of individual licensees led by Regional personnel. During these inspections, we will examine the translation of the 10 CFR 50, Appendix K, and RCP operation mode evaluations into plant procedures.

Potential Impact Number: 23705 **SRP Section Number:** 15.6.1

Document: NRC GENERIC LETTER 83-10B
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

This letter provides guidance for a Combustion Engineering NSSS for the development of either satisfactory setpoints for RCP trip or the technical bases for allowing continued RCP operation in the event of a SBLOCA.

Potential Impact Number: 23706 **SRP Section Number:** 15.6.1

Document: NRC GENERIC LETTER 83-10C
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS' (GENERIC LETTER NO. 83-l0c)

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Summary:

This letter provides guidance for a Westinghouse NSSS for the development of either satisfactory setpoints for RCP trip or the technical bases for allowing continued RCP operation in the event of a SBLOCA.

Potential Impact Number: 23707 **SRP Section Number:** 15.6.1

Document: NRC GENERIC LETTER 83-10D
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

We have completed our evaluation of your analyses of LOFT Test L3-6 and conclude that the evaluations acceptably predict the test results. Therefore, we find the currently approved B&W evaluation model for small break LOCAs in continued conformance with Appendix K to 10 CFR 50 for the case of limited RCP operation after reactor trip and for the range of licensed B&W reactor designs.

We have reviewed industry analyses and performed our own analyses to determine whether RCP trip is necessary during LOCAs, and evaluated the desirability of continued RCP operation during non-LOCA transients and accidents, including steam generator tube ruptures. We have 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, some of the calculations show that for certain small break LOCAs, primarily those with only one of the two High Pressure Safety Injection (HPSI) Pumps assumed available, continued operation of the RCPs or continued operation of the RCPs followed by delayed RCP trip could lead to core damage.

In summary, we have concluded that the need for RCP trip following a transient or accident should be determined by each licensee on a case-by-case basis, considering the Owners Group input. However, the staff must 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 licensees to comply with the Commission's rules and regulations.

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We recognize that possible differences exist between the requirements of 10 CFR 50.46, which assure ample core cooling capacity, and the approaches described in the enclosure which are based upon assuring proper operator/ system response under conditions that may be faced during accidents and transients. Accordingly, in such cases, we will consider a request for exemption from specific requirements of 10 CFR 50.46 pursuant to 10 CFR 50.12.

For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual licensees, we strongly urge that licensees work collectively with owners of similar plants (i.e., owners Group) and propose setpoints and methods consistent with other licensees.

However, once the technical bases for the decision are established, we intend to conduct inspections of individual licensees led by Regional personnel. During these inspections, we will examine the translation of the 10 CFR 50, Appendix K, and RCP operation mode evaluations into plant procedures.

Potential Impact Number: 23708 **SRP Section Number:** 15.6.1

Document: NRC GENERIC LETTER 83-10E
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

We have completed our evaluation of your analyses of LOFT Test L3-6 and conclude that the evaluations acceptably predict the test results. Therefore, we find the currently approved B&W evaluation model for small break LOCAs in continued conformance with Appendix K to 10 CFR 50 for the case of limited RCP operation after reactor trip and for the range of licensed B&W reactor designs.

We have reviewed industry analyses and performed our own analyses to determine whether RCP trip is necessary during LOCAs, and evaluated the desirability of continued RCP operation during non-LOCA transients and accidents, including steam generator tube ruptures. We have 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, some of the calculations show that for certain small break LOCAs, primarily those with only one of the two High Pressure Safety Injection (HPSI) Pumps assumed available, continued operation of the

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RCPs or continued operation of the RCPs followed by delayed RCP trip could lead to core damage.

In summary, we have concluded that the need for RCP trip following a transient or accident should be determined by each licensee on a case-by-case basis, considering the Owners Group input. However, the staff must 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 licensees to comply with the Commission's rules and regulations.

We recognize that possible differences exist between the requirements of 10 CFR 50.46, which assure ample core cooling capacity, and the approaches described in the enclosure which are based upon assuring proper operator/ system response under conditions that may be faced during accidents and transients. Accordingly, in such cases, we will consider a request for exemption from specific requirements of 10 CFR 50.46 pursuant to 10 CFR 50.12.

For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual licensees, we strongly urge that licensees work collectively with owners of similar plants (i.e., owners Group) and propose setpoints and methods consistent with other licensees.

However, once the technical bases for the decision are established, we intend to conduct inspections of individual licensees led by Regional personnel. During these inspections, we will examine the translation of the 10 CFR 50, Appendix K, and RCP operation mode evaluations into plant procedures.

Potential Impact Number: 23709 **SRP Section Number:** 15.6.1

Document: NRC GENERIC LETTER 83-10F
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

This letter provides guidance for a Babcock and Wilcox NSSS for the development of either satisfactory setpoints for RCP trip or the technical bases for allowing continued RCP operation in the event of a SBLOCA.

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Potential Impact Number: 23710 **SRP Section Number:** 15.6.1

Document: NRC GENERIC LETTER 85-12
Title: IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

The purpose of this letter is to inform you of (1) the staff's conclusions regarding the Westinghouse Owners Group (WOG) submittals on reactor coolant pump trip in response to Generic Letters 83-10c and d. and (2) provide guidance concerning implementation of the reactor coolant pump trip criteria. Our Safety Evaluation (SE) on this subject is enclosed for your use.

With regard to the WOG submittals referenced in Section V of the enclosed SE, we conclude that the methods employed by the WOG to justify manual reactor coolant pump (RCP) trip are consistent with the guidelines and criteria provided in Generic Letters 83-10c and d. The approved Westinghouse Small Break LOCA Evaluation Model was used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

We have determined that the Information provided by the WOG in Support of the alternative RCP trip criteria is acceptable on a generic basis. A suitable reactor coolant pump trip criterion can be selected by each licensee to minimize reactor coolant pump trip during steam generator tube ruptures and non-LOCA events, while still providing for RCP trip for small break LOCAs.

In reviewing the WOG RCP trip Criteria we note that the process of criterion selection involves a number of considerations which were assigned plant-specific status by the WOG during the process of the trip criteria review. Accordingly, we request that operating reactor licensees select and implement an appropriate RCP trip criterion based upon the WOG methodology.

Those applicants and licensees who choose not to endorse the WOG methodology should submit a schedule for submittal of plant specific RCP trip criteria or justification for non-trip of RCPs within 45 days of receipt of this letter. .

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Potential Impact Number: 23711 **SRP Section Number:** 15.6.1

Document: NRC GENERIC LETTER 86-05
Title: IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

The purpose of this letter is to inform you of (1) the staff's conclusions regarding the B&W Owners Group (BWOG) submittals on reactor coolant pump trip in response to Generic Letters 83-10e and f, and (2) provide guidance concerning implementation of the reactor coolant pump trip criterion. Our Safety Evaluation (SE) on this subject is enclosed for your use.

With regard to the BWOG submittals referenced in Section V of the enclosed SE, we conclude that the methods employed by the BWOG to justify manual reactor coolant pump (RCP) trip are consistent with the guidelines and criteria provided in Generic Letters 83-10e and f. The approved B&W Small Break LOCA Evaluation Model was used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

We have determined that the information provided by the BWOG in support of the loss-of-subcooling RCP trip criterion is acceptable. The generic information presented by the BWOG, however, does not address plant specific concerns about instrumentation uncertainties, potential reactor coolant pump problems and operator training and procedures as requested in Generic Letter 83-10. This information, contained in Section IV of the SE, is now being requested to assess implementation of the RCP trip criterion. Accordingly, for those applicants and licensees who choose to endorse the BWOG methodology, we request that operating reactor licensees implement the RCP trip criterion based upon the BWOG methodology.

Those applicants and licensees who choose not to endorse the BWOG methodology should submit a schedule for submittal of plant specific RCP trip criteria or justification for non-trip of RCPs within 45 days of receipt of this letter.

Potential Impact Number: 23712 **SRP Section Number:** 15.6.1

Document: NRC GENERIC LETTER 86-06

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Title: IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

The purpose of this letter is to inform you of (l) the staff's conclusions regarding the CE Owners Group (CEOG) submittals on reactor coolant pump trip in response to Generic Letters 83-10a and b, and (2) provide guidance concerning implementation of the reactor coolant pump trip criterion. Our Safety Evaluation (SE) on this subject is enclosed for your use

With regard to the CEOG submittals referenced in Section V of the enclosed SE, we conclude that the methods employed by the CEOG to justify manual reactor coolant pump (RCP) trip are consistent with the guidelines and criteria provided in Generic Letters 83-10a and b. The approved CE Small Break LOCA Evaluation Model was used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

We have determined that the information provided by the CEOG in support of the trip-two/leave-two staggered reactor coolant pump trip criterion is acceptable. The generic information presented by the CEOG, however, does not address plant specific concerns about instrumentation uncertainties, potential reactor coolant pump problems and operator training and procedures as requested in Generic Letter 83-10. This information, contained in Section IV of the SE, is now being requested to assess implementation of the RCP trip criterion. Accordingly, for those applicants and licensees who choose to endorse the CEOG methodology, we request that operating reactor licensees implement the RCP trip criterion based upon the CEOG methodology.

Those applicants and licensees who choose not to endorse the CEOG methodology should submit a schedule for submittal of plant specific RCP trip criteria or justification for non-trip of RCPs within 45 days of receipt of this letter.

Potential Impact Number: 23725 **SRP Section Number:** 10.2.3

Document: C&S: ASTM E208
Title: Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels

Summary:

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ASTM E-208 is referenced as guidance in Acceptance Criteria related to testing the nil-ductility transition temperature of turbine disk material. The current version of this standard is ASTM E-208, 1991.

Potential Impact Number: 23726 **SRP Section Number:** 10.2.3

Document: C&S: ASTM A370
Title: Standard Test Methods and Definitions for Mechanical Testing of Steel Products

Summary:

ASTM A-370 is referenced as guidance in Acceptance Criteria related to mechanical testing of turbine disk material. The current version of this standard is ASME SA-370, 1992.

Potential Impact Number: 23727 **SRP Section Number:** 10.2.3

Document: FINAL SER EPRI CH 13
Title: EPRI URD-EPD: MAIN TURBINE-GENERATOR SYSTEMS

Summary:

Chapter 13 of the Evolutionary Requirements Document proposes changes to the design and operation of the main turbine that will improve the reliability and availability of the unit. 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 has incorporated this solution into Section 3.3.1.1.1.11 of Chapter 13 by requiring a one-piece rotor using either an integral forging or welded design. Because SRP Section 10.2.3, "Turbine Disk Integrity," gives specific guidelines on fracture toughness, material selection, and inservice inspection primarily for shrunk-on disk designs, and a one-piece disk design will be used in the ALWR, SRP Section 10.2.3 does not apply. Additionally, although not actually specifying materials, EPRI requires that materials, fabrication techniques, and coating processes that have proved to increase resistance to intergranular stress corrosion cracking and to other erosion and corrosion will be used in the designs of rotors, nozzles, and blading. This must be balanced by the requirements proposed by EPRI that the materials selected can be repaired in the field.

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In the DSER for Chapter 13, the staff concluded that the use of a one-piece rotor may require that additional emphasis be placed on post-machining inspection and that the Evolutionary Requirements Document should address this matter. This was identified as an open issue.

EPRI stated that the turbine manufacturer routinely includes detailed and documented nondestructive examinations of rotors (and discs, whether separate or not) at various stages in the fabrication process. Provisions for in-process fabrication inspections are not considered to be safety-related issues. Because turbine inspection is beyond the scope of the Evolutionary Requirements Document, the staff will review applications for FDA/DC or a combined license to ensure that applicants provide requirements for stringent post-machining inspections to the NRC staff for review and approval before plant operation. Therefore, this DSER open issue is closed.

Potential Impact Number: 23731 **SRP Section Number:** 10.2.3

Document: ADVANCE SER ABWR CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

The turbine rotor design will be a solid forged monoblock rotor rather than one with shrunk-on disks. The current practice employed by the turbine manufacturers is to bore the center of the monoblock rotor to remove metal impurity and permit internal inspection. A forged or welded rotor will not be as susceptible to stress corrosion cracking as experienced in the shrunk-on disks. However, the one-piece rotor design requires stringent part-machining inspections. Therefore, the applicant referencing the ABWR design must submit inspection requirements for a one-piece rotor to the NRC staff for review and approval before plant operation. Further, the applicant must submit an actual turbine inspection schedule following the third refueling outage. The actual turbine inspection schedule should be based on a probability calculation of turbine missile generation.

Potential Impact Number: 23732 **SRP Section Number:** 10.2.3

Document: ADVANCE SER CE CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

APPENDIX II
Potential Impacts

Summary:

Each finish machined forging is subjected to 100-percent volumetric (ultra-sonic), surface, and visual examinations using the manufacturer's acceptance criteria. In Amendment T, ABB-CE committed to perform surface crack testing on all parts critical to rotor integrity, such as weld inspection holes, balancing holes, and geometric transitions. The disc is not keyed to the rotor in this design and high stress regions are limited in location, such that these commitments should provide assurance that inservice inspection of all high stress regions is facilitated. On this basis, DSER Open Items 10.2-5 and 10.2-6 are resolved.

In Amendment N, ABB-CE modified the CESSAR-DC to describe its steam turbine design. Unlike most domestic nuclear steam turbine rotor designs, it uses neither a shrunk-on disc design nor a one-piece forged rotor design. Instead, barrel construction is utilized which consists of a series of welded ring forgings. This design is in effect a one-piece, welded, hollow rotor design, with the end forgings having shaft protrusions. Each forging is chemically analyzed, mechanically tested for strength and toughness, and has an upset ratio of at least two. Each is separately heat treated and inspected prior to the final welding process.

Forgings to be used in the welded rotor are rough machined with minimum stock allowance prior to heat treatment. All individual rough machined rotor forgings to be used for the welded rotor are subjected to 100-percent volumetric (ultrasonic) examination. Due to oversize of each forging, indications found during the ultrasonic testing may be removed during final machining of the forging. The manufacturer's acceptance criteria includes the requirement that subsurface sonic indications are evaluated to assure that they will not grow to a size which compromises the integrity of the unit during the service life. Prior to welding, all surfaces which would not be accessible after welding will be surface examined. After welding and final machining, all surfaces exposed to steam will be surface examined, with particular attention placed on the areas of stress risers and welds. The welds will be ultrasonically examined in the radial and radial-tangential sound beam directions. On this basis, DSER Open Item 10.2-7 is resolved.

Potential Impact Number: 23733 **SRP Section Number:** 2.4.13

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION

APPENDIX II
Potential Impacts

Summary:

10 CFR Part 20 was revised with the following sections and appendices being removed: Sections 20.1 to 20.601; Appendices A, B, and D for Sections 20.1 to 20.601, and 20.1008.

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 Impact Number: 23737 **SRP Section Number:** 11.2

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION

Summary:

10 CFR Part 20 was revised with the following sections and appendices being removed: sections 20.1 to 20.601, appendices A, B, C, and D for sections 20.1 to 20.601, and 20.1008.

Potential Impact Number: 23738 **SRP Section Number:** 11.3

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION

Summary:

10 CFR Part 20 was revised with the following sections and appendices being removed: sections 20.1 to 20.601, appendices A, B, C, and D for sections 20.1 to 20.601, and 20.1008.

Potential Impact Number: 23739 **SRP Section Number:** 11.4

APPENDIX II
Potential Impacts

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION

Summary:

10 CFR Part 20 was revised with the following sections and appendices being removed: sections 20.1 to 20.601, appendices A, B, C, and D for sections 20.1 to 20.601, and 20.1008.

Potential Impact Number: 23740 **SRP Section Number:** 11.5

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION

Summary:

10 CFR Part 20 was revised with the following sections and appendices being removed: sections 20.1 to 20.601, appendices A, B, C, and D for sections 20.1 to 20.601, and 20.1008.

Potential Impact Number: 23741 **SRP Section Number:** 12.2

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION

Summary:

10 CFR Part 20 was revised with the following sections and appendices being removed: sections 20.1 to 20.601, appendices A, B, C, and D for sections 20.1 to 20.601, and 20.1008.

Potential Impact Number: 23752 **SRP Section Number:** 3.6.1

Document: ADVANCE SER ABWR CH 3

APPENDIX II
Potential Impacts

Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The application of the LBB (leak-before-break) methodology to piping systems is permitted by GDC 4.

The analyses for LBB referred to in GDC 4 should be based on such plant-specific data as piping geometry, materials, piping loads and pipe support locations. The staff must review the LBB analyses for specific piping designs before an applicant can exclude the dynamic effects from the design basis for the piping system.

In the ABWR DFSER, the staff noted that COL applicants seeking approval of the LBB approach for high-energy piping systems in the ABWR plant should submit an LBB plant-specific analysis in accordance with GDC 4 and include certain identified information. GE addressed this item in SSAR Section 3.6.5.2 by stating that the COL applicant shall prepare a plant-specific LBB analysis report and submit the report to the NRC staff for approval. This is acceptable.

Although the staff is currently using the methodology and acceptance criteria provided in SRP Section 3.6.3 and NUREG-1061, Volume 3, the staff recognizes that the LBB technology is continually evolving. Therefore, the staff will review LBB requests for the ABWR plant on a case-by-case basis using the staff's methodology and acceptance criteria in effect at the time of the submittal.

Potential Impact Number: 23753 **SRP Section Number:** 3.6.1

Document: ADVANCE SER CE CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

Under the broad-scope revision to GDC 4, the NRC allows the use of advanced technology to exclude from structural design consideration the dynamic effects of pipe ruptures in nuclear power plants, provided it is demonstrated that the probability of pipe rupture is extremely low under conditions consistent with the design bases for the piping. The demonstration of low probability of pipe rupture utilizes a deterministic

fracture mechanics analysis that evaluates the stability of postulated, small, through-wall flaws in piping and the ability to detect leakage through the flaws long before the flaw could grow to unstable sizes. The concept underlying such analyses is referred to as LBB.

The limitations and acceptance criteria for LBB used by the NRC staff are discussed in NUREG-1061, Volume 3, "Evaluation of Potential for Pipe Breaks, Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," dated November 1984.

The application of the LBB approach to ALWRs seeking design certification under 10 CFR Part 52 is acceptable when appropriate bounding limits are established during the design certification phase using preliminary analyses results and verified during the combined license phase by the COL applicant performing the appropriate ITAAC discussed herein. Bounding analyses as described above must be performed by ABB-CE requesting the application of LBB during the design certification phase. NUREG-1061, Volume 3, constitutes the current NRC-approved LBB methodology and acceptance criteria.

Potential Impact Number: 23754 **SRP Section Number:** 3.6.1

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

The revised GDC 4 of Appendix A to 10 CFR Part 50 permits elimination of the dynamic effects of postulated high-energy pipe ruptures from the design basis of ALWRs using advanced fracture mechanics analyses. The limitations and acceptance criteria for leak-before-break (LBB) applications in ALWRs are the same as those established for currently operating plants.

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. This approval should 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 ITAAC. In the SRM for SECY 93-087 the Commission approved the staff's position on LBB for the ALWRs.

Potential Impact Number: 23755 **SRP Section Number:** 3.6.1

Document: ADVANCE SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

GE evaluated the effects of postulated pipe breaks in high-energy fluid systems with regard to pipe whip, jet impingement, flooding, room pressurization, and environmental parameters such as temperature, pressure, humidity, and radiation. However, the staff stated in the DSER (SECY-91-153) that GE excluded consideration of pipe breaks and the resulting dynamic effects in the postulation of piping failures in main steam and feedwater systems. GE justified the exclusion stating that the piping in these systems met the LBB criterion. The staff expects that a bona fide LBB analysis should use plant- specific data such as piping geometry, materials, fabrication procedures, loads, degradation mechanisms, and pipe support locations. In its evaluation of the LBB exclusion, which is discussed in Section 3.6.3 of this report, the staff concluded that LBB could not be considered in the analysis of pipe failures. GE subsequently provided an analysis of a MSL and a main feedwater line pipe failure in SSAR Section 6.2.3; the staff's review of this analysis is found in Section 6.2.1.7 of this report. In a meeting with the staff, GE committed to removing references to LBB from the SSAR. This was identified as Confirmatory Item 3.4.1-1 in the DFSER and is discussed in Section 3.4.1 of this report.

Potential Impact Number: 23756 **SRP Section Number:** 3.6.2

Document: ADVANCE SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

In SECY-93-087, the staff examined the safety impact of eliminating the OBE as it pertains to civil structures, piping systems, and equipment seismic qualification. Several recommendations were made by the staff to ensure that eliminating the OBE

would not result in a significant decrease in the overall plant safety margin. In its July 21, 1993, SRM, the Commission approved and agreed with the staff's positions, including that 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 guidelines provided in SRP Section 3.6.2, Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," have been revised for the ABWR as follows:

B.1.b.(1).(a): Footnote 2 should read, "For those loads and conditions in which Level A and Level B stress limits have been specified in the Design Specification (excluding earthquake loads)."

B.1.b.(1).(d): "The maximum stress as calculated by the sum of Eqs. (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering those loads and conditions thereof for which level A and level B stress limits have been specified in the system's Design Specification (i.e., sustained loads, occasional loads, and thermal expansion) excluding earthquake loads should not exceed 0.8(1.8 Sh + SA)."

Potential Impact Number: 23757 **SRP Section Number:** 3.6.2

Document: ADVANCE SER CE CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

In SECY-93-087, the staff examined the safety impact of eliminating the OBE as it pertains to civil structures, piping systems, and equipment seismic qualification. Several recommendations were made by the staff to ensure that eliminating the OBE would not result in a significant decrease in the overall plant safety margin. In its July 21, 1993, SRM, the Commission approved and agreed with the following staff's positions, including that 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.

APPENDIX II
Potential Impacts

The guidelines provided in SRP Section 3.6.2, Branch Technical Position Mechanical Engineering Branch (MEB) 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," have been revised for the System 80+ as follows:

B.1.b.(1).(a):Footnote 2 should read, "For those loads and conditions in which Level A and Level B stress limits have been specified in the Design Specification (excluding earthquake loads)."

B.1.b.(1).(d):"The maximum stress as calculated by the sum of Eqs. (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering those loads and conditions thereof for which level A and level B stress limits have been specified in the system's Design Specification (i.e., sustained loads, occasional loads, and thermal expansion) excluding earthquake loads should not exceed 0.8(1.8 Sh + SA)."

Potential Impact Number: 23764 **SRP Section Number:** 6.2.2

Document: NRC BULLETIN 93-002
Title: DEBRIS PLUGGING OF EMERGENCY CORE COOLING SUCTION STRAINERS

Summary:

This bulletin notifies licensees of a previously unrecognized contributor to the potential loss of net positive suction head (NPSH) for the Emergency Core Cooling Systems (ECCS) for Light Water Reactors during the recirculation phase of a loss-of-coolant accident (LOCA). Regulatory Guide 1.82, Revision 1, "Water Sources for Long-Term Recirculation Cooling Following a Loss of Coolant Accident," discusses, among other things, the blockage of strainers due to fibrous debris.

All holders of operating licenses for nuclear power reactors, were requested to take the following actions: Identify fibrous air filters or other temporary sources of fibrous material, not designed to withstand a LOCA, which are installed or stored in your primary containment. Take any immediate compensatory measures which may be required to assure the functional capability of the ECCS. Take prompt action to remove any such material. Because of the low probability of a LOCA event, the staff considers removal of this material at the next shutdown, or within 120 days, whichever comes first, to be sufficiently prompt. If the facility is currently in a shutdown, you are requested to remove such material prior to restart.

APPENDIX II
Potential Impacts

The actions requested by this bulletin represent a new staff position and are considered necessary to ensure that licensees are in compliance with existing NRC rules and regulations where these conditions are applicable. Therefore, this bulletin is being issued as a compliance backfit.

Potential Impact Number: 23765 **SRP Section Number:** 3.9.6

Document: NRC GENERIC LETTER 89-10 SUP 6
Title: INFORMATION ON SCHEDULE AND GROUPING, AND STAFF RESPONSES TO ADDITIONAL PUBLIC QUESTIONS

Summary:

On February 25, 1993, the NRC staff held a public workshop to discuss GL 89-10 and to answer questions from the public on the inspections of licensee programs developed in response to the GL. In this supplement to the GL, the staff further clarifies the positions on the schedule for completing the MOV testing to verify design-basis capability recommended in GL 89-10 and grouping of MOVs to establish valve setup conditions.

This GL supplement and its enclosure (1) restate staff positions contained in the GL and its earlier supplements, (2) provide additional guidance for meeting the staff positions contained in the generic letter and its earlier supplements, (3) relax the staff position on the need for in situ testing of each MOV, and (4) require the submittal of specific information if a licensee intends to extend its GL 89-10 schedule.

Potential Impact Number: 23766 **SRP Section Number:** 6.2.2

Document: ADVANCE SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

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.

To address this issue for the ABWR, the staff believes 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 sizing margin to account for uncertainty in the synergetic effects of strainer clogging from insulation, corrosion products, and other debris.

Potential Impact Number: 23767 **SRP Section Number:** 5.2.4

Document: NRC GENERIC LETTER 91-17 Revision Date: 10/17/91
Title: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

Generic letter 91-17 provides information on the U.S. Nuclear Regulatory Commission's resolution of Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants." The NRC has resolved this issue without developing any new requirements. The bases for resolution are further documented in the enclosed NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants".

Potential Impact Number: 23772 **SRP Section Number:** 5.2.4

Document: NUREG 0313 **Revision:** 2 , 1/1/88
Title: TECHNICAL REPORT ON MATERIAL SELECTION AND PROCESSING GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING. FINAL REPORT

Summary:

Substantial efforts in research and development have been sponsored by the BWR Owners Group for IGSCC Research. The results of this program, along with other related work by vendors, consulting firms, and confirmatory research sponsored by the NRC, have permitted the development of revised Staff Positions regarding the IGSCC

problems. The technical bases for these positions are detailed in NUREG-0313, Rev. 2 "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." NUREG-0313, Rev. 2 was promulgated by Generic Letter 88-01.

Potential Impact Number: 23773 **SRP Section Number:** 5.2.4

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

Ultrasonic testing of RCPB components will improve in the near future, as indicated by ASME Section XI, Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." The NRC has referenced 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. Therefore, the staff concluded in the DFSER that GE should include in its SSAR PSI program provisions that ultrasonic testing be performed in accordance with Appendices VII and VIII pursuant to 10 CFR 50.55a(g)(3).

Potential Impact Number: 23774 **SRP Section Number:** 5.2.4

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

'The ASME has published ultrasonic testing standards in Section XI, Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." The NRC has approved the 1989 Edition of the ASME Code, Section XI which includes Appendix VII. The applicable Code Edition for the CESSAR-DC is the 1989 Edition. In Amendment N, ABB-CE revised the CESSAR-DC to require that the Section XI requirements are to be augmented with the requirements in Appendix VIII.'

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Potential Impact Number: 23775 **SRP Section Number:** 5.2.4

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Carbon steel piping in which high energy single-phase fluid flows have been observed to experience erosion of its walls over time. As a result of two ruptures of such pipes, the following measures were called for: (1) more stringent inspection of carbon steel piping; and (2) replacement of all piping that does not meet minimum thickness standards. This issue has been resolved.

Potential Impact Number: 23776 **SRP Section Number:** 5.2.4

Document: NRC BULLETIN 87-01
Title: THINNING OF PIPE WALLS IN NUCLEAR POWER PLANTS

Summary:

All licensees have either explicitly or implicitly committed to maintain the functional capability of high-energy piping systems that are a part of the licensing basis for the facility. An important part of this commitment is that piping will be maintained within allowable thickness values. The purpose of this bulletin is to request that licensees submit information concerning their programs for monitoring the thickness of pipe walls in high-energy single-phase and two-phase carbon steel piping systems.

Potential Impact Number: 23777 **SRP Section Number:** 5.2.4

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

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ASME Class 1, 2, and 3 carbon and low alloy steel piping items that are susceptible to wall thinning as a result of the single phase (water) erosion-corrosion phenomenon will be subject to examination in accordance with Subsection IWH of ASME Code Section XI.

As noted in CESSAR-DC Sections 5.2.4 and 6.6.1, although ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWH has not yet been formally approved for use by the nuclear industry, the COL applicant will review Subsection IWH after NRC approval, and ensure that the System 80+ design PSI and ISI programs comply. Where the design will not practically accommodate full adherence to IWH, the COL applicant may propose alternatives as allowed in 10 CFR 50.55a(a)(3)(i) and (ii). This is part of COL Action Item 5-3 as noted in CESSAR-DC Table 1.10-1. On this basis, DSER Open Item 5.2.4-7 is resolved.

Potential Impact Number: 23779 **SRP Section Number:** 5.2.4

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

... However, the staff concluded in the DFSER that the COL applicant must monitor the construction of structural supports around the RPV and the installation of auxiliary equipment to ensure that the design access that will be provided is not adversely affected. This was DFSER COL Action Item 5.2.4-3.

GE responded to this item in SSAR Section 5.2.6.2, which states that the COL applicant will submit the complete plant-specific ISI/PSI program to the NRC including references to the edition and addenda of the ASME Code Section XI that will be used for the selection of components subject to examination, a description of the components exempt from examination by the applicable code, and isometric drawings used for the examination. This is acceptable. GE also stated that the COL applicant will submit plans for preservice examination of the RPV welds to address the degree of compliance with RG 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," Revision 1. Further, the SSAR, Section 5.2.4.2 states that all items within the Class 1 boundary are designed to provide access for the examinations required by ASME Section XI, IWB-2500. This is also acceptable.

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Potential Impacts

Potential Impact Number: 23780 **SRP Section Number:** 3.9.6

Document: NRC GENERIC LETTER 87-06
Title: PERIODIC VERIFICATION OF LEAK TIGHT INTEGRITY OF PRESSURE ISOLATION VALVES (GENERIC LETTER 87-06)

Summary:

Pressure isolation valves (PIVs) are defined for each interface as any two valves in series within the reactor coolant pressure boundary which separate the high pressure reactor coolant system (RCS) from an attached low pressure system.

Periodic tests for verifications of the leak tight integrity of all pressure isolation valves are necessary in order to assure the integrity of the RCPB in accordance with 10 CFR 50, Appendices A and B.

A number of events have occurred in LWRs involving leakage past pressure isolation valves, failures of the valves, inadvertent actuation of the valves, or mispositioning of the valves. NRC is therefore seeking assurances that the RCPB is being protected. Accordingly, you are requested to submit a list of all pressure isolation valves in your plant. For each valve, describe the periodic tests or other measures performed to assure the integrity of the valve as an independent barrier at the RCPB along with the acceptance criteria for leakage, if any, operational limits, if any, and frequency of test performance.

Potential Impact Number: 23781 **SRP Section Number:** 3.9.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

The purpose of this TMI Action Plan item is to establish the adequacy of current requirements for safety-related valve testing. It recommends a study which would result in recommendations for alternate means of verifying performance requirements.

This issue was given a medium priority ranking and was later divided into four parts during resolution: (1) pressure isolation valves; (2) check valves; (3) reevaluation of

thermal-overload protection provisions of Regulatory Guide 1.106 for MOVs; and (4) in-situ testing of MOVs.

The investigation of alternatives to leak rate testing of pressure isolation valves, including check valves, was integrated into the resolution of Issue 105, "Interfacing Systems LOCA." These alternatives include non-intrusive methods to detect check valve disk position and motion, as well as surveillance of internal parts by various means. Any new issue regarding testing of check valves that may be identified in the future will be prioritized as a new generic issue. The results of the staff's study of MOV thermal overload protection were published in NUREG-1296. The staff concluded that, although misinterpreted by the industry at times, the guidelines in Regulatory Guide 1.106 were adequate. Several suggestions for improving MOV thermal overload protection were outlined in NUREG-1296. In addition, letters were sent to the pertinent IEEE and ASME subcommittees encouraging the development of standards for MOV thermal overload protection. In-situ testing and surveillance of check valves is being addressed by an industry effort; in-situ testing of MOVs was resolved with the issuance of Generic Letter 89-10. Thus, this issue was RESOLVED and requirements were established.

Potential Impact Number: 23782 **SRP Section Number:** 3.9.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

PORVs and block valves were originally designed as non-safety components in the reactor pressure control system for use only when plants are in operation. The block valves were installed because of expected leakage from the PORVs. Neither the PORVs nor the block valves were required to safely shut down a plant or mitigate the consequences of accidents. RSB determined in 1983 that PORVs are relied upon to mitigate a design basis SGTR. However, the acceptability of relying on non-safety grade PORVs to mitigate a design basis accident (e.g., an SGTR) was questioned by RSB. This issue was identified in a DL memorandum and addressed the assessment of the need for improving the reliability of PORVs and block valves.

In resolving this issue, the staff issued Generic Letter 90-06 which required TS revisions at PWRs with PORVs and block valves. This resolution represented new staff positions for some licensees and CP holders and was considered a backfit. The

regulatory analysis for the issue was reported in NUREG-1316. In addition, revisions to SRP Sections 3.2.2, 5.2.2, and 5.4.7 were forwarded to NRR for incorporation into NUREG-0800. The Commission was informed of the staff's resolution in SECY-90-153. Thus, this issue was RESOLVED and new requirements were established.

Potential Impact Number: 23783 **SRP Section Number:** 3.9.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue was established to address the uncertainty that BWR HPCI steam supply valves and RWCU valves would close under flow conditions created by a pipe break.

Based on both the RWCU and HPCI event sequences and the Event V consequences, this issue was given a high priority ranking. In resolving the issue, the staff conducted a two-phase valve test program: Phase 1 was reported in NUREG/CR-5406 and Phase 2 was reported in NUREG/CR-5558. In addition, laboratory tests of DC-powered MOVs were conducted and reported in NUREG/CR-5720.

The results of the Phase 1 test program were factored into the development of Generic Letter No. 89-10, thus providing licensees with the best guidance available at that time regarding how they should assure that their MOVs would perform their design basis function. The results of the Phase 2 test program were used in the development of Supplement 3 to Generic Letter No. 89-10, which provided licensees with further guidance. The staff also conducted training for inspectors and provided computer software to aid in identifying these problems on site. Thus, this issue was RESOLVED and requirements were issued to licensees in Generic Letter No. 89-10.

Potential Impact Number: 23786 **SRP Section Number:** 3.9.3

Document: ADVANCE SER CE CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

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Summary:

The staff requested that ABB-CE include the effects of thermal stratification in the design of the surge line. ABB-CE responded by committing to include thermal stratification and thermal striping effects in the surge line analysis. More generally, by NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," the staff is requesting 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 stresses from temperature stratification or temperature oscillations that could be induced by leaking valves. ABB-CE was requested to conform to the guidelines of NRC Bulletins 88-08 and 88-11 for piping systems connected to the RCS.

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. The original issuance of NRC Bulletin 88-08 was in response to the discovery of cracks attributed to high stresses resulting from thermal stratification in the SI nozzle at Farley, Unit 2. The source of this stratification was determined to be the leakage of colder fluid which passed the check valve, isolating the SI line from the RCS cold leg. Subsequently, Bulletin Supplements 1, 2, and 3 made recommendations for inspection and reported on incidents of apparent outleakage causing line failure at a foreign reactor. These supplements broadened the concern for operating plants to all lines in which stratified flow could occur.

NRC Information Notice (IN) 84-87 and NRC IN 91-38 discuss thermal gradient produces pipe deflections, support loads, pipe bending stresses, and local stresses observed in both BWR and PWR feedwater piping. NRC Bulletin 88-11 was issued in response to the results of an inspection of a pressurizer SL at a plant which showed large, unexpected movements that closed the gaps between the line and pipe whip restraints. The movements were attributed to thermal stratification which might have occurred under certain operating conditions when large temperature differences exist between the RCS and the pressurizer. The bulletin required all PWR licensees to establish and implement a program to assure the structural integrity of the SL when subjected to thermal stratification. The structural reevaluation should consider the cyclic effects of the additional bending stresses in the pipe as well as the local stresses induced by thermal striping (rapid oscillation of the thermal boundary interface along the piping inside surface).

The staff asked ABB-CE to describe how the issues of temperature oscillations and thermal stratification will be considered for the System 80+ piping design in the

CESSAR-DC. ABB-CE described the programs sponsored by the owners and operators of ABB-CE operating plants to address Bulletins 88-08 and 88-11. ABB-CE stated that the results obtained from programs are used to evaluate the presence of stratified flow and the influence on thermal stresses in these lines for System 80+ and that all piping systems subjected to stratified flow are evaluated for additional thermal stresses due to thermal stratification..

On the basis of its review of this information, the staff concludes that ABB-CE provided an adequate approach and methodology for resolving Bulletin 88-08, including Supplements 1, 2 and 3, and Bulletin 88-11.

Potential Impact Number: 23787 **SRP Section Number:** 3.9.3

Document: ADVANCE SER CE CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

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 a letter dated September 11, 1992, the staff provided a guidance document which identified the necessary changes to existing seismic design criteria that are acceptable for implementing the proposed rule change as it pertains to the design of safety-related SSCs for ALWRs. This document described specific supplemental criteria for fatigue, SAM, and piping stress limits which should be applied when the OBE is eliminated. For fatigue evaluation, two SSE events with 10 maximum stress cycles per event (or an equivalent number of fractional cycles) should be considered. The effects of SAMs due to the SSE should be considered in combination with the effects of other normal operation loadings that might occur concurrently. For Class 1 primary stress evaluation, seismic loads need not be evaluated for consideration of Level B service limits for Eq. 9. However, for satisfaction of primary plus secondary stress range limits in Eq. 10, the full SSE stress range or a reduced range corresponding to an equivalent number of fractional cycles shall be included for Level B service limits. These load sets should also be used for evaluating fatigue effects. In addition, the stress due to the larger of the full range of SSE SAM or the resultant range of thermal expansion plus half the SSE anchor motion range, shall not exceed 6.0 Sm. This was defined as Eq. 12a. For Class 2 and 3 piping, seismic loads

are not required for consideration of occasional loads in satisfying the Level B service limits for Eq. 9. SAM stresses are not required for consideration of secondary stresses in Eq. 10. However, stresses due to the combination of range of moments due to thermal expansion and SSE anchor motions shall not exceed 3.0 Sh. This was defined as Eq. 10b.

ABB-CE designs the System 80+ plant in accordance with the single-earthquake design based on the SSE. CESSAR-DC Section 3.7.3.2 states that seismic Category I subsystems, components, and equipment are designed for a total of two SSE events with 10 maximum stress cycles per event. Alternatively, an equivalent number of fractional vibratory cycles to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE 344-1987. The staff finds the elimination of OBE to be consistent with SECY-93-087 and, therefore, acceptable. The OBE elimination is also discussed in Section 3.1.1 of this report.

Potential Impact Number: 23788 **SRP Section Number:** 3.9.3

Document: ADVANCE SER CE CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

In regards to "functional capability", the staff noted that previously ABB-CE had stated that the criteria in Texas Utilities Electric Co. letter TXX-3423 would be used to satisfy functional capability requirements in piping systems and that the NRC reviewed and accepted the criteria of TXX-3423. The staff indicated that the above criteria were only applicable to stainless steel elbows and that ABB-CE should also provide criteria for other piping products and joints and other materials. ABB-CE included a new CESSAR-DC Section 3.9.3.1.4.3 for Class 1, 2, and 3 piping systems (Amendment Q) to state that the piping is designed to meet a D/t ratio of less than 50 in accordance with NUREG-1367 and is therefore acceptable.

The staff conducted a detailed review in 3.12.5.12 of the NUREG-1367 commitment and noted that, to satisfy functional capability requirements, all ASME Code Class 1, 2, and 3 piping systems are designed to meet a pipe diameter-to-wall thickness (D/t) ratio of less than 50 in accordance with NUREG-1367 and the allowable stress from the

ASME Code of 3.0Sm (Class 1) or 3.0Sh (Class 2 and 3) but not greater than 2.0Sy. The objective of NUREG-1367 was to determine whether present Code rules, and potential changes to the Code rules, are sufficient to ensure maintenance of functional capability. Through a review of dynamic testing sponsored by the EPRI and the NRC, it was established that stress levels corresponding to the current ASME Code Eq. (9) stress limits for Level D do not result in a loss of piping functional capability provided that certain conditions are met. The conditions are as follows: dynamic loads are reversing; dynamic moments are calculated using an elastic response spectrum analysis with +15 percent peak broadening and with no more than 5 percent damping; steady-state stresses do not exceed 0.25Sy; Do/t does not exceed 50; and external pressure does not exceed internal pressure. The staff therefore finds the ABB-CE functional capability limits for System 80+ piping acceptable provided that all of the NUREG-1367 conditions are met.

Potential Impact Number: 23789 **SRP Section Number:** 3.9.3

Document: ADVANCE SER CE CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The staff noted: 'In SECY-90-016 dated January 12, 1990, the NRC staff recommended that the Commission approve the staff's resolution of the intersystem loss-of-coolant accident (ISLOCA) issue for advanced LWR plants by requiring that low-pressure piping systems that interface with the RCPB be designed to withstand full RCS pressure to the extent practicable. In its June 26, 1990 SRM, the Commission approved the staff position, as an applicable regulation pursuant to SECY-92-287, provided that all elements of the low-pressure system are considered.'

ABB-CE provided its proposed implementation of the issue resolution for the System 80+ design and the staff evaluated the proposal in section 3.9.3.1.1. Specifically, the staff evaluated the minimum pressure for which low-pressure systems should be designed to ensure reasonable protection against burst failure should the low-pressure system be subjected to full RCS pressure. In establishing the minimum design pressure, the following goals were used as the basis for selection:

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(1) The likelihood of rupture (burst) of the pressure boundary is low (The NRC staff used a goal of approximately 10 percent failure probability (or conversely, a goal of 90 percent survival probability) for rupture.)) ;
(2) The likelihood of intolerable leakage of flange joints or valve bonnets is reasonably low although some leakage might occur;
(3) Some piping components might undergo gross yielding and permanent deformation.

Potential Impact Number: 23792 **SRP Section Number:** 6.2.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.63(a)(2)

Summary:

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.

Potential Impact Number: 23793 **SRP Section Number:** 6.2.4

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

The ability to maintain "appropriate containment integrity" should be addressed. "Appropriate containment integrity" for station blackout means that adequate containment integrity is ensured by providing the capability, independent of the preferred and blackout unit's onsite emergency ac power supplies, for valve position indication and closure for containment isolation valves that may be in the open position at the onset of a station blackout.

Potential Impact Number: 23794 **SRP Section Number:** 3.6.1

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Document: NRC POLICY STATEMENT 54 FR 18649
Title: POLICY STATEMENT ON ADDITIONAL APPLICATIONS OF LEAK-BEFORE-BREAK TECHNOLOGY

Summary:

Having considered all public comments received, the Commission has decided not to undertake any rulemaking to extend the applicability of LBB to ECCS or EQ at this time. In large part, any safety benefits associated with ECCS can presently be more readily obtained under the recent ECCS rule. The use of exemptions for applying LBB to environmental qualification was permitted in the revision to General Design Criterion 4 (52 FR 41288). This option continues to remain open.

Potential Impact Number: 23800 **SRP Section Number:** 3.8.3

Document: ADVANCE SER CE CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The staff asked ABB-CE to commitment, in Sections 3.2 and 3.9.3.4, that concrete expansion anchor bolts which are used for pipe support base plates be designed to the applicable factors of safety and baseplate flexibility accountability requested in I&E Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," Revision 2, November 8, 1979. ABB-CE responded by committing to the requirements of ACI-349, "Code Requirements for Nuclear Safety Related Concrete Structures." The staff found this commitment to be unacceptable because the staff has not currently endorsed ACI-349, Appendix B. ACI-349, Appendix B, is presently being revised, in part to address certain unconservative approaches relating to expansion anchor capacities. The staff did note the design of expansion anchors is primarily a structural issue. The adequacy of the design will be determined by structural review and the staff review of this issue is discussed in Section 3.7.3 of the SER. On this basis, the open item was resolved as a mechanical issue.

Analyst Note: The reference to Section 3.7.3 of the SER should probably have been to Section 3.8.3. Section 3.7.3 deals with Seismic Subsystem Analysis, and has no

existing reference to ACI-349. Section 3.8.3 deals with Concrete and Steel Internal Structures of Steel or Concrete Containments, and references ACI-349.

Potential Impact Number: 23801 **SRP Section Number:** 3.9.3

Document: ADVANCE SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

ASME Code, Section III requires that the cumulative damage from fatigue be evaluated for all ASME Code Class 1 piping. The cumulative fatigue usage factor should take into consideration all cyclic effects caused by the plant operating transients for a 60-year design life. However, recent test data indicates that the effects of the reactor environment could reduce the fatigue resistance of certain materials. A comparison of the test data with the Code requirements indicates that the margins in the ASME Code fatigue design curves might be less than originally intended. At this time, the staff is assessing the potential generic implication of this issue on all operating plants. Depending on the severity of the issue, certain actions might be required to generically address this concern.

The 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 staff reviewed these guidelines and found them to be acceptable.

Potential Impact Number: 23803 **SRP Section Number:** 3.9.3

Document: ADVANCE SER CE CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The staff requested that for all ASME Code, Section III, Class 2 and 3 components, equipment, and supports that are designed for a 60-year design life and subject to

loadings that would result in thermal or dynamic fatigue, ABB-CE should commit to perform fatigue analyses similar to those required for Class 1 components in Subsection NB of the ASME Code, Section III. In response, ABB-CE stated that Class 2 and 3 components are reviewed for thermal fatigue effects using the ASME Section III NC-3219.2 for guidance and that fatigue analysis is performed in accordance with NC-3200 for these components if required. The staff finds that the commitment for fatigue evaluation of Class 2 and 3 components is similar to those required for Class 1 components and is, therefore, acceptable.

Potential Impact Number: 23806 **SRP Section Number:** 3.9.3

Document: ADVANCE SER CE CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: APPLICATION OF NATIONAL CODES AND STANDARDS IN DESIGN CERTIFICATION

Summary:

The staff noted that ASME Section III requires fatigue analyses on the cumulative damage for all ASME Code Class 1 piping. The cumulative fatigue usage factor should take into consideration all cyclic effects caused by the plant operating transients for a 60-year design life. However, recent test data indicates that the effects of the reactor environment could reduce the fatigue resistance of certain materials. A comparison of the test data with the Code requirements indicates that the margins in the ASME Code fatigue design curves might be less than originally intended. At this time, the staff is assessing the potential generic implication of this issue on all operating plants. Depending on the severity of the issue, certain actions might be required to generically address this concern.

The vendor stated that observations of significant environmental degradation of the cyclic behavior of materials in LWR environments are primarily related to high strain ranges, slow strain rates, high oxygen content of LWR primary water environments, high sulfur content of carbon and low alloy steels, and low flow rate conditions. The absence of any one of these conditions is considered to be sufficient to preclude any significant environmental degradation of the fatigue behavior of materials exposed to typical PWR primary coolant environment. The vendor further noted that since System 80+ components are not exposed to high oxygen content environments at elevated temperatures, and no carbon or low alloy steel is directly exposed to the primary coolant, no significant environmental degradation of the cyclic behavior of System 80+

components will occur, and that the existing fatigue curves contained in Appendix I to ASME Code, Section III, may be used as the basis for analyzing fatigue of System 80+ components, if and when such analyses are required. The staff found this position acceptable for System 80+ piping systems.

Potential Impact Number: 23807 **SRP Section Number:** 3.9.3

Document: ADVANCE SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

SSAR Section 3.9.3.1 states that the design life for the ABWR is 60 years. SSAR Sections 3.9.3.1 and 3.9.7.2 state that COL applicants will identify all ASME Code Class 2, 3, and QG D components that will be subjected to loadings that could result in thermal or dynamic fatigue, and to provide the analyses which are similar to those required by ASME Code, Section III, Subsection NB (ASME Class 1). These Sections of the SSAR also state that COL applicants will identify ASME Code Class 2 or 3 or QG D components that are subjected to cyclic loadings, including operating vibration loads and thermal transient effects, of a magnitude and/or duration so severe that the 60-year design life cannot be assured by required Code calculations and, if similar designs have not already been evaluated, either provide an appropriate analysis to demonstrate the required design life or provide designs to mitigate the magnitude or duration of the cyclic loads. SSAR Section 3.9.3.1 provides criteria on the magnitude of severe thermal transients that should be evaluated for possible effect on plant life. These transients are temperature rate changes faster than 830 C/hour (1494 F/hour), when the total fluid temperature change is greater than 380 C (684 F). It also states specifically that COL applicants will perform ASME Code Class 1 fatigue analysis of the SRV discharge piping in the wetwell and the SRV quenchers. This is acceptable. On the basis of current data, the staff is of the opinion that the margins built in to the ASME fatigue design curves might not be sufficient to account for variations in the original fatigue test data because of various environmental effects. Therefore, consistent with the staff position discussed in Section 3.12.5.7 above, the staff's position for ASME Code Class 2 and 3 piping for which a fatigue analysis is performed is that the environmental effects shall be considered in the fatigue analysis. SSAR Section 3.9.3.1, discussed above, states that environmental effects will be considered in the fatigue analyses of the SRV discharge piping and SRV quenchers in accordance with the requirements for ASME Code, Section III, Class 1 carbon steel piping discussed in Subsection 3.9.3.1.1.7 of this report. This is acceptable.

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Potential Impact Number: 23808 **SRP Section Number:** 3.9.3

Document: ADVANCE SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

Thermal stratification is a phenomenon which can occur in long runs of horizontal piping when two streams of fluid at different temperatures flow in separate layers without appreciable mixing. Under such stratified flow conditions, the top of the pipe may be at a much higher temperature than the bottom. This thermal gradient produces pipe deflections, support loads, pipe bending stresses and local stresses that may not have been accounted for in the original piping design. The effects of thermal stratification have been observed in both BWR and PWR feedwater piping as discussed in NRC Information Notice (IN) 84-87 and NRC IN 91-38.

3.9.3.1.

The staff asked GE to justify (1) the adequacy of the piping analysis load input and (2) the omission of the high-cycle fatigue effects of thermal striping from the analysis. In addition, the staff asked GE to provide additional justification for their methodology including test information to support their thermal stratification load definition.

In accordance with NRC Bulletin 88-08, the staff is requesting that Licensees and applicants review system connected to the reactor coolant system (RCS) to determine if any sections of such piping that cannot be isolated can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves. The staff requested that GE review the ABWR design to determine if this phenomenon could occur. In discussing GE's reasons, the staff concludes that the ABWR design adequately addresses the potential problems described in Bulletin 88-08, Supplements 1 and 2. However GE did not address the potential problem describe din Supplement 3 to Bulletin 88-08.

3.12.5.9

For the affected piping sections, GE will require that either (1) the gate valve in each of the unisolable piping sections be located at a distance equal or greater than 25 pipe diameters from the RPV nozzle, or (2) stress analysis be performed to show that stresses and fatigue from potential stratification and thermal striping are acceptable per

the ASME Code. On the basis of the above, the staff concludes that GE has adequately addressed the potential problem described in Supplement 3 to Bulletin 88-08.

Potential Impact Number: 23812 **SRP Section Number:** 3.9.3

Document: ADVANCE SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff noted: 'In SECY-90-016, the staff recommended that the Commission approve the staff's resolution of the intersystem loss-of-coolant accident (ISLOCA) issue for ALW reactor plants by requiring that low-pressure piping systems that interface with the RCPB be designed to withstand full RCS pressure to the extent practicable. In its June 26, 1990, SRM, the Commission approved the staff's position provided that all elements of the low-pressure system are considered.'

The staff evaluated GE's approach for implementing the ISLOCA resolution for the ABWR. Specifically, the staff evaluated the minimum pressure for which low-pressure systems should be designed to ensure reasonable protection against burst failure should the low-pressure system be subjected to full RCS pressure. In establishing the minimum design pressure, the following goals were used as the basis for selection:

(1) The likelihood of rupture (burst) of the pressure boundary is low (The NRC staff used a goal of approximately 10 percent failure probability (or conversely, a goal of 90 percent survival probability) for rupture.)) ;
(2) The likelihood of intolerable leakage of flange joints or valve bonnets is reasonably low although some leakage might occur;
(3) Some piping components might undergo gross yielding and permanent deformation.

Potential Impact Number: 23814 **SRP Section Number:** 3.9.3

Document: ADVANCE SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

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In a letter dated January 28, 1993, GE revised Note 6 to SSAR Table 3.9-2 to state 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 requirements of NUREG-1367, "Functional Capability of Piping Systems," dated November 1992. This document contains methodology acceptable to the staff for ensuring the functional capability of essential piping systems.

Note 9 to the SSAR, Table 3.9-2, states that all ASME Code Class 1, 2, and 3 piping systems that are essential for safe shutdown under the postulated events listed in the table are designed to meet the recommendations in NUREG-1367, "Functional Capability of Piping Systems," dated November 1992. In no case shall the piping stress exceed the limits designated for Service Level D in the ASME Code, Section III. The Service Level D limits are 3.0 Sm (not to exceed 2.0 Sy) for ASME Code Class 1 piping and 3.0 Sh (not to exceed 2.0 Sy) for Class 2 and 3 piping. Through dynamic testing conducted by EPRI, GE and the NRC, it has been established that these stress levels do not result in a loss of piping functional capability. Thus, the staff concludes the methodology and stress levels for ensuring the functional capability of piping systems acceptable.

Potential Impact Number: 23817 **SRP Section Number:** 3.9.3

Document: ADVANCE SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

With regards to Issue 78 "Monitoring of Fatigue Transient Limits for Reactor Coolant System," the staff found that the conditions and the methodology proposed by GE constitute supplemental guidelines that enhance the design margins beyond the requirements of the ASME Code, Section III for fatigue evaluation and are acceptable as discussed in Section 3.9.3.1. Based on this information, the staff has determined that GE has adequately addressed this issue for the ABWR design. In SSAR Section 3.9.3.1.1.7, GE commits to perform additional evaluations for environmental effects on the fatigue design of ASME Code Section III, Class 1 carbon steel piping in accordance with GE document, 408HA414.

Potential Impact Number: 23818 **SRP Section Number:** 3.9.3

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Document: ADVANCE SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

In SECY-90-016, the staff requested the Commission's approval to decouple the level of the OBE ground motion from that of the safe-shutdown earthquake (SSE). The Commission approved the staff's position in its staff requirements memorandum (SRM) of June 26, 1990. In SECY-93-087, the staff recommended to eliminate the OBE from design for advanced light-water reactors. The Commission approved the staff recommendations in its SRM dated July 21, 1993.

In 3.1.1.1 the staff noted its review of commitments specified in the SSAR to ensure that appropriate measures and adequate safety margins are maintained when the OBE is eliminated from the design include evaluation of American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components and core support structures, The staff further noted that the loads and load combinations used for evaluating ASME Code Class 1, 2, and 3 components and core support structures are provided in the ABWR SSAR and discussed in Section 3.9.3.1.

Potential Impact Number: 23822 **SRP Section Number:** 3.9.3

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Section 3.3 of Chapter 1 states that the plant design life for the ALWR will be 60 years. This proposed design life raises questions relative to the margins available in the current ASME fatigue design curves. These margins were established almost 30 years ago and were obtained from best-fit curves of fatigue test data by applying a factor of either 2 on stress or 20 on cycles, whichever was more conservative at each point. These factors were originally intended to cover such effects as environment, size effect, and scatter of data. However, on the basis of limited data currently available, the staff concludes that these margins may not be sufficient to account for variations in the original fatigue test data as a result of various environmental effects. In its letter dated May 17, 1991, the staff requested a commitment in Section 4.7.3 of Chapter 1 to consider such effects in the designs of applicable ASME Code Class 1 systems, components, and equipment. In its letter dated August 1, 1991, EPRI stated that if

additional data or research results yield findings requiring changes to the current ASME fatigue design curves, the code consensus process will provide the proper vehicle to address such findings. The staff does not agree with all of the discussions in this response and concludes that the above commitment is not completely acceptable. The ASME Code curves may not be revised for many years. Therefore, the staff's position is that until these curves are revised, all ALWR applicants and all licensees applying for license renewal should propose appropriate fatigue design curves that will be reviewed by the staff.

Potential Impact Number: 23825 **SRP Section Number:** 6.2.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: 50 FR 3498, 50 FR 3498, 50.44 (c) (iv)

Summary:

At 50 FR 3498, the Commission amended 10 CFR Part 50, Section 50.44 of its regulations to include hydrogen control capability for BWRs with Mark III containments and PWRs with ice condenser containments. The amendment requires improved hydrogen control systems that can handle an amount of hydrogen equivalent to that generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region.

At 50 FR 3499, the Commission noted that numerous commenters suggested that the implementation of the hydrogen control rule should be deferred until the severe accident rulemaking, when applicable research and PRAs will be completed. The Commission agreed with these comments relative to PWRs with large dry containments.

With regards to BWRs with Mark III containments and PWRs with ice condenser containments, the Commission believes that these containments can safely accommodate the burning in a single event of the hydrogen from about a 25% metal-water reaction. Since the TMI-2 accident showed that a 45-50% metal-water reaction was possible, the commission believes that it is necessary to enhance the hydrogen control system with these type containments.

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Potential Impact Number: 23826 **SRP Section Number:** 6.2.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: 50.34 (f), 50.34 (f), 50.34 (f) (1) (xii), 50.34 (f) (2) (ix), 50.34 (f) (3) (v)

Summary:

Paragraphs (1)(xii), (2)(ix), and (3)(v) of this section, pertaining to hydrogen control measures, must be met by all applicants covered by this rule. However, the Commission may decide to impose additional requirements and the issue of whether compliance with these provisions, together with 10 CFR 50.44 and Criterion 50 of Appendix A to 10 CFR Part 50, is sufficient for issuance of the manufacturing license may be considered in the manufacturing license proceeding.

(f) (1) (xii) Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of this section. As a minimum include consideration of a hydrogen ignition and post-accident inerting system.

(f) (2) (ix) Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (f)(1)(xii) of this section is sufficient at the construction permit stage.

(f)(3)(v) Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that: Containment integrity will 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, Subsubarticle 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.

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Potential Impact Number: 23828 **SRP Section Number:** 3.9.3

Document: NRC BULLETIN 88-08
Title: THERMAL STRESSES IN PIPING CONNECTED TO REACTOR COOLANT SYSTEMS

Summary:

A problem with thermal fatigue cracking in ECCS piping at Farley 2 is described. The licensees were asked to nondestructively examine the welds of any unisolable sections of piping connected to the RCS that may have been subjected to excessive thermal stresses. The GL also requested that weld heat-affect-zones and high stress locations be examined.

Potential Impact Number: 23831 **SRP Section Number:** 3.9.3

Document: NRC BULLETIN 88-08 Sup 3
Title: THERMAL STRESSES IN PIPING CONNECTED TO REACTOR COOLANT SYSTEMS

Summary:

This supplement provides information to licensees relating to experiences with thermal stratification in unisolable piping connected to the Reactor Coolant System which has caused pipe cracks to develop. The supplement also requests licensees to review their plants design and operations to identify and fix any such potential conditions.

Potential Impact Number: 23832 **SRP Section Number:** 3.9.3

Document: NRC MEMORANDUM 04/02/93, from W. Minners to F. Gillespie
Title: PROPOSED RESOLUTION OF GENERIC ISSUE 105, 'INTERFACING SYSTEMS LOCA IN LWR'S'

Summary:

RES proposed a new SRP section as part of the resolution of Generic Issue 105.

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Potential Impact Number: 23833 **SRP Section Number:** 3.9.3

Document: NRC MEMORANDUM 06/03/93, from E. Beckjord to J. Taylor
Title: TECHNICAL RESOLUTION OF GENERIC ISSUE 105, 'INTERFACING SYSTEMS LOSS OF COOLANT ACCIDENT IN LWR'S'

Summary:

In the memorandum presenting the resolution of Generic Issue 105, RES informed the EDO that a draft SRP section covering design review of systems interfacing with the reactor coolant system in ALWRs was provided to NRR for information and use as appropriate.

Potential Impact Number: 23835 **SRP Section Number:** 3.9.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue addresses rupture of low pressure piping systems which interface with the reactor coolant pressure boundary due to pressure isolation system failure.
In resolving the issue, the staff conducted analyses of units representative of all NSSS vendors and considered: (1) human errors, both as initiators and during recovery operations; (2) component fragilities, to determine likely low pressure system break locations; and (3) the post-ISLOCA auxiliary building environment, to determine the survivability of recovery equipment. It was concluded that the units studied posed little risk from ISLOCA.

In addition to the above analyses, previous PWR ISLOCA studies were reexamined with data updated to include the seven years of operating experience that had accrued since the initial analyses were undertaken. None of the studies supported generic requirements for PWRs, whether on absolute risk reduction or cost-beneficial bases. The study of ISLOCA at a BWR confirmed past PRA studies which generally indicated little risk contribution from ISLOCA sequences.

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The staff found that ISLOCAs at PWRs were plant-specific in nature; however, the ongoing IPE program (GL 88-20) includes licensee analysis of ISLOCA sequences. With respect to future applicants, a draft SRP Section covering design review of systems interfacing with the RCS in ALWRs was provided (Memorandum for F. Gillespie from W. Minners, "Proposed Resolution of Generic Issue 105, `Interfacing Systems LOCA in LWRs,'" April 2, 1993), to NRR for information and use as appropriate. A supplement to Information Notice 92-36 was also recommended to share insights from the ISLOCA program and to inform licensees of the availability of material useful for IPE ISLOCA analyses not yet completed, or as a check on analyses already completed. Thus, this issue was RESOLVED and no new requirements were established.

Potential Impact Number: 23838 **SRP Section Number:** 5.2.4

Document: NRC GENERIC LETTER 83-15
Title: IMPLEMENTATION OF REGULATORY GUIDE 1.150, 'ULTRASONIC TESTING OF REACTOR VESSEL WELDS DURING PRESERVICE AND INSERVICE EXAMINATIONS, REVISION 1

Summary:

Regulatory Guide 1.150, Revision 1 is being issued as an active guide without prior issuance as a draft guide for comment.

Except in those cases in which an applicant or licensee proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described in Revision 1 of the subject Guide will be used in the evaluation of (1) the results of inservice examination programs of all operating reactors after July 15, 1981, and (2) the results of preservice examination program of all reactors under construction performed after January 15, 1982. We recommend that all licensees modify their technical specifications if appropriate, to make them consistent with the recommendations in the subject Guide.

Potential Impact Number: 23840 **SRP Section Number:** 9.3.4

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

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Summary:

This issue deals with the high rate of Reactor Coolant Pump (RCP) seal failures that challenge the makeup capacity of the ECCS in PWRs.

Depending on the assumptions, this issue could be ranked as low priority. However, this conclusion would be based on an optimistic assessment of operator response, the high cost and large occupational exposure incurred by annual seal replacement, and the exclusion of the down-time and associated costs due to minor seal failures.

If operator response is poor, then the potential reduction in public risk would justify a high priority. Other solutions may be less costly and incur less occupational exposure which could markedly improve the value/impact ratio. Any reduction in the overall RCP seal failure frequency would reduce unscheduled shutdown and the high associated costs. Therefore, based on the potential for a large reduction in public risk, the belief that better solutions can be found, and the potential for significant saving of replacement power cost, this issue should be a HIGH priority.

Potential Impact Number: 23844 **SRP Section Number:** 9.5.5

Document: NRC GENERIC LETTER 89-13
Title: SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

Summary:

Perform functional testing with the heat exchanger operating, if practical, at its design heat removal rate to verify its capabilities. Temperature and flow compensation should be made in the calculations to adjust the results to the design conditions. Trend the results, as explained above, to monitor degradation. An example of this type of heat exchanger would be that used to cool a diesel generator. Engine jacket water flow and temperature and service water flow and temperature could be monitored and trended during the diesel generator surveillance testing.

If it is not practical to test the heat exchanger at the design heat removal rate, then trend test results for the heat exchanger efficiency or the overall heat transfer coefficient. Verify that heat removal would be adequate for the system operating with the most limiting combination of flow and temperature. III.

Perform efficiency testing (for example, in conjunction with surveillance testing) with the heat exchanger operating under the maximum heat load that can be obtained practically. Test results should be corrected for the off-design conditions. Design heat removal capacity should be verified. Results should be trended, as explained above, to identify any degraded equipment.

Potential Impact Number: 23847 **SRP Section Number:** 9.5.1

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The NRC expects any new reactor design to propose fire protection systems based on the best technology available, not on the methods allowed for plants already operating or in the advance stages of design and construction. Therefore, the staff evaluated the fire protection system of the System 80+ design against the criteria of SRP Section 9.5.1 (BTP CMEB 9.5-1 Rev. 2), which meets the requirements of GDC 3, and against the following proposed applicable regulation for fire protection: The standard design must comply with 10 CFR Part 50, Appendix R, Section III.G.1.a and ensure that safe shutdown can be achieved assuming all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. The design must ensure that smoke, hot gasses, or fire suppressant, will not migrate into other fire areas to the extent that could adversely affect safe shutdown capabilities including operator actions. The control room is excluded because an alternative shutdown capability is provided which is physically and electrically independent of the control room. In the reactor containment, redundant shutdown systems must be provided with fire protection to ensure, to the extent practical, that one shutdown division be free of fire damage. Because of unique design layout, other areas may be accepted on an individual basis.

Potential Impact Number: 23848 **SRP Section Number:** 9.5.1

Document: FINAL SER EPRI CH 9
Title: EPRI URD-EPD: SITE SUPPORT SYSTEMS

Summary:

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The intention has always been that when any advanced reactor designs were proposed, fire protection would be provided on the basis of the best technology available, not on the basis of methods allowed for plants already operating or in advanced stages of design and construction. On this basis, the staff evaluated the criteria in the Evolutionary Requirements Document for the fire protection system against the criteria of SRP Section 9.5.1 (BTP CMEB 9.5-1, July 1981) and supplemental guidance issued by the Commission. Three examples of such supplemental guidance are (1) Generic Letter 81-12, which contains information on safe shutdown methodology; (2) Generic Letter 86-10, which contains important technical information, such as conformance with National Fire Protection Association codes and standards; and (3) the Commission's staff requirements memorandum (SRM) dated June 26, 1990.

In SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," the staff concluded that fire issues that have been raised through operating experience and through the External Events Program 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 that the Commission enhance NRC's current guidance. In its SRM of June 26, 1990, on SECY-90-016, 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. The designers of standard plants must demonstrate that safe shutdown of their plants can be achieved, assuming that all equipment in any one fire area has been rendered inoperable by fire and that reentry to the fire area for repairs and for operator actions is not possible. The control room is excluded from this approach, subject to the need for an independent alternative shutdown capability that is physically and electrically independent of the control room. Fire protection for redundant shutdown systems in the reactor containment building should ensure, to as great an extent as possible, that one shutdown division will be free of fire damage. Consideration should be given to safety-grade provisions for the fire protection systems to ensure that the remaining shutdown capabilities are protected. In addition, it should be demonstrated that smoke, hot gases, or fire suppressants will not migrate into other fire areas to the extent that safe shutdown capabilities, including operator actions, could be adversely affected.

Potential Impact Number: 23849 **SRP Section Number:** 9.5.1

Document: FINAL SER ABWR CH 9

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Title: AUXILIARY SYSTEMS

Summary:

The NRC expects any new reactor design to propose fire protection systems based on the best technology available, not on the methods allowed for plants already operating or in advanced stages of design and construction. There- fore, the staff evaluated the fire protection system of the ABWR against the criteria of SRP Section 9.5.1 (BTP CMEB 9.5-1 Rev. 2), which meets the requirements of GDC 3, and against the following proposed applicable regulation for fire protection:

The standard design must comply with 10 CFR Part 50 Appendix R, Section III.G.1.a and ensure that safe shutdown can be achieved assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. The design must ensure that smoke, hot gases, or fire suppressant will not migrate into other fire areas to the extent that could adversely affect safe-shutdown capabilities including operator actions. The control room is excluded because an alternative shutdown capability is provided which is physically and electrically independent of the control room. In the reactor containment, redundant shutdown systems must be provided with fire protection to ensure, to the extent practical, that one shutdown division be free of fire damage. Because of unique design layout, other areas may be accepted on an individual basis.

Potential Impact Number: 23851 **SRP Section Number:** 9.5.1

Document: DRAFT FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

The staff has a concern regarding the non-inclusion of recommendations (see Enclosure 1, "Recommended Program to Resolve Generic Issue 51" to Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment") as additional interface criteria, including provisions for biocide treatment before layup for other systems such as fire protection systems which use raw service water as a source.

Potential Impact Number: 23852 **SRP Section Number:** 9.5.1

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Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

In the DSER, the staff indicated that the minimum acceptable criteria for a fire protection water-supply system includes the following: (1) A dedicated fire-protection water-supply and distribution system must be designed and installed in accordance with National Fire Protection Association Standards 11, 13, 14, 15, 20, and 24 to meet the anticipated needs for fixed water-suppression systems and manual hose stations. (2) The sprinkler systems in the reactor building and the wet standpipe systems in the reactor and control buildings must be designed in compliance with ANSI B31.1 and analyzed to remain functional following a safe-shutdown earthquake. 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. The remainder of the water systems must be designed to the appropriate fire-protection standards. 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.

Potential Impact Number: 23854 **SRP Section Number:** 9.5.1

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The sprinkler systems in the reactor building and the wet standpipe systems in the reactor and control buildings are designed in compliance with ANSI B31.1 and analyzed to remain functional following an SSE (seismically analyzed). Portions of the water supply system, including a tank, a pump, and part of the yard supply main, are designed to these requirements also. The remainder of the water systems are designed to the appropriate fire protection standards. During normal operation, the seismically-analyzed and non-seismically-analyzed systems are 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.

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Potential Impact Number: 23855 **SRP Section Number:** 9.5.1

Document: FINAL SER EPRI CH 9
Title: EPRI URD-EPD: SITE SUPPORT SYSTEMS

Summary:

In resolution of a confirmatory issue from the DSER, the staff has established that seismically sensitive relays will not be used in fire protection systems in which seismic induced failure could lead to unacceptable consequences.

Potential Impact Number: 23857 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 4-77
Title: Recommendations for Developing Fire Protection Services for the Public

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section . The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23858 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 4A-69
Title: Fire Department Organization

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements

in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23859 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 6-74
Title: Recommendations for Organization of Industrial Fire Loss Prevention

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23860 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 7-74
Title: Management of Fire Emergencies

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23861 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 8-74
Title: Effects of Fire on Operations Management Responsibility

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Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23862 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 10-75
Title: Standard for Portable Fire Extinguishers

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23863 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 11-75
Title: Standard for Low Expansion Foam and Combined Agent Systems

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23864 **SRP Section Number:** 9.5.1

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Document: C&S: NFPA 11A-70
Title: Standard for Medium-and High-Expansion Foam Systems

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23865 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 11B-74
Title: Standard for Low Expansion Foam and Combined Agent Systems

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23866 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 12-73
Title: Standard on Carbon Dioxide Extinguishing Systems

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

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Potential Impact Number: 23867 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 12A-73
Title: Standard on Halon 1301 Fire Extinguishing Systems

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23868 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 12B-73
Title: Standard on Halon 1211 Fire Extinguishing Systems

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23869 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 13
Title: INSTALLATION OF SPRINKLER SYSTEMS

Summary:

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NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23870 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 14-74
Title: Standard for the Installation of Standpipe and Hose Systems

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23871 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 15
Title: WATER SPRAY FIXED SYSTEMS

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23872 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 16-73

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Title: Standard on Deluge Foam-Water Sprinkler and Foam-Water Spray Systems

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23873 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 20-73
Title: Standard for the Installation of Centrifugal Fire Pumps

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23874 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 24-73
Title: Standard for the Installation of Private Fire Service Mains and Their Appurtenances

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

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Potential Impact Number: 23875 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 26-58
Title: Recommended Practice for the Supervision of Valves Controlling Water Supplies for Fire Protection

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23876 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 27-75
Title: Standard on Industrial Fire Brigades

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23877 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 30
Title: FLAMMABLE AND COMBUSTIBLE LIQUIDS CODE

Summary:

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NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23878 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 50A
Title: Standard for Gaseous Hydrogen Systems at Consumer Sites

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23879 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 51B-76
Title: Standard for Fire Prevention in Use of Cutting and Welding Processes

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23880 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 69-73

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Title: Standard on Explosion Prevention Systems

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23881 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 70-75
Title: National Electrical Code

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23882 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 72D-75
Title: Standard for the Installation, Maintenance, and Use of Protective Signaling Systems

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

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Potential Impact Number: 23883 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 72E-74
Title: Standard on Automatic Fire Detectors

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23884 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 80-75
Title: Standard for Fire Doors and Windows

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23885 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 90A
Title: Standard for Installation of Air Conditioning and Ventilating Systems

Summary:

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NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23886 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 92-72
Title: Waterproofing and Draining of Floors

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23887 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 197-66
Title: Training Standard on Initial Fire Attack

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23888 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 204-68

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Title: Guide for Smoke and Heat Venting

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23889 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 220
Title: Standard on Types of Building Construction

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23890 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 251
Title: FIRE TESTS OF BUILDING CONSTRUCTION AND MATERIALS

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

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Potential Impact Number: 23891 **SRP Section Number:** 9.5.5

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

In the FSER, the staff indicated that, according to GE, specific information regarding the design and capability of the cooling water system and the keep-warm system were interface requirements. These included a requirement that expected water loss over 7-day period and system volume capacity needed to ensure adequate volume is available to maintain system water level and pump NPSH without refill.

These requirements were identified as DFSER Open Item 9.5.5-1. Upon further evaluation, the staff has determined that these requirements do not meet the definition of interface requirements implied in 10 CFR Part 52 and can be accomplished by identifying a COL action item in the SSAR requiring the COL applicant referencing the ABWR design to provide this information. GE has included this information in the SSAR.

Potential Impact Number: 23892 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 259-76
Title: Standard Test Method for Potential Heat of Building Materials

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23893 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 802-74

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Title: Recommended Fire Protection Practice Facilities Handling Radioactive Materials for Nuclear Research Reactors

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23894 **SRP Section Number:** 9.5.1

Document: C&S: NFPA 1962
Title: Standard for the Care, Use, and Service Testing of Fire Hose Including Couplings and Nozzles

Summary:

NFPA standards referenced in BTP CMEB 9.5-1, to SRP Section 9.5.1, have been withdrawn, updated, and/or revised since the last revision of the SRP Section. The NFPA Standards provide the basis for much of the fire protection program requirements in the SRP (e.g., the design and installation of fire detection and suppression equipment).

Potential Impact Number: 23895 **SRP Section Number:** 9.5.5

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

Specific information regarding standpipe capacity, circulation and keep-warm pump characteristics, system volume, flow rates, and heat removal capabilities, plus expected water loss over a 7-day period, are identified as interface issues to be provided once the DG vendor has been selected and diesel engine data are available. This information must be provided as part of the application for a combined license. This

was identified as COL Action Item 9.5.5-1 in the DSER. ABB-CE addressed this issue in Section 9.5.5.2 of Amendment Q to the CESSAR-DC by stating the COL applicant will provide plant-specific information for the DGECWS. The staff finds this acceptable.

Potential Impact Number: 23896 **SRP Section Number:** 12.1

Document: ADVANCE SER ABWR CH 12
Title: RADIATION PROTECTION

Summary:

The SRP acceptance criteria provide assurance that the radiation doses resulting from exposure to licensed radioactive sources outside the body and inside the body can each be maintained well within the limits of 10 CFR Part 20 and ALARA. The balancing of internal and external exposure necessary to ensure that their sum is ALARA is an operational concern that will be reviewed in conjunction with a combined license (COL) application. The Part 20, as amended, contains a number of new programmatic requirements that do not affect plant design.

Programmatic and operational radiation protection concerns will be addressed by the COL applicant.

The staff's review finds that the radiation protection measures incorporated in the ABWR design will provide reasonable assurance that occupational doses can be maintained ALARA and below the limits of 10 CFR Part 20 during all plant operations.

Potential Impact Number: 23898 **SRP Section Number:** 12.3 - 12.4

Document: NUREG 1434
Title: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6

Summary:

SRP Section 12.3-12.4 references the General Electric STS (NUREG-0123) as they relate to radiation protection considerations. Improved STS have been promulgated for industry use (microfiche - 63977:234, 63979:001, and 63980:002).

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Potential Impact Number: 23900 **SRP Section Number:** 12.5

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION
Locations: Subpart H, Subpart H, Appendix A

Summary:

Subpart H contains new rules for respiratory protection and controls to restrict internal exposure in restricted areas. Protection factors for respirators are presented in Appendix A.

Potential Impact Number: 23902 **SRP Section Number:** 9.5.1

Document: C&S: ANSI B.31.1 Revision Date: 1/1/73
Title: POWER PIPING (1973)

Summary:

ANSI B31.1-1973 is cited in Branch Technical Positon (BTP) CMEB 9.5-1. The BTP does not reference the current version (ASME B31.1-1992) of the standard.

Potential Impact Number: 23903 **SRP Section Number:** 9.5.1

Document: C&S: ASTM E119
Title: METHOD FOR FIRE TEST OF BUILDING CONSTRUCTION AND MATERIALS

Summary:

ASTM E-119-1976 is cited in Branch Technical Positon (BTP) CMEB 9.5-1. The BTP does not reference the current version (1988) of the standard.

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Potential Impact Number: 23904 **SRP Section Number:** 9.5.1

Document: C&S: ASTM E84
Title: TEST METHOD FOR SURFACE BURNING CHARACTERISTICS OF BUILDING MATERIALS

Summary:

ASTM E-84-1976 is cited in Branch Technical Positon (BTP) CMEB 9.5-1. The BTP does not reference the current version (1991) of the standard.

Potential Impact Number: 23905 **SRP Section Number:** 9.5.1

Document: C&S: IEEE 383
Title: TYPE TEST OF CLASS 1E ELECTRIC CABLES FIELD SPLICES AND CONNECTIONS

Summary:

IEEE Std 383-1974 is cited in Branch Technical Positon (BTP) CMEB 9.5-1. The BTP does not reference the current version (1974, R-1992) of the standard.

Potential Impact Number: 23906 **SRP Section Number:** 9.5.1

Document: C&S: ASTM D3286-73
Title: Standard Test Method for Gross Calorific Value of Coal and Coke by the Isoperibol Bomb Calorimeter

Summary:

ASTM D-3286 -1973 is cited in Branch Technical Positon (BTP) CMEB 9.5-1. The BTP does not reference the current version (1991) of the standard.

Potential Impact Number: 23907 **SRP Section Number:** 9.5.1

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Document: C&S: Factory Mutual Approval Guide
Title: Equipment, Materials, Services for Conservation of Property

Summary:

The Factory Mutual System Approval Guide (date not specified) is cited in Branch Technical Positon (BTP) CMEB 9.5-1. The BTP does not reference the current version (1990) of the standard.

Potential Impact Number: 23909 **SRP Section Number:** 2.4.2

Document: ADVANCE SER ABWR CH 2
Title: SITE CHARACTERISTICS

Summary:

The staff's review of the ABWR water level (flood) design is contained in Section 3.4 of this report. The bounding site parameters that were considered in the staff's evaluation are as follows:
- Floods: The ABWR shall be located on the site so that the level associated with the design-basis flood will be no higher than 30.5 cm (1 ft) below the plant grade.
- Potential Dam Failures (Seismically-Induced): Failure of existing and potential upstream or downstream water control structures shall not contribute to the water level exceeding 30.5 cm (1 ft) below grade nor compromise the ultimate heat sink.
- Probable Maximum Surge and Seiche Flooding: The probable maximum surge and seiche flooding shall be no higher than 30.5 cm (1 ft) below grade (see SSAR Table 2.1-1).
- Probable Maximum Tsunami: The probable maximum tsunami flooding shall be no higher than 30.5 cm (1 ft) below grade (see SSAR Table 2.1-1).
- Ground Water: GE states in the SSAR that the ABWR design is intended to be compatible with ground water levels up to 61 cm (2 ft) below plant grade.

Potential Impact Number: 23910 **SRP Section Number:** 2.4.2

Document: ADVANCE SER CE CH 2
Title: SITE CHARACTERISTICS

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Summary:

Pursuant to 10 CFR Part 52.47(a)(1)(iii) an applicant seeking design certification is required to (1) provide site parameters postulated for the design, and (2) analyze and evaluate the design in terms of these parameters. ABB-CE provided a list of site parameters in CESSAR-DC Table 2.0-1. The staff's evaluation of the parameters in Table 2.0-1 follows.

Maximum Flood (or Tsunami) Level - In the CESSAR-DC, ABB-CE specifies that the site-specific maximum flood level is 0.3 m (1 ft) below grade. This flood level, which is consistent with the requirement of the EPRI Evolutionary Utility Requirements Document (URD), is acceptable for the ALWR design because the minimum design-basis flood level for preventing damages to seismic Category I SSC is at 0.3 m (1 ft) below plant grade. The maximum flood level is discussed in Section 3.4 of this report.

Potential Impact Number: 23911 **SRP Section Number:** 2.5.4

Document: ADVANCE SER CE CH 2
Title: SITE CHARACTERISTICS

Summary:

Pursuant to 10 CFR Part 52.47(a)(1)(iii) an applicant seeking design certification is required to (1) provide site parameters postulated for the design, and (2) analyze and evaluate the design in terms of these parameters. ABB-CE provided a list of site parameters in CESSAR-DC Table 2.0-1. The staff's evaluation of the parameters in Table 2.0-1 follows.
...

Soil Properties - The minimum soil-bearing capacity of 59 ton/m2 (12 ksf) is acceptable for the design of the ALWR. Also, sites with liquefaction potential at the site-specific SSE level are excluded. This information is reviewed and evaluated in Sections 2.5.4.8 and 2.5.4.10.

Potential Impact Number: 23912 **SRP Section Number:** 2.5.4

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Document: ADVANCE SER ABWR CH 2
Title: SITE CHARACTERISTICS

Summary:

The staff's review of the soil properties considered in the seismic design of ABWR SSCs is contained in Section 3.7 of this report. The bounding site parameters that were considered in the staff's evaluation are as follows:
- Minimum Shear Wave Velocity: GE states in the SSAR that the minimum embedment depth for the reactor building shall be 26 m (85 ft). The minimum shear wave velocity of soil shall be 305 m/s (1000 ft/s).
- Liquefaction Potential: GE states in the SSAR that no liquefaction potential exists for soils under and around all seismic Category I structures, including seismic Category I buried pipelines and electrical ducts.
- Minimum Bearing Capacity (demand): GE states in the SSAR that the minimum (demand) bearing capacity of soil shall be 0.72 MPa (15 kip/ft2).

As discussed in Section 3.7.1 of this report, the staff concludes that the ABWR standard plant design has adequately defined the above bounding parameters.

Potential Impact Number: 23913 **SRP Section Number:** 5.2.2

Document: C&S: ASME III
Title: CLASS 1 AND 2 SYSTEMS

Summary:

Section NB-7611, "Spring-Loaded Safety Valves," of ASME III is referenced as part of the SRP Acceptance Criteria. No code year is specified. This section has been modified since 1977, and a new title and section number have been assigned. The new section is NB-7511.1, "Spring-Loaded Valves."

Potential Impact Number: 23915 **SRP Section Number:** 10.4.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

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Summary:

EPRI proposed to eliminate the main steam isolation valve leakage control system and provide an alternative leakage pathway using the main steam line and the condenser. The staff provided its positions on the seismic classification of the condenser if the MSIVLCS were eliminated, to ensure that the condenser will maintain structural integrity during and following an SSE.

Potential Impact Number: 23916 **SRP Section Number:** 10.4.1

Document: DRAFT FINAL SER ABWR CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

GE proposed to take credit for plateout and holdup of radionuclides following a LOCA, in lieu of a main steam isolation valve leakage control system. In FSER section 10.4.1, the staff stated that the main condenser must be capable, in that case, of maintaining its integrity following an SSE and that FSER sections 3.2.1, 10.3 and 15.3.1.2 contain additional discussion and evaluation of the main condenser to meet this position.

Potential Impact Number: 23917 **SRP Section Number:** 10.4.1

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

Item II.E discussed staff positions on seismic analysis of condenser anchorages and the piping inlet nozzle to the condenser to ensure that they are capable of maintaining their structural integrity during and after an SSE.

Potential Impact Number: 23919 **SRP Section Number:** 2.5.2

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Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

A seismic PRA rather than the PRA-based margins approach described above was the primary approach used to evaluate the advanced boiling water reactor (ABWR) seismic design. The staff compared the hazard estimates derived by Lawrence Livermore National Laboratory (LLNL) using the historical earthquake method (NUREG/CR-4885, "Seismic Hazard Characterization of the Eastern United States: Comparative Evaluation of the EPRI and LLNL Studies") and by EPRI with the results obtained using the ABWR bounding seismic hazard curve. The ABWR bounding hazard curve was exceeded by the LLNL mean hazard curves for the Pilgrim, Seabrook, and Watts Bar sites. These three sites in the Eastern United States were selected because of their relatively high seismic hazard. The three LLNL curves were arithmetic mean hazard curves using input from five ground motion experts. The staff used both the LLNL and EPRI seismic hazard estimates to quantify core damage frequency. 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. However, the ABWR design was found to be capable of resisting earthquakes significantly larger than a safe shutdown earthquake of 0.3g based on an estimated plant HCLPF of about 0.6g using the seismic margins methodology.

Recent analyses of the seismic inputs to the LLNL hazard program suggest that uncertainties may be reduced in future LLNL results. As an example of reduced uncertainty in the LLNL estimates, the staff compared the LLNL hazard estimates of four (rather than five) ground motion experts. Results that included the estimates of the fifth ground motion expert were generally higher. Thus, using the estimates of only four ground motion experts might provide a means of showing the effect of reduced uncertainty. However, the LLNL results using the estimates of the four ground motion experts still exceeded the EPRI bounding curve. Thus, it is the staff's conclusion that the LLNL hazard results suggest the EPRI bounding curve for rock sites is not conservative.

Potential Impact Number: 23920 **SRP Section Number:** 2.5.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

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Summary:

EPRI proposed to eliminate the OBE as a design basis event and to use alternative criteria for dynamic analysis methods.

Potential Impact Number: 23921 **SRP Section Number:** 3.11

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: 50.44(c)(3)(iv)(E), 50.44(c)(3)(iv)(E), 50.44(c)(3)(v)(A), 50.44(c)(vi)

Summary:

If the hydrogen control system relies on post-accident inerting, the systems and components required to establish and maintain safe shutdown and containment integrity must be designed and qualified for the environment caused by such inerting. Furthermore, inadvertent full inerting during normal plant operations must not adversely affect systems and components needed for safe operation of the plant.

Each licensee with a boiling light-water nuclear power reactor with a Mark III type of containment and each licensee with a pressurized light-water nuclear power reactor with an ice condenser type of containment issued a construction permit before March 28, 1979, for a reactor that does not rely upon an inerted atmosphere to control hydrogen inside the containment, shall provide its nuclear power reactor with systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity. These systems and components must be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur.

(iv)(A)Each applicant for or holder of an operating license for a boiling light-water nuclear power reactor with a Mark III type of containment or for a pressurized light-water nuclear power reactor with an ice condenser type of containment issued a construction permit before March 28, 1979, shall submit an analysis to the Commission as specified in Section 50.4.

(B) The analysis required by paragraph (c)(3)(vi)(A) of this section must: ... (5) Show that for those reactors described in paragraph (c)(3)(iv) of this section that do not rely upon an inerted atmosphere to control hydrogen inside the containment: ... (ii)

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Systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity will be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen, including the effect of local detonations, unless such detonations can be shown unlikely to occur.

Potential Impact Number: 23922 **SRP Section Number:** 2.4.2

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.17 (a)(1)(vi)

Summary:

The application (for an early site permit) must contain the information required by 10 CFR 50.33 (a)-(d), the first three sentences of 50.34(a)(1), and, to the extent approval of emergency plans is sought under paragraph (b)(2)(ii) of this section, the information required by 50.33 (g) and (j), and 50.34(b)(6)(v). In particular, the application should describe the following:... (vi) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site (see Appendix A to 10 CFR Part 100)...

Potential Impact Number: 23923 **SRP Section Number:** 2.4.2

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.47 (a)(1)

Summary:

(a) The requirements of this paragraph apply to all applications for design certification. (1) An application for design certification must contain: (i) The technical information which is required of applicants for construction permits and operating licenses by 10 CFR Part 20, Part 50 and its appendices, and Parts 73 and 100, and which is technically relevant to the design and not site-specific; (ii) Demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in

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10 CFR 50.34(f); (iii) The site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters.

Potential Impact Number: 23928 **SRP Section Number:** 2.4.5

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.17 (a)(1)(vi)

Summary:

The application (for an early site permit) must contain the information required by 10 CFR 50.33 (a)-(d), the first three sentences of 50.34(a)(1), and, to the extent approval of emergency plans is sought under paragraph (b)(2)(ii) of this section, the information required by 50.33 (g) and (j), and 50.34(b)(6)(v). In particular, the application should describe the following:... (vi) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site (see Appendix A to 10 CFR Part 100)...

Potential Impact Number: 23929 **SRP Section Number:** 2.4.5

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.47 (a)(1)

Summary:

(a) The requirements of this paragraph apply to all applications for design certification. (1) An application for design certification must contain: (i) The technical information which is required of applicants for construction permits and operating licenses by 10 CFR Part 20, Part 50 and its appendices, and Parts 73 and 100, and which is technically relevant to the design and not site-specific; (ii) Demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f); (iii) The site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters....

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Potential Impact Number: 23930 **SRP Section Number:** 2.4.5

Document: ADVANCE SER ABWR CH 2
Title: SITE CHARACTERISTICS

Summary:

The staff's review of the ABWR water level (flood) design is contained in Section 3.4 of this report. The bounding site parameters that were considered in the staff's evaluation are as follows:

- Floods: The ABWR shall be located on the site so that the level associated with the design-basis flood will be no higher than 30.5 cm (1 ft) below the plant grade.

- Potential Dam Failures (Seismically- Induced): Failure of existing and potential upstream or downstream water control structures shall not contribute to the water level exceeding 30.5 cm (1 ft) below grade nor compromise the ultimate heat sink.

- Probable Maximum Surge and Seiche Flooding: The probable maximum surge and seiche flooding shall be no higher than 30.5 cm (1 ft) below grade (see SSAR Table 2.1- 1).

- Probable Maximum Tsunami: The probable maximum tsunami flooding shall be no higher than 30.5 cm (1 ft) below grade (see SSAR Table 2.1-1).

- Ground Water: GE states in the SSAR that the ABWR design is intended to be compatible with ground water levels up to 61 cm (2 ft) below plant grade.

Potential Impact Number: 23931 **SRP Section Number:** 2.4.5

Document: ADVANCE SER CE CH 2
Title: SITE CHARACTERISTICS

Summary:

Pursuant to 10 CFR Part 52.47(a)(1)(iii) an applicant seeking design certification is required to (1) provide site parameters postulated for the design, and (2) analyze and

evaluate the design in terms of these parameters. ABB-CE provided a list of site parameters in CESSAR-DC Table 2.0-1. The staff's evaluation of the parameters in Table 2.0-1 follows. ..

Maximum Flood (or Tsunami) Level - In the CESSAR-DC, ABB-CE specifies that the site- specific maximum flood level is 0.3 m (1 ft) below grade. This flood level, which is consistent with the requirement of the EPRI Evolutionary Utility Requirements Document (URD), is acceptable for the ALWR design because the minimum design-basis flood level for preventing damages to seismic Category I SSC is at 0.3 m (1 ft) below plant grade. The maximum flood level is discussed in Section 3.4 of this report.

Potential Impact Number: 23932 **SRP Section Number:** 2.4.3

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.17 (a)(1)(vi)

Summary:

The application (for an early site permit) must contain the information required by 10 CFR 50.33 (a)-(d), the first three sentences of 50.34(a)(1), and, to the extent approval of emergency plans is sought under paragraph (b)(2)(ii) of this section, the information required by 50.33 (g) and (j), and 50.34(b)(6)(v). In particular, the application should describe the following:... (vi) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site (see Appendix A to 10 CFR Part 100)...

Potential Impact Number: 23933 **SRP Section Number:** 2.4.3

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.47 (a)(1)

Summary:

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(a) The requirements of this paragraph apply to all applications for design certification. (1) An application for design certification must contain: (i) The technical information which is required of applicants for construction permits and operating licenses by 10 CFR Part 20, Part 50 and its appendices, and Parts 73 and 100, and which is technically relevant to the design and not site-specific; (ii) Demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f); (iii) The site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters....

Potential Impact Number: 23934 **SRP Section Number:** 2.4.3

Document: ADVANCE SER ABWR CH 2
Title: SITE CHARACTERISTICS

Summary:

The staff's review of the ABWR water level (flood) design is contained in Section 3.4 of this report. The bounding site parameters that were considered in the staff's evaluation are as follows:

- Floods: The ABWR shall be located on the site so that the level associated with the design-basis flood will be no higher than 30.5 cm (1 ft) below the plant grade.

- Potential Dam Failures (Seismically- Induced): Failure of existing and potential upstream or downstream water control structures shall not contribute to the water level exceeding 30.5 cm (1 ft) below grade nor compromise the ultimate heat sink.

- Probable Maximum Surge and Seiche Flooding: The probable maximum surge and seiche flooding shall be no higher than 30.5 cm (1 ft) below grade (see SSAR Table 2.1- 1).

- Probable Maximum Tsunami: The probable maximum tsunami flooding shall be no higher than 30.5 cm (1 ft) below grade (see SSAR Table 2.1-1).

- Ground Water: GE states in the SSAR that the ABWR design is intended to be compatible with ground water levels up to 61 cm (2 ft) below plant grade.

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Potential Impact Number: 23935 **SRP Section Number:** 2.4.3

Document: ADVANCE SER CE CH 2
Title: SITE CHARACTERISTICS

Summary:

Pursuant to 10 CFR Part 52.47(a)(1)(iii) an applicant seeking design certification is required to (1) provide site parameters postulated for the design, and (2) analyze and evaluate the design in terms of these parameters. ABB-CE provided a list of site parameters in CESSAR-DC Table 2.0-1. The staff's evaluation of the parameters in Table 2.0-1 follows. ...

Maximum Flood (or Tsunami) Level - In the CESSAR-DC, ABB-CE specifies that the site- specific maximum flood level is 0.3 m (1 ft) below grade. This flood level, which is consistent with the requirement of the EPRI Evolutionary Utility Requirements Document (URD), is acceptable for the ALWR design because the minimum design-basis flood level for preventing damages to seismic Category I SSC is at 0.3 m (1 ft) below plant grade. The maximum flood level is discussed in Section 3.4 of this report.

Potential Impact Number: 23936 **SRP Section Number:** 2.4.6

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.17 (a)(1)(vi)

Summary:

The application (for an early site permit) must contain the information required by 10 CFR 50.33 (a)-(d), the first three sentences of 50.34(a)(1), and, to the extent approval of emergency plans is sought under paragraph (b)(2)(ii) of this section, the information required by 50.33 (g) and (j), and 50.34(b)(6)(v). In particular, the application should describe the following:... (vi) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site (see Appendix A to 10 CFR Part 100)...

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Potential Impact Number: 23937 **SRP Section Number:** 2.4.6

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.47 (a)(1)

Summary:

(a) The requirements of this paragraph apply to all applications for design certification. (1) An application for design certification must contain: (i) The technical information which is required of applicants for construction permits and operating licenses by 10 CFR Part 20, Part 50 and its appendices, and Parts 73 and 100, and which is technically relevant to the design and not site-specific; (ii) Demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f); (iii) The site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters....

Potential Impact Number: 23938 **SRP Section Number:** 2.4.6

Document: ADVANCE SER ABWR CH 2
Title: SITE CHARACTERISTICS

Summary:

The staff's review of the ABWR water level (flood) design is contained in Section 3.4 of this report. The bounding site parameters that were considered in the staff's evaluation are as follows:

- Floods: The ABWR shall be located on the site so that the level associated with the design-basis flood will be no higher than 30.5 cm (1 ft) below the plant grade.

- Potential Dam Failures (Seismically- Induced): Failure of existing and potential upstream or downstream water control structures shall not contribute to the water level exceeding 30.5 cm (1 ft) below grade nor compromise the ultimate heat sink.

- Probable Maximum Surge and Seiche Flooding: The probable maximum surge and seiche flooding shall be no higher than 30.5 cm (1 ft) below grade (see SSAR Table 2.1- 1).

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- Probable Maximum Tsunami: The probable maximum tsunami flooding shall be no higher than 30.5 cm (1 ft) below grade (see SSAR Table 2.1-1).

- Ground Water: GE states in the SSAR that the ABWR design is intended to be compatible with ground water levels up to 61 cm (2 ft) below plant grade.

Potential Impact Number: 23939 **SRP Section Number:** 2.4.6

Document: ADVANCE SER CE CH 2
Title: SITE CHARACTERISTICS

Summary:

Pursuant to 10 CFR Part 52.47(a)(1)(iii) an applicant seeking design certification is required to (1) provide site parameters postulated for the design, and (2) analyze and evaluate the design in terms of these parameters. ABB-CE provided a list of site parameters in CESSAR-DC Table 2.0-1. The staff's evaluation of the parameters in Table 2.0-1 follows

Maximum Flood (or Tsunami) Level - In the CESSAR-DC, ABB-CE specifies that the site- specific maximum flood level is 0.3 m (1 ft) below grade. This flood level, which is consistent with the requirement of the EPRI Evolutionary Utility Requirements Document (URD), is acceptable for the ALWR design because the minimum design-basis flood level for preventing damages to seismic Category I SSC is at 0.3 m (1 ft) below plant grade. The maximum flood level is discussed in Section 3.4 of this report.

Potential Impact Number: 23940 **SRP Section Number:** 2.5.4

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.17 (a)(1)(vi)

Summary:

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The application (for an early site permit) must contain the information required by 10 CFR 50.33 (a)-(d), the first three sentences of 50.34(a)(1), and, to the extent approval of emergency plans is sought under paragraph (b)(2)(ii) of this section, the information required by 50.33 (g) and (j), and 50.34(b)(6)(v). In particular, the application should describe the following:... (vi) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site (see Appendix A to 10 CFR Part 100)...

Potential Impact Number: 23941 **SRP Section Number:** 2.5.4

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.47 (a)(1)

Summary:

a) The requirements of this paragraph apply to all applications for design certification. (1) An application for design certification must contain: (i) The technical information which is required of applicants for construction permits and operating licenses by 10 CFR Part 20, Part 50 and its appendices, and Parts 73 and 100, and which is technically relevant to the design and not site-specific; (ii) Demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f); (iii) The site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters....

Potential Impact Number: 23943 **SRP Section Number:** 6.2.2

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

The fine debris that could be introduced into the IRWST is prevented from entering the SIS suction header piping by a debris screen. These screens are located at each end of the four wing walls. These wing wall assemblies extend from the IRWST floor to above the maximum IRWST water level. The wing wall screens have the capability to

remove particles greater than 0.23 cm (0.09 in.) in diameter. The IRWST screen design is described in CESSAR-DC Appendix 19.8A, Section 2.9.

Section 6.8.2.2.1 of CESSAR-DC requires that the COL applicant submit an analysis, consistent with RG 1.82, of the suction inlet screen area based on the insulation type and quantity in the containment. This analysis must show that the System 80+ screen are is at least three times over that indicated by RG 1.82. The staff finds this commitment sufficient to meet the staff's current position that emergency core cooling system (ECCS) suction strainers be sized in accordance with RG 1.82, Revision 1, but with a factor of three sizing margin.

Potential Impact Number: 23944 **SRP Section Number:** 3.9.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Repeated thermal cycling of RCS components produces some degree of fatigue degradation of the materials which could lead to failure, thereby increasing the likelihood of a LOCA.

STS for newer OLs require licensees to keep account of the number of transient occurrences to ensure that transient limits, based on design assumptions, are not exceeded. However, a number of older plants for which detailed fatigue analyses were performed on ASME Section III, Code Class 1 (reactor coolant pressure boundary) components do not have TS requirements for monitoring actual transient occurrences. These transients could significantly affect the fatigue life of the RCS. This issue originated as MPA B-70, but a letter to licensees was never issued for the collection of information.

Potential Impact Number: 23954 **SRP Section Number:** 3.9.3

Document: NUREG 1367
Title: FUNCTIONAL CAPABILITY OF PIPING SYSTEMS

Summary:

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Microfiche 64401:001. NUREG-1367 is referenced in the ABWR ASER and CE 80+ ASER. General Design Criterion 1 of Appendix A to Part 50 of Title 10 of the Code of Federal Regulations requires, in part, that structures, systems, and components important to safety be designed to withstand the effects of earthquakes without a loss of capability to perform their safety function. The function of a piping system is to convey fluids from one location to another. The functional capability of a piping system might be lost if, for example, the cross-sectional flow area of the pipe were deformed to such an extent that the required flow through the pipe would be restricted. The objective of this report is to examine the present rules in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, and potential changes to these rules, to determine if they are adequate for ensuring the functional capability of safety-related piping systems in nuclear power plants.

Potential Impact Number: 23958 **SRP Section Number:** 2.4.7

Document: C&S: ANSI N170-76
Title: Standards for Determining Design Basis Flooding at Power Reactor Sites

Summary:

SRP Section 2.4.7 cites ANSI N170 1976.

The current version is ANS 2.8 1992.

Potential Impact Number: 23959 **SRP Section Number:** 2.4.8

Document: C&S: ANSI N170
Title: Standards for Determining Design Basis Flooding at Power Reactor Sites

Summary:

SRP Section 2.4.8 cites ANSI N170 with no date specified.

The current version is ANS 2.8 1992.

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Potential Impact Number: 23960 **SRP Section Number:** 3.6.1

Document: C&S: IEEE 308
Title: IEEE STANDARD CRITERIA FOR CLASS 1E POWER SYSTEMS FOR NUCLEAR POWER GENERATING STATIONS

Summary:

SRP Section 3.6.1 cites IEEE 308 with no date specified.

The current version is IEEE 308 1991.

Potential Impact Number: 23961 **SRP Section Number:** 3.6.1

Document: C&S: IEEE 279
Title: STANDARD CRITERIA FOR SAFETY SYSTEMS FOR NUCLEAR POWER GENERATING STATIONS

Summary:

SRP Section 3.6.1 cites IEEE 279 with no date specified.

The current version is IEEE 603 1991.

Potential Impact Number: 23964 **SRP Section Number:** 3.8.3

Document: C&S: AISC Specifications
Title: Specification for Design, Fabrication and Erection of Structural Steel for Buildings

Summary:

SRP Section 3.8.3 cites AISC Specifications with no date specified.

The current version is AISC S327 1984.

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Potential Impact Number: 23966 **SRP Section Number:** 3.8.3

Document: C&S: ACI 318
Title: Building Code Requirements for Reinforced Concrete

Summary:

SRP Section 3.8.3 cites ACI 318 with no date specified.

The current version is ACI 318 1992.

Potential Impact Number: 23967 **SRP Section Number:** 3.8.3

Document: C&S: ANSI N45.2.5
Title: Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete, Structural Steel, Soils, and Foundations

Summary:

SRP Section 3.8.3 cites ANSI N45.2.5 with no date specified.

ASME N45.2.5 1978 was in effect at the time of the last SRP revision and is still current.

Potential Impact Number: 23968 **SRP Section Number:** 3.8.4

Document: C&S: ACI 318
Title: Building Code Requirements for Reinforced Concrete

Summary:

SRP Section 3.8.4 cites ACI 318 with no date specified.

The current version is ACI 318 1992.

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Potential Impact Number: 23969 **SRP Section Number:** 3.8.4

Document: C&S: ICBO UBC-79
Title: UNIFORM BUILDING CODE

Summary:

SRP Section 3.8.4 cites Uniform Building Code 1979.

The current version is Uniform Building Code 1991.

Potential Impact Number: 23970 **SRP Section Number:** 3.8.4

Document: C&S: AISC Specifications
Title: Specification for Design, Fabrication and Erection of Structural Steel for Buildings

Summary:

SRP Section 3.8.4 cites AISC with no date specified.

The current version is AISC S327 1984.

Potential Impact Number: 23971 **SRP Section Number:** 3.8.4

Document: C&S: ACI 349
Title: Code Requirements for Nuclear Safety-Related Concrete Structures

Summary:

SRP Section 3.8.4 cites ACI 349 with no date specified.

The current version is ACI 349 1990.

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Potential Impact Number: 23972 **SRP Section Number:** 3.8.4

Document: C&S: ACI 531-79
Title: Building Code Requirements for Masonry Structures

Summary:

SRP Section 3.8.4 cites ACI 531 1979 and Commentary ACI 531R 1979.

The current version is ACI 530 1992.

Potential Impact Number: 23973 **SRP Section Number:** 3.8.4

Document: C&S: ANSI N210-76
Title: Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants, Design

Summary:

SRP Section 3.8.4 cites ANSI N210 1976.

The current version is ANS 57.2 1983.

Potential Impact Number: 23974 **SRP Section Number:** 3.8.5

Document: C&S: AISC Specifications
Title: Specification for Design, Fabrication and Erection of Structural Steel for Buildings

Summary:

SRP Section 3.8.5 cites AISC with no date specified.

The current version is AISC S327 1984.

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Potential Impact Number: 23975 **SRP Section Number:** 3.8.5

Document: C&S: ACI 349
Title: Code Requirements for Nuclear Safety-Related Concrete Structures

Summary:

SRP Section 3.8.5 cites ACI 349 with no date specified.

The current version is ACI 349 1990.

Potential Impact Number: 23977 **SRP Section Number:** 3.10

Document: C&S: ANSI N278.1-75
Title: Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard, Reactor Plants and their Maintenance

Summary:

SRP Section 3.10 cites ANSI N278.1 1975.

The cited version was reaffirmed in 1984.

Potential Impact Number: 23979 **SRP Section Number:** 3.11

Document: C&S: IEEE 627
Title: Standard for Design Qualification of Safety Systems Equipment Used in Nuclear Power Generating Stations

Summary:

SRP Section 3.11 cites IEEE 627 with no date specified.

IEEE 627 1980 was in effect at the time of the last SRP revision and is still current.

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Potential Impact Number: 23980 **SRP Section Number:** 3.11

Document: C&S: IEEE 650
Title: Standard for Qualification of Class IE Static Battery Chargers and Inverters for Nuclear Power Generating Stations

Summary:

SRP Section 3.11 cites IEEE 650 with no date specified.

The current version is IEEE 650 1990.

Potential Impact Number: 23981 **SRP Section Number:** 3.11

Document: C&S: IEEE 649
Title: Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations

Summary:

SRP Section 3.11 cites IEEE 649 with no date specified.

The current version is IEEE 649 1991.

Potential Impact Number: 23982 **SRP Section Number:** 3.11

Document: C&S: IEEE 383
Title: TYPE TEST OF CLASS 1E ELECTRIC CABLES FIELD SPLICES AND CONNECTIONS

Summary:

SRP Section 3.11 cites IEEE 383 with no date specified.

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The current version is IEEE 383 1974 R92.

Potential Impact Number: 23983 **SRP Section Number:** 3.11

Document: C&S: IEEE 381
Title: Standard Criteria for Type Tests of Class IE Modules Used in Nuclear Power Generating Stations

Summary:

SRP Section 3.11 cites IEEE 381 with no date specified.

The current version is 1977 R84 now withdrawn.

Potential Impact Number: 23984 **SRP Section Number:** 3.11

Document: C&S: IEEE 317
Title: IEEE STANDARD FOR ELECTRIC PENETRATION ASSEMBLIES IN CONTAINMENT STRUCTURES FOR NUCLEAR POWER GENERATING STATIONS

Summary:

SRP Section 3.11 cites IEEE 317 with no date specified.

The current version is IEEE 317 1983 R92.

Potential Impact Number: 23987 **SRP Section Number:** 5.2.2

Document: C&S: IEEE 279
Title: STANDARD CRITERIA FOR SAFETY SYSTEMS FOR NUCLEAR POWER GENERATING STATIONS

Summary:

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SRP Section 5.2.2 cites IEEE 279 with no date specified.

The current version is IEEE 603 1991.

Potential Impact Number: 23990 **SRP Section Number:** 5.3.2

Document: C&S: ASTM E208
Title: Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels

Summary:

SRP Section 5.3.2 cites ASTM E208 with no date specified.

The current version is ASTM E208 1991.

Potential Impact Number: 23991 **SRP Section Number:** 5.4.7

Document: C&S: IEEE 338
Title: IEEE STANDARD CRITERIA FOR THE PERIODIC SURVEILLANCE TESTING OF NUCLEAR POWER GENERATING STATION SAFETY SYSTEMS

Summary:

SRP Section 5.4.7 cites IEEE 338 with no date specified.

The current version is IEEE 338 1987 R93.

Potential Impact Number: 23993 **SRP Section Number:** 6.3

Document: C&S: ANSI N658
Title: Single Failure Criteria for PWR Fluid systems

Summary:

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SRP Section 6.3 cites ANSI N658 with no date specified.

This standard was also published as ANS 58.9. The 1981 version was in effect at the time of the last SRP revision and was reaffirmed in 1987.

Potential Impact Number: 23994 **SRP Section Number:** 8.1

Document: C&S: IEEE 317
Title: IEEE STANDARD FOR ELECTRIC PENETRATION ASSEMBLIES IN CONTAINMENT STRUCTURES FOR NUCLEAR POWER GENERATING STATIONS

Summary:

SRP Section 8.1 cites IEEE 317 with no date specified.

The current version is IEEE 317 1983 R92.

Potential Impact Number: 23995 **SRP Section Number:** 8.2

Document: C&S: ANSI C37.04
Title: Standard Rating Structure for AC High-Voltage Circuit Breakers Rated on a Symmetrical Basis

Summary:

SRP Section 8.2 cites ANSI C37.04 with no date specified.

The 1979 version was in effect when the SRP was last revised.

The 1979 version was reaffirmed in 1989.

Potential Impact Number: 23996 **SRP Section Number:** 8.2

Document: C&S: ANSI C37.09

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Title: Standard Test Procedure for AC High-Voltage Circuit Breakers Rated on a Symmetrical Current Basis

Summary:

SRP Section 8.2 cites ANSI C37.09 with no date specified.

The 1979 version was in effect when the SRP was last revised.

The 1979 version was reaffirmed in 1989.

Potential Impact Number: 23997 **SRP Section Number:** 8.3.1

Document: C&S: IEEE 338
Title: IEEE STANDARD CRITERIA FOR THE PERIODIC SURVEILLANCE TESTING OF NUCLEAR POWER GENERATING STATION SAFETY SYSTEMS

Summary:

SRP Section 8.3.1 cites IEEE 338 with no date specified.

The current version is IEEE 338 1987 R93.

Potential Impact Number: 23998 **SRP Section Number:** 8.3.1

Document: C&S: IEEE 317
Title: IEEE STANDARD FOR ELECTRIC PENETRATION ASSEMBLIES IN CONTAINMENT STRUCTURES FOR NUCLEAR POWER GENERATING STATIONS

Summary:

SRP Section 8.3.1 cites IEEE 317 with no date specified.

The current version is IEEE 317 1983 R92.

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Potential Impact Number: 23999 **SRP Section Number:** 8.3.1

Document: C&S: IEEE 379
Title: IEEE STANDARD APPLICATION OF THE SINGLE-FAILURE CRITERION TO NUCLEAR POWER GENERATING STATION SAFETY SYSTEMS

Summary:

SRP Section 8.3.1 cites IEEE 379 with no date specified.

The current version is IEEE 379 1988.

Potential Impact Number: 24000 **SRP Section Number:** 8.3.1

Document: C&S: IEEE 384
Title: IEEE STANDARD CRITERIA FOR INDEPENDENCE OF CLASS 1E EQUIPMENT AND CIRCUITS

Summary:

SRP Section 8.3.1 cites IEEE 384 with no date specified.

The current version is IEEE 384 1992.

Potential Impact Number: 24001 **SRP Section Number:** 8.3.1

Document: C&S: IEEE 384
Title: IEEE STANDARD CRITERIA FOR INDEPENDENCE OF CLASS 1E EQUIPMENT AND CIRCUITS

Summary:

SRP Section 8.3.1 cites IEEE 304 with no date specified.

This is a typo and should have bee IEEE 384.

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The current version is IEEE 384 1992.

Potential Impact Number: 24002 **SRP Section Number:** 8.3.1

Document: C&S: IEEE 387
Title: IEEE STANDARD CRITERIA FOR DIESEL-GENERATOR UNITS APPLIED AS STANDBY POWER SUPPLIES FOR NUCLEAR POWER GENERATING STATIONS

Summary:

SRP Section 8.3.1 cites IEEE 387 with no date specified.

The current version is 1984. IEEE 387 was withdrawn in 1993.

Potential Impact Number: 24003 **SRP Section Number:** 8.3.1

Document: C&S: IEEE 308
Title: IEEE STANDARD CRITERIA FOR CLASS 1E POWER SYSTEMS FOR NUCLEAR POWER GENERATING STATIONS

Summary:

SRP Section 8.3.1 cites IEEE 308 with no date specified.

The current version is IEEE 308 1991.

Potential Impact Number: 24010 **SRP Section Number:** 9.1.4

Document: C&S: ANS 57.1
Title: Design Requirements for Light Water Reactor Fuel Handling Systems

Summary:

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SRP Section 9.1.4 cites ANS 57.1 / ANSI N208 with no date specified.

The current version is ANS 57.1 1992.

Potential Impact Number: 24011 **SRP Section Number:** 9.1.5

Document: C&S: ANS 57.2
Title: Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants, Design

Summary:

SRP Section 9.1.5 cites ANS 57.2 / ANSI N210 with no date specified.

The current version is ANS 57.2 1983.

Potential Impact Number: 24012 **SRP Section Number:** 9.1.5

Document: C&S: ANS 57.1
Title: Design Requirements for Light Water Reactor Fuel Handling Systems

Summary:

SRP Section 9.1.5 cites ANS 57.1 / ANSI N208 with no date specified.

The current version is ANS 57.1 1992.

Potential Impact Number: 24013 **SRP Section Number:** 9.2.2

Document: C&S: IEEE 279
Title: STANDARD CRITERIA FOR SAFETY SYSTEMS FOR NUCLEAR POWER GENERATING STATIONS

Summary:

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SRP Section 9.2.2 cites IEEE 279 with no date specified.

The current version is IEEE 603 1991.

Potential Impact Number: 24014 **SRP Section Number:** 9.5.4

Document: C&S: DEMA Standard
Title: Diesel Manufacturers Association (DEMA) Standard

Summary:

SRP Section 9.5.4 cites DEMA with no date specified.

DEMA 1974 was in effect at the time of the last SRP revision and is still current.

Potential Impact Number: 24015 **SRP Section Number:** 9.5.5

Document: C&S: DEMA Standard
Title: Diesel Manufacturers Association (DEMA) Standard

Summary:

SRP Section 9.5.5 cites DEMA with no date specified.

DEMA 1974 was in effect at the time of the last SRP revision and is still current.

Potential Impact Number: 24016 **SRP Section Number:** 9.5.7

Document: C&S: DEMA Standard
Title: Diesel Manufacturers Association (DEMA) Standard

Summary:

SRP Section 9.5.7 cites DEMA with no date specified.

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DEMA 1974 was in effect at the time of the last SRP revision and is still current.

Potential Impact Number: 24017 **SRP Section Number:** 9.5.8

Document: C&S: DEMA Standard
Title: Diesel Manufacturers Association (DEMA) Standard

Summary:

SRP Section 9.5.8 cites DEMA with no date specified.

DEMA 1974 was in effect at the time of the last SRP revision and is still current.

Potential Impact Number: 24018 **SRP Section Number:** 12.3 - 12.4

Document: C&S: ANSI N13.1-69
Title: Guide for Sampling Airborne Radioactive Materials in Nuclear Facilities

Summary:

SRP Section 12.3 - 12.4 cites ANSI N13.1 1969.

The cited version was reaffirmed in 1993.

Potential Impact Number: 24019 **SRP Section Number:** 12.5

Document: C&S: ANSI N13.6-72
Title: Practice for Occupational Radiation Exposure Record Systems

Summary:

SRP Section 12.5 cites ANSI N13.6 1972.

The cited version was reaffirmed in 1989.

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Potential Impact Number: 24020 **SRP Section Number:** 12.5

Document: C&S: ANSI N13.5-72
Title: Performance Specifications for Direct-Reading and Indirect-Reading Pocket Dosimeters for X- and Gamma Radiation

Summary:

SRP Section 12.5 cites ANSI N13.5 1972.

The cited version was reaffirmed in 1989.

Potential Impact Number: 24021 **SRP Section Number:** 12.5

Document: C&S: ANSI N13.2-69
Title: Guide to Administrative Practices in Radiation Monitoring

Summary:

SRP Section 12.5 cites ANSI N13.2 1969.

The cited version was reaffirmed in 1988.

Potential Impact Number: 24022 **SRP Section Number:** 13.1.2 - 13.1.3

Document: C&S: ANSI N18.1
Title: Selection, Qualification and Training of Personnel for Nuclear Power Plants

Summary:

SRP Section 13.1.2-13.1.3 cites ANSI N18.1 with no date specified.

The current version is ANS 3.1 1993.

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Potential Impact Number: 24023 **SRP Section Number:** 13.1.2 - 13.1.3

Document: C&S: ANSI N18.7
Title: Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants

Summary:

SRP Section 13.1.2-13.1.3 cites ANSI N18.7 with no date specified.

The current version is ANS 3.2 1988.

Potential Impact Number: 24026 **SRP Section Number:** 13.2.2

Document: C&S: ANS 3.1
Title: Selection, Qualification and Training of Personnel for Nuclear Power Plants

Summary:

SRP Section 13.2.2 cites ANS 3.1 with no date specified.

The current version is ANS 3.1 1993.

Potential Impact Number: 24027 **SRP Section Number:** 13.4

Document: C&S: ANSI N18.7
Title: Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants

Summary:

SRP Section 13.4 cites ANSI N18.7 with no date specified.

The current version is ANS 3.2 1988.

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Potential Impact Number: 24028 **SRP Section Number:** 13.4

Document: C&S: ANS 3.1-78
Title: Selection, Qualification and Training of Personnel for Nuclear Power Plants

Summary:

SRP Section 13.4 cites ANS 3.1 1978.

The current version is ANS 3.1 1993.

Potential Impact Number: 24029 **SRP Section Number:** 13.4

Document: C&S: ANSI N18.1
Title: Selection, Qualification and Training of Personnel for Nuclear Power Plants

Summary:

SRP Section 13.4 cites ANSI N18.1 with no date specified.

The current version is ANS 3.1 1993.

Potential Impact Number: 24030 **SRP Section Number:** 13.4

Document: C&S: ANS 3.2-78
Title: Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants

Summary:

SRP Section 13.4 cites ANS 3.2 1978.

The current version is ANS 3.2 1988.

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Potential Impact Number: 24031 **SRP Section Number:** 13.5.1.1

Document: C&S: ANS 3.2
Title: Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants

Summary:

SRP Section 13.5.1 cites ANS 3.2 with no date specified.

The current version is ANS 3.2 1988.

Potential Impact Number: 24034 **SRP Section Number:** 13.6

Document: C&S: ANSI N18.17
Title: Industrial Security for Nuclear Power Plants

Summary:

SRP Section 13.6 cites ANSI N18.17 with no date specified.

The latest version of this standard is ANS 3.3 1988.

Potential Impact Number: 24037 **SRP Section Number:** 15.5.1 - 15.5.2

Document: C&S: ANS N212
Title: Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants

Summary:

SRP Section 15.5.1-15.5.2 cites ANS Trial Use Standard N212 with no date specified.

The current version is ANS 52.1 1983 R88.

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Potential Impact Number: 24038 **SRP Section Number:** 15.5.1 - 15.5.2

Document: C&S: ANSI N18.2-74
Title: Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants

Summary:

SRP Section 15.5.1-15.5.2 cites ANSI N18.2 1974.

The current version is ANS 51.1 1983 R88.

Potential Impact Number: 24039 **SRP Section Number:** 17.1

Document: C&S: NFPA 232
Title: Standard for the Protection of Records

Summary:

SRP Section 17.1 cites NFPA 232 with no date specified.

The current version is NFPA 232 1991.

Potential Impact Number: 24041 **SRP Section Number:** 17.2

Document: C&S: ANSI N45.2.12
Title: Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants

Summary:

SRP Section 17.2 cites ANSI N45.2.12 with no date specified.

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ASME N45.2.12 1977 was in effect at the time of the last SRP revision and is still current.

Potential Impact Number: 24044 **SRP Section Number:** 13.6

Document: FINAL SER EPRI CH 9
Title: EPRI URD-EPD: SITE SUPPORT SYSTEMS

Summary:

The basis for the staff's review of Section 5 of Chapter 9 includes 10 CFR 73.55; Section 13.6, Revision 2, "Physical Security," of the SRP (NUREG-0800); and NUREG-0908, "Acceptance Criteria for the Evaluation of Nuclear Power Reactor Security Plans."

Potential Impact Number: 24045 **SRP Section Number:** 13.6

Document: NUREG 0908
Title: ACCEPTANCE CRITERIA FOR THE EVALUATION OF NUCLEAR POWER REACTOR SECURITY PLANS

Summary:

NUREG-0908 was developed to assist the NRC security plan review process of new or revised plans and licensee development of new or revised plans by outlining specific criteria for the elements of plans.

Section 3 outlines criteria and guidance 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.

NUREG-0908 (Microfiche 15516:132) was used by the NRC Staff in their review of the EPRI Evolutionary Plant requirements document.

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Potential Impact Number: 24046 **SRP Section Number:** 13.6

Document: NUREG 0908
Title: ACCEPTANCE CRITERIA FOR THE EVALUATION OF NUCLEAR POWER REACTOR SECURITY PLANS

Summary:

NUREG-0908 has been developed to assist the NRC security plan review process of new or revised plans and licensee development of new or revised plans by outlining specific criteria for the elements of 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.

NUREG-0908 (Microfiche 15516:132) was used by the NRC Staff in their review of the EPRI Evolutionary Plant requirements document.

Potential Impact Number: 24047 **SRP Section Number:** 13.6

Document: FINAL SER ABWR CH 13
Title: CONDUCT OF OPERATIONS

Summary:

The staff is satisfied that the list of vital equipment includes all active and passive plant equipment essential to safe shutdown of the reactor, including necessary support systems, the reactor vessel and the remainder of the reactor coolant system pressure boundary within primary containment, the suppression pool, spent fuel in the fuel pool, and any associated piping, equipment, and controls whose failure could result in an offsite release in excess of 10 CFR Part 100 limits. The staff finds this to be compatible with NRC Review Guideline 17 (January 23, 1978, memorandum from R. Clark to safeguards licensing staff).

The staff expects that at least 60 days before loading fuel, a licensee shall have confirmed that no portion of as-built vital systems are located outside of designated vital

areas or can be prevented from performing their safety functions from outside the vital areas (e.g., by reach-rod valve manipulation). This verification should include piping, valves, and motor control centers that are required for boundary integrity, for performance of safe reactor shutdown cooling function, and to isolate safety-related equipment from non-safety-related equipment. This was identified as DFSER COL Action Item 13.6.3.3-2. By SSAR Amendment 25, GE identified in Section 13.6.3.11 that the COL applicant will confirm that locations of vital systems and system operations met the above requirements.

SSAR Section 13.6.3.6 discusses bullet-resistant walls and doors and security grills and screens incorporated into the building design, with the stated intent of minimizing forcible access to the control room. Responses to RAIs Q910.13, Q910.23, Q910.24 did not resolve staff uncertainty as to the adequacy of barriers in all man-sized openings in physical barriers that separate other vital from non-vital areas. Also, the staff position on the effectiveness of the ventilation system barriers described in SSAR Section 13.6.3.6 remained as described in RAI Q910.13; that is, consideration may need to be given to how accessible, isolated, and hidden from view these barriers will be, as well as whether they can be penetrated with hand tools available on site.

Potential Impact Number: 24048 **SRP Section Number:** 13.6

Document: FINAL SER CE80 CH 13
Title: CONDUCT OF OPERATIONS

Summary:

In accordance with SRP Section 13.6 and Review Guideline 17 of the NRC Security Plan Evaluation Report Workbook, the staff considers that all seismic Category I systems should be designated as vital equipment in the sense of 10 CFR Part 73.

Potential Impact Number: 24049 **SRP Section Number:** 13.6

Document: FINAL SER EPRI CH 9
Title: EPRI URD-EPD: SITE SUPPORT SYSTEMS

Summary:

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Potential Impacts

In the DSER for Chapter 9, the staff identified guidance designation of vital equipment as a confirmatory issue because EPRI had committed to modify Section 5.2.1 of Chapter 9 to reference NRC Review Guideline 17 instead of NUREG-1178 as the applicable guidance. The staff has verified that Chapter 9 of the Evolutionary Requirements Document contains this requirement. Therefore, this DSER confirmatory issue is closed.

Potential Impact Number: 24050 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: 73.70(d)

Summary:

A log indicating name, badge number, time of entry, and time of exit of all individuals granted access to a vital area except those individuals entering or exiting the reactor control room. The licensee shall retain this log as a record for three years after the last entry is made in the log.

Potential Impact Number: 24051 **SRP Section Number:** 13.6

Document: FINAL SER ABWR CH 13
Title: CONDUCT OF OPERATIONS

Summary:

The types of door controls specified in SSAR Section 13.6.3.4 are generally acceptable, but insufficient detail is provided to determine compatibility with RG 5.12, Revision 0. This description of access control methods also does not address the positive control requirement of 10 CFR 73.55(d)(7)(i)(B) and the record-keeping requirement of 10 CFR 73.70(d), which requires logging individuals' times of entry to and exit from each vital area. This was identified as DSER (SECY-91-235) Open Item 36 and DFSER COL Action Item 13.6.3.5-1. By SSAR Amendment 25, GE identified in Section 13.6.3.13 that the COL applicant will demonstrate that door controls are compatible with RG 5.12, the positive control requirement of 10 CFR 73.55(d)(7)(i)(B) and the record-keeping

requirement of 10 CFR 73.70(d). This issue will be considered during review of the plant-specific security plan.

Furthermore, SSAR Chapter 19, Appendix 19B.3.10, requires a COL applicant to evaluate the effects of the security system on required operator actions during all emergency modes of operation. The staff's position is that this analysis 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. This was identified as DSER (SECY-91-235) Open Item 37 and DFSER COL Action Item 13.6.3.5-2. By SSAR Amendment 25, GE identified in Section 13.6.3.8 that the COL applicant will provide site-specific security, contingency, and guard training plans. Evaluation of compliance with the vital equipment prompt access requirements of 10 CFR 73.55(d)(7)(ii) will be resolved during review of the plant-specific security plan.

Potential Impact Number: 24052 **SRP Section Number:** 13.6

Document: NUREG 0908
Title: ACCEPTANCE CRITERIA FOR THE EVALUATION OF NUCLEAR POWER REACTOR SECURITY PLANS

Summary:

NUREG-0908 has been developed to assist the NRC security plan review process of new or revised plans and licensee development of new or revised plans by outlining specific criteria for the elements of 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.

NUREG-0908 (Microfiche 15516:132) was used by the NRC Staff in their review of the EPRI Evolutionary Plant requirements document.

Potential Impact Number: 24053 **SRP Section Number:** 13.6

Document: FINAL SER EPRI CH 6
Title: EPRI URD-EPD: BUILDING DESIGN AND ARRANGEMENT

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Summary:

Section 2.3.5.4 of Chapter 6 specifies that, to the extent practicable, plant arrangement and building design features will be used to help protect against insider sabotage. The rationale portion of this section states that this requirement, coupled with appropriate administrative procedures, will eliminate the possibility that a badged individual can reach and enter all divisions of safety equipment within a short time. In its June 24, 1991, letter, EPRI committed to change Chapter 6, Sections 1.5.2 and 2.3.5, and Chapter 9, Section 5.2.4.1, to require security controls access to vital areas only at vital area boundaries and not between redundant divisions of vital components. The staff has confirmed that these changes have been made and resolve its concerns about the possibility that the security system might interfere with safe plant operation. Protection against insider sabotage still will be provided by vital area access controls. Additional protection against insider sabotage may result from the vulnerability analysis required in Chapter 9. Further protection against insider sabotage will be provided by access authorization and fitness-for-duty programs required by 10 CFR Parts 73 and 26.

Potential Impact Number: 24054 **SRP Section Number:** 13.6

Document: NUREG 0908
Title: ACCEPTANCE CRITERIA FOR THE EVALUATION OF NUCLEAR POWER REACTOR SECURITY PLANS

Summary:

NUREG-0908 has been developed to assist the NRC security plan review process of new or revised plans and licensee development of new or revised plans by outlining specific criteria for the elements of 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.

NUREG-0908 (Microfiche 15516:132) was used by the NRC Staff in their review of the EPRI Evolutionary Plant requirements document.

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Potential Impacts

Potential Impact Number: 24055 **SRP Section Number:** 13.6

Document: NUREG 0908
Title: ACCEPTANCE CRITERIA FOR THE EVALUATION OF NUCLEAR POWER REACTOR SECURITY PLANS

Summary:

NUREG-0908 has been developed to assist the NRC security plan review process of new or revised plans and licensee development of new or revised plans by outlining specific criteria for the elements of plans.

Section 7 outlines criteria and guidelines for general communication among elements of the security force and with law enforcement authorities.

NUREG-0908 (Microfiche 15516:132) was used by the NRC Staff in their review of the EPRI Evolutionary Plant requirements document.

Potential Impact Number: 24056 **SRP Section Number:** 13.6

Document: FINAL SER CE80 CH 13
Title: CONDUCT OF OPERATIONS

Summary:

In CESSAR-DC Section 13.6.1(E), ABB-CE requires security officers to have equipment for continuous communication with the alarm stations and the plant control room to have dedicated communications to the central alarm system (CAS) and the secondary alarm system (SAS). The staff was not satisfied that ABB-CE had adequately addressed ensuring the transmissibility of radio signals as well as equipment shielding to prevent radios from interfering with plant equipment. This issue was designated as part of DSER Open Item 9.5.2-3. The staff's resolution of this issue is discussed in Section 9.5.2 of this report.

Potential Impact Number: 24057 **SRP Section Number:** 13.6

Document: NUREG 0908

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Potential Impacts

Title: ACCEPTANCE CRITERIA FOR THE EVALUATION OF NUCLEAR POWER REACTOR SECURITY PLANS

Summary:

NUREG-0908 has been developed to assist the NRC security plan review process of new or revised plans and licensee development of new or revised plans by outlining specific criteria for the elements of plans.

Section 8 outlines criteria and guidelines for evaluation and audit of the physical security program.

NUREG-0908 (Microfiche 15516:132) was used by the NRC Staff in their review of the EPRI Evolutionary Plant requirements document.

Potential Impact Number: 24058 **SRP Section Number:** 13.6

Document: NUREG 0674
Title: SECURITY PERSONNEL TRAINING AND QUALIFICATION CRITERIA

Summary:

This document supplements information contained in NUREG-0576 and NUREG-0219. It identifies the basic criteria for use in evaluating the responsiveness of guard training plans to the requirements of Appendix B, 10 CFR 73. It should serve as a useful reference in the preparation of guard training plans as well as for staff review of those plans.

NUREG-0674 is cited in SRP 13.6 as Acceptance Criteria.

Potential Impact Number: 24073 **SRP Section Number:** 10.4.2

Document: RG 1.28 **Revision:** 3
Title: QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION)

Summary:

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Potential Impacts

RG 1.28 endorses ANSI/ASME NQA-1-1983 which includes the provisions of ANSI N45.2.13 on quality assurance requirements related to procurement.

Potential Impact Number: 24086 **SRP Section Number:** 5.2.4

Document: NRC BULLETIN 88-09
Title: THIMBLE TUBE THINNING IN WESTINGHOUSE REACTORS

Summary:

This bulletin was issued to holders of operating licenses or construction permits for Westinghouse plants that utilize bottom mounted instrumentation. The bulletin requested that addressees establish and implement an inspection program to periodically confirm incore neutron monitoring system thimble tube integrity. The request was based on operational experience with thimble tube thinning and the resulting concern of degradation of the RCS pressure boundary that could lead to a potentially non-isolable leak of reactor coolant. The staff noted in the bulletin that there are currently no inservice inspection or testing requirements for thimble tubes and now believes such inspections may be necessary in connection with GDC 14.

(Note: This potential impact was entered during a review of SRP Section 5.3.3 integrated impacts)

Potential Impact Number: 24095 **SRP Section Number:** 10.4.6

Document: FINAL SER ABWR CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

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).

Potential Impact Number: 24096 **SRP Section Number:** 10.4.6

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Potential Impacts

Document: RG 1.56 **Revision:** 1
Title: MAINTENANCE OF WATER PURITY IN BOILING WATER REACTORS

Summary:

Regulatory Position C.3 refers to ASTM D2187-71 for measurement of the total exchange capacity of the cation resin.

Potential Impact Number: 24105 **SRP Section Number:** 3.11

Document: C&S: ASTM D2220-68
Title: SPECIFICATION FOR POLY (VINYL CHLORIDE) INSULATION FOR WIRE AND CABLE 75 C OPERATION

Summary:

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.

Potential Impact Number: 24117 **SRP Section Number:** 6.5.1

Document: C&S: ANSI B124.1-71
Title: UL STANDARD FOR SAFETY TEST PERFORMANCE OF AIR FILTER UNITS

Summary:

RG 1.52 is cited in this SRP Section. ANSI B124.1 1971 is endorsed by RG 1.52. The latest version of this standard is UL 900 1985.

Potential Impact Number: 24124 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: 73.56

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Summary:

This section sets forth requirements for establishment and implementation of a personnel access authorization program as part of the site Physical Security Plan.

Potential Impact Number: 24125 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: 73.56(g)

Summary:

Each licensee shall audit its access authorization program within 12 months of the effective date of implementation of this program and at least every 24 months thereafter to ensure that the requirements of this section are satisfied.

Each licensee who accepts the access authorization program of a contractor or vendor as provided for by paragraph (a)(4) of this section shall have access to records and shall audit contractor or vendor programs every 12 months to ensure that the requirements of this section are satisfied. Licensees may accept audits of contractors and vendors conducted by other licensees. Each sharing utility shall maintain a copy of the audit report, to include findings, recommendations and corrective actions. Each licensee retains responsibility for the effectiveness of any contractor and vendor program it accepts and the implementation of appropriate corrective action.

Potential Impact Number: 24126 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: 73.57

Summary:

This section sets forth requirements for criminal history checks of individuals granted unescorted access to a nuclear power facility or access to safeguards information.

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Potential Impact Number: 24135 **SRP Section Number:** 15.4.8

Document: C&S: ICRP 2 1959
Title: PERMISSIBLE DOSE FOR INTERNAL RADIATION

Summary:

RG 1.77 is cited in this SRP Section. ICRP 2 1959 is endorsed by RG 1.77. The latest version of this standard is ICRP 30 1990.

Potential Impact Number: 24136 **SRP Section Number:** 15.6.4

Document: C&S: ICRP 2 1959
Title: PERMISSIBLE DOSE FOR INTERNAL RADIATION

Summary:

RG 1.5 is cited in this SRP Section. ICRP 2 1959 is endorsed by RG 1.5. The latest version of this standard is ICRP 30 1990.

Potential Impact Number: 24137 **SRP Section Number:** 15.6.5.A

Document: C&S: ICRP 2 1959
Title: PERMISSIBLE DOSE FOR INTERNAL RADIATION

Summary:

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.

Potential Impact Number: 24138 **SRP Section Number:** 15.6.5.A

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Potential Impacts

Document: C&S: ICRP 2 1959
Title: PERMISSIBLE DOSE FOR INTERNAL RADIATION

Summary:

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.

Potential Impact Number: 24150 **SRP Section Number:** 3.8.1

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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. 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 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.

Potential Impact Number: 24151 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR26
Title: FITNESS-FOR-DUTY PROGRAM

APPENDIX II
Potential Impacts

Summary:

Part 26 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, and by licensees authorized to possess, use, or transport formula quantities of strategic special nuclear material (SSNM).

Appendix A to Part 26 provides guidelines for drug and alcohol testing programs.

This PI supersedes PIs 22794 and 23745.

Potential Impact Number: 24152 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR26
Title: FITNESS-FOR-DUTY PROGRAM
Location: 26.80

Summary:

Each licensee subject to this Part shall audit the fitness-for-duty program nominally every 12 months. In addition, audits must be conducted. nominally every 12 months, of those portions of fitness-for-duty programs implemented by contractors and vendors.

Potential Impact Number: 24153 **SRP Section Number:** 3.8.1

Document: NUREG CR-5973 **Revision:** 0 , 1/1/93
Title: CODES AND STANDARDS AND OTHER GUIDANCE CITED IN REGULATORY DOCUMENTS

Summary:

Since the last revision of the SRP Section 3.8.1 many RGs and industry standards, such as ASME B&PV Code, Section III, Divisions 1 and 2 have changed.

The SRP does not reference specific issues of industry standards, implying that the latest edition endorsed by a RG is to be used. For example, RG 1.136, "Materials, Construction, and Testing of Concrete Containments," Rev.2, June 1981, endorses

only articles CC-1000, CC- 2000, CC-4000, CC-5000, and CC-6000 of the 1980 edition of the Code. 10CFR50, paragraph 50.55a, "Codes and Standards," endorses various Code revisions depending on its application.

Meanwhile, the Code has been revised and the latest revision, as indicated in the NUREG/CR 5973, has been issued in 1992.

The above indicates that there is a lack of consistency in the regulatory requirements. Furthermore, the revisions of the Code may not be acceptable to the staff. There is a need for a critical review of the revised Codes that are applicable to nuclear plants and guidance provided as appropriate.

Potential Impact Number: 24157 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Location: 73.21

Summary:

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.

Potential Impact Number: 24158 **SRP Section Number:** 13.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

The safety concern of this item deals with the consideration of alternatives to the basic design of nuclear power plants with the emphasis primarily on reduction of the vulnerability of reactors to industrial sabotage. Extensive efforts and resources are expended in designing nuclear plants to minimize the risk to the public health and

safety from equipment or system malfunction or failure. However, reduction of the vulnerability of reactors to industrial sabotage is treated as a plant physical security function and not as a plant design requirement. Although present reactor designs do provide a great deal of inherent protection against industrial sabotage, extensive physical security measures are still required to provide an acceptable level of protection. An alternate approach would be to more fully consider reactor vulnerabilities to sabotage along with economy, operability, reliability, maintainability, and safety during the preliminary design phase. Since emphasis is being placed on standardizing plants, it is especially important to consider measures which could reduce the vulnerability of reactors to sabotage. Design features to enhance physical protection must be consistent with present and future system safety requirements.

In resolving this issue, the staff concluded that insider sabotage at operating nuclear power plants has not been a significant problem in the U.S. Existing requirements (10 CFR 73.55) dealing with plant physical security, controlled access to vital areas, screening for reliable personnel, etc., appear to be effective. The staff found no design modification that would completely eliminate or mitigate the threat of insider sabotage. The staff's technical findings were published in NUREG-1267. Thus, this issue was RESOLVED and no new requirements were established.

Potential Impact Number: 24159 **SRP Section Number:** 13.6

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

The staff resolved this issue without the establishment of new requirements. However, future plants may decrease vulnerability to sabotage by several means. Each division of safety system functions should be totally independent and separated, both mechanically and electrically, as well as the inclusion of 3-hour fire barriers, physical protection of each division from flooding, and physical protection from pipe breaks, both inside and outside the containment. The site security system requirements should include ensuring the compatibility of security system requirements with the plant arrangement and safety system design, definition of vital systems, layout of vital components, security barriers, intrusion detection systems, isolation zone requirements, security alarms, access control, security communications, power supply, and data management. Since there exists a potential for sabotage by a "knowledgeable insider" with authorized access or for acts of sabotage that could occur

during maintenance activities, advanced light water reactor (ALWR) plant designers should also analyze the vulnerability of their designs to insider sabotage before finalizing the designs.

SSAR Section 19B.2.4 contains a summary that describes this issue, the Electric Power Research Institute (EPRI) ALWR requirements document, and the proposed resolution of Issue A-29 for the ABWR. It states that the ABWR design will comply with the ALWR requirements document as defined in the SSAR. It also indicates that the ABWR design will mitigate the acts of sabotage through physical separations in the plant arrangement of engineering safety systems and the design and location of barriers to resist threats.

In SSAR Section 19B.2.4, GE states that a sabotage vulnerability analysis shall be conducted prior to finalizing design. The staff has verified that GE established a COL action item in Table 1.9-1 addressing this requirement. This is an acceptable approach. Based on the above, the staff concludes that GE has adequately addressed this issue for the ABWR design.

Potential Impact Number: 24161 **SRP Section Number:** 13.6

Document: FINAL SER CE80 CH 13
Title: CONDUCT OF OPERATIONS

Summary:

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 Volume II, Chapter 9, Section 5.2.2.1 (Rev. 1) of the URD, Electric Power Research Institute (EPRI) states: "To optimize system design and compatibility of plant arrangement and system design with sabotage protection, a sabotage vulnerability analysis shall be conducted prior to finalizing design." In its summary and recommendations on design and equipment, the staff noted in NUREG-1267, "Technical Resolution of Generic Safety Issue A-29, `Nuclear Power Plant Design for Reduction of Vulnerability to Industrial Sabotage,'" (September 1987) the potential for insider access to and sabotage of redundant safety systems, particularly of "systems in standby mode that lack a status indicator in the control room," and recommended a systematic evaluation to identify these types of vulnerabilities and possible corrective actions.

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Potential Impacts

In Amendment T, ABB-CE revised Appendix 13A to incorporate the review for vulnerabilities to sabotage. The review used the System 80+ PRA as the basis for determining the vulnerabilities to sabotage. From this review, ABB-CE incorporated additional valves that will be alarmed in the control room and would enhance the ability to detect valve mispositions in critical systems. The final design, however, will still have to rely on site-specific installed security systems. As noted in Section 13.6.1 F of CESSAR-DC, a site-specific vulnerability analysis is required by the COL applicant to ensure that significant changes do not occur in the sabotage effects on each dominant accident sequence and, where practical, attempt to optimize the specific site system arrangements and design with respect to sabotage protection before finalizing the design. The staff intends to review the site-specific vulnerability analysis and final design during its review of the COL site-specific security, contingency, and guard training plans required by 10 CFR 50.34 and 10 CFR Part 73.

Potential Impact Number: 24162 **SRP Section Number:** 13.6

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

In its DSER for Chapter 9, the staff concluded that the Evolutionary Requirements Document had adequately addressed Issue A-29 with respect to external sabotage, but that it had not adequately addressed insider sabotage. At that time, the discussion of this issue in Appendix B to Chapter 9 of the Evolutionary Requirements Document made no reference to the potential for sabotage by a "knowledgeable insider" with authorized access or to acts of sabotage that could occur during maintenance activities. In the DSER, the staff noted that in a letter to the NRC dated August 18, 1989, EPRI had committed to expand Section 5.2.2.1 of Chapter 9 to require that ALWR plant designers analyze the vulnerability of their designs to insider sabotage before finalizing the designs. The staff has confirmed that this section of the Evolutionary Requirements Document includes a requirement for the analysis to consider both insider and outsider threats. Therefore, this confirmatory issue is resolved.

Potential Impact Number: 24168 **SRP Section Number:** 3.9.1

Document: FINAL SER CE80 CH 20

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Potential Impacts

Title: GENERIC ISSUES

Summary:

The resolution of Issue 79 is documented in GL 92-02 and NUREG-1374. There were no new design requirements in GLT 92-02. GI-79 was resolved based on detailed analysis of the St. Lucie vessel NCC stresses.

By DSER Open Item 20.2-11, the staff requested that ABB-CE submit additional information regarding the applicability of the St. Lucie and System 80 analyses to System 80+.

The staff also requested in the DSER that ABB-CE verify the number of NCC events for the lifetime of the System 80+ design (for inclusion in code fatigue analysis). NUREG-0933, NCC events are estimated to occur about 0.04 times per year, which would be about 3 times in 60 years, and, thus, the commitment of ABB-CE to include 30 NCC events in the System 80+ design is acceptable.

Potential Impact Number: 24171 **SRP Section Number:** 6.5.1

Document: C&S: ANSI N509-76
Title: Nuclear Power Plant Air Cleaning Units and Components

Summary:

RG 1.52 is cited in this SRP Section. ANSI N509 1976 is endorsed by RG 1.52. The latest version of this standard is ASME N509 1993.

Potential Impact Number: 24172 **SRP Section Number:** 6.5.1

Document: C&S: ANSI N510-75
Title: Testing of Nuclear Air Treatment Systems

Summary:

RG 1.52 is cited in this SRP Section. ANSI N510 1975 is endorsed by RG 1.52. The latest version of this standard is ASME N510 1989.

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Potential Impacts

Potential Impact Number: 24173 **SRP Section Number:** 13.6

Document: NRC POLICY STATEMENT 50 FR 32138
Title: POLICY STATEMENT ON SEVERE REACTOR ACCIDENTS REGARDING FUTURE DESIGNS AND EXISTING PLANTS

Summary:

Under the "Policy for New Plant Applications", "Criteria and Procedural Requirements", the Commission stated their belief that a new design for a nuclear power plant (as well as a proposed custom 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 as reflected in the CP Rule (10 CFR 50.34(f));

b. Demonstration of technical resolution of all applicable Unresolved Safety Issues and the medium- and high-priority Generic Safety Issues, including a special focus on assuring the reliability of decay heat removal systems and the reliability of both AC and DC electrical supply systems;

c. Completion of a Probabilistic Risk Assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety; 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.

In discussing application of these criteria, the Commission stated that it also recognizes the importance of such potential contributors to severe accident risk as human performance and sabotage. The issues of both insider and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized as special considerations in the design and in the operating procedures developed for new plants.

Potential Impact Number: 24176 **SRP Section Number:** 5.3.2

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

APPENDIX II
Potential Impacts

Summary:

The COL applicant should submit plant-specific calculations of RTNDT, stress intensity factors, and pressure-temperature curves.

Potential Impact Number: 24177 **SRP Section Number:** 5.3.2

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The COL applicant should submit plant-specific material fracture toughness data and the resulting pressure-temperature curves for staff review and approval.

Potential Impact Number: 24181 **SRP Section Number:** 5.3.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

ABB-CE 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 (recommended in RG 1.44), "Control of the Use of Sensitized Stainless Steel," has been used by ABB-CE since the mid-1970s for verifying non-sensitization of austenitic stainless steel materials and weldments. ABB-CE 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 ABB-CE'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 tests for ASTM A-262, Practice A or E.

Potential Impact Number: 24184 **SRP Section Number:** 3.8.4

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Potential Impacts

Document: NRC STAFF POSITION (Proposed) March 3, 1994
Title: DYNAMIC LATERAL PRESSURES ON EARTH-RETAINING WALLS AND EMBEDDED WALLS OF NUCLEAR POWER PLANT STRUCTURES

Summary:

The objective of this technical position is to identify the appropriateness of the methodology of calculating the dynamic lateral earth pressures on retaining walls and embedded exterior walls as described in the Bechtel Corporation's Topical Report BC-TOP-4, "Seismic Analysis of Structures and Equipment for Nuclear Power Plants," Rev. 4, (in publication).

Potential Impact Number: 24189 **SRP Section Number:** 3.11

Document: C&S: IEEE 382-72
Title: Standard for Qualification of Actuators for Power Operated Valve Assemblies With Safety-Related Functions for Nuclear Power Plants

Summary:

RG 1.73 is cited in this SRP Section. IEEE 382 1972 is endorsed by RG 1.73. The latest version of this standard is IEEE 382 1985.

Potential Impact Number: 24195 **SRP Section Number:** 6.5.1

Document: C&S: IEEE 279
Title: STANDARD CRITERIA FOR SAFETY SYSTEMS FOR NUCLEAR POWER GENERATING STATIONS

Summary:

RG 1.52 is cited in this SRP Section. IEEE 279 1971 is endorsed by RG 1.52. The latest version of this standard is IEEE 603 1991.

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Potential Impact Number: 24201 **SRP Section Number:** 8.1

Document: C&S: IEEE 484-75
Title: IEEE RECOMMENDED PRACTICE FOR INSTALLATION DESIGN AND INSTALLATION OF LARGE LEAD STORAGE BATTERIES FOR GENERATING STATIONS AND SUBSTATIONS

Summary:

RG 1.128 is cited in this SRP Section. IEEE 484 1975 is endorsed by RG 1.128. The latest version of this standard is IEEE 484 1987.

Potential Impact Number: 24202 **SRP Section Number:** 8.1

Document: C&S: IEEE 450-75
Title: IEEE RECOMMENDED PRACTICE FOR MAINTENANCE, TESTING AND REPLACEMENT OF LARGE LEAD STORAGE BATTERIES FOR GENERATING STATIONS AND SUBSTATIONS

Summary:

RG 1.129 is cited in this SRP Section. IEEE 450 1975 is endorsed by RG 1.129. The latest version of this standard is IEEE 450 1987.

Potential Impact Number: 24206 **SRP Section Number:** 8.1

Document: C&S: IEEE 308-80
Title: IEEE STANDARD CRITERIA FOR CLASS 1E POWER SYSTEMS FOR NUCLEAR POWER GENERATING STATIONS

Summary:

RG 1.9 is cited in this SRP Section. IEEE 308 1980 is endorsed by RG 1.9. The latest version of this standard is IEEE 308 1991.

APPENDIX II
Potential Impacts

Potential Impact Number: 24214 **SRP Section Number:** 15.4.8

Document: RG 1.77 **Revision:** 0
Title: ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS

Summary:

RG 1.77 is cited in this SRP Section. ICRP 2 1959 is endorsed by RG 1.77. The latest version of this standard is ICRP 30 1990.

Potential Impact Number: 24231 **SRP Section Number:** 13.5.2.1

Document: FINAL SER CE80 CH 13
Title: CONDUCT OF OPERATIONS

Summary:

In summary, the staff finds ABB-CE's approach to procedure development as described in CESSAR-DC Section 13.5.1 acceptable since it will be addressed by the COL applicant referencing the System 80+ certified design. The COL applicant will be required to submit site-specific information described in 10 CFR 52.79(b) at the COL stage. COL Action Item 13.5-1 is found to be acceptable.

Potential Impact Number: 24232 **SRP Section Number:** 13.5.2.1

Document: FINAL SER ABWR CH 13
Title: CONDUCT OF OPERATIONS

Summary:

The staff determined that development of detailed procedures and associated training materials is beyond the scope of the ABWR design certification and is the responsibility of the COL applicant.

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Potential Impact Number: 24233 **SRP Section Number:** 13.5.2.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix B, criteria VI

Summary:

Measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality. These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed. Changes to documents shall be reviewed and approved by the same organizations that performed the original review and approval unless the applicant designates another responsible organization.

Potential Impact Number: 24240 **SRP Section Number:** 10.3

Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

The capability of all systems and components necessary to provide core cooling and decay heat removal following a station blackout should be determined, including station battery capacity, condensate storage tank capacity, compressed air capacity, and instrumentation and control requirements.

If a system is required for decay heat removal specifically to meet the station blackout duration, the following criteria should be met:

1. The system 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
2. 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.

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Potential Impact Number: 24241 **SRP Section Number:** 10.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.63(a)(2)

Summary:

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. Utilities are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.

Potential Impact Number: 24242 **SRP Section Number:** 9.5.1

Document: NRC LETTER From Zwolinski, J. To Farrar, D.
Title: TECHNICAL SPECIFICATIONS RELATING TO THE USE OF A MOBILE VOLUME REDUCTION SYSTEM AT DRESDEN STATION

Summary:

This letter documented staff acceptance criteria for the onsite storage and use of large liquid propane systems.

Potential Impact Number: 24243 **SRP Section Number:** 9.5.1

Document: NRC LETTER From Richardson, J. To Neils, G.H.
Title: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT TITLED 'GUIDELINES FOR PERMANENT BWR HYDROGEN WATER CHEMISTRY INSTALLATIONS,' 1987 REVISION

Summary:

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This letter documented staff acceptance criteria for the onsite storage and use of large liquid hydrogen and liquid oxygen systems and hydrogen systems supplying hydrogen to the volume control tank in PWRs and for cooling the main electric generators in PWRs and BWRs.

Potential Impact Number: 24244 **SRP Section Number:** 9.5.1

Document: C&S: EPRI NP-5283-SR-A
Title: GUIDELINES FOR PERMANENT BWR HYDROGEN WATER CHEMISTRY INSTALLATIONS

Summary:

This topical report documented acceptance criteria for the onsite storage and use of large liquid hydrogen and liquid oxygen systems.

Potential Impact Number: 24245 **SRP Section Number:** 9.5.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

The conclusion section of the discussion of Generic Issue 136 documents a staff recommendation to revise the SRP to include the acceptance criteria developed through the staff's evaluation of the Dresden Mobile Volume Reduction System and the staff's review of the BWROG topical report on the BWR hydrogen water chemistry control system.

Potential Impact Number: 24246 **SRP Section Number:** 9.5.1

Document: NRC MEMORANDUM 12/14/92, From Beckjord, E. to Gillespie, F.

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Title: GENERIC LETTER FOR IMPLEMENTATION OF RESOLUTION OF GENERIC SAFETY ISSUE 106, 'PIPING AND THE USE OF HIGHLY COMBUSTIBLE GAS IN VITAL AREAS'

Summary:

RES recommended that the SRP be revised to include the information developed in the resolution of Generic Issue 106 and reference EPRI topical report HP-5283-SR-A.

Potential Impact Number: 24247 **SRP Section Number:** 6.5.2

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

The staff has determined that in its evaluation of the evolutionary designs, the current insights from source-term research as described in draft NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," June 1992, regarding fission-product releases into the containment would be utilized.

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 a TID 14844 source term release." Similar wording appears in subparagraphs (vii), (viii), and (xxvi). ABB-CE has implemented the new source term technology summarized in Draft NUREG-1465, however, not the old TID 14844 source term cited in the regulation.

The NRC staff has encouraged the development and implementation of the new source term technology and, as stated below, concurs with ABB-CE's approach. Based on the staff's review and ABB-CE's commitments in Chapter 15 of CESSAR-DC, the staff concludes that the special circumstances described in 10 CFR 50.12(a)(ii) exist in that the regulation need not be applied in this particular circumstance to achieve the underlying purpose because ABB-CE has proposed acceptable alternatives that accomplish the intent of the regulation. On this basis, the staff concludes that an exemption from the requirements of 10 CFR 50.34(f)(2)(vii), (viii), (xxvi), and (xxviii) is justified.

APPENDIX II
Potential Impacts

Potential Impact Number: 24248 **SRP Section Number:** 6.5.2

Document: FINAL SER CE80 CH 1
Title: INTRODUCTION AND GENERAL DISCUSSION

Summary:

The staff proposed design-specific regulations and justified exemptions from existing regulations in order to complete the regulatory framework for the CE 80+ design. An index to these proposed regulations and exemptions is provided in Section 1.6 of this report.

The staff recognized that if certain regulations were applied to the System 80+ design, the use of those regulations would not serve the underlying purpose, or would not be needed for achieving the fundamental intent, of 10 CFR 52.48.

The Commission approved the staff's recommendation to proceed with design-specific rulemakings as part of individual design certifications to resolve selected technical and severe accident issues as "applicable regulations" for the System 80+ standard design. The proposed design certification rule will then include these "applicable regulations" for the System 80+ design for the purposes of 10 CFR 52.48, 52.54, 52.59, and 52.63.

As identified in Section 1.6, Section 15A of the FSER describes an exemption from "TID 14844" Radiological Source Term , which is codified in 10 CFR 100.

Potential Impact Number: 24249 **SRP Section Number:** 3.9.3

Document: NRC MEMORANDUM 05/05/92, from Robert L. Baer to James A. Norberg Revision Date: 5/5/92
Title: RECOMMENDATIONS FOR SRP REVISIONS RELATED TO SNUBBERS

Summary:

RES proposed SRP changes as part of the resolution of Generic Issue 113 which addressed the dynamic qualification and testing of large bore hydraulic snubbers.

Potential Impact Number: 24250 **SRP Section Number:** 6.5.4

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Potential Impacts

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

Section 15A.3.1 documents a staff review of the transition from predominantly elemental iodine to particulate form. In Regulatory Guide 1.4 the chemical form of iodine is specified predominately elemental iodine (91 percent), with 5 percent assumed to be particulate iodine. However, as documented in this review, "in draft NUREG-1465, 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 (I2) and hydrogen iodide (HI). Once within the containment, highly soluble cesium iodide will readily dissolve in water pools forming iodide (I-) in solution and will deposit onto the interior surfaces. The staff also stated in NUREG-1465 that the radiation-induced conversion of iodide (I-) in water into elemental iodine (I2) is strongly 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."

Potential Impact Number: 24251 **SRP Section Number:** 3.8.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

In Section 3.8.1 of the FSER for the ABWR design certification review, there is a reference to a staff position for shell buckling in Appendix E of the FSER.

Potential Impact Number: 24256 **SRP Section Number:** 9.3.2

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

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Summary:

GE resolved Open item 9.3.2.2-2 from the DFSER by explaining that whenever core uncovering is suspected, the reactor vessel is rapidly depressurized, and thus pressurized reactor water samples are not necessary. GE revised its SSAR to document this explanation and on this basis the DFSER Open Item 9.3.2.2-2 was resolved.

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. The ABWR FSER discusses the modifications which can be summarized as follows: (1) The system will have the capability to sample and analyze for activity in the reactor coolant and the containment atmosphere 24 hours following the accident. (2) Pressurized reactor water samples are not necessary. (3) There is no need for the PASS to analyze dissolved gases. (4) Boron concentration in the reactor coolant will not be required for the first 8 hours following the accident. (5) The PASS will not be required to have the capability to perform chloride analysis.

Potential Impact Number: 24257 **SRP Section Number:** 3.8.3

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

Appendix 3A in Chapter 3 of the System 80+ FSER presents the staff position on steel embedments and provides staff exceptions to Appendix B to ACI 349.

Potential Impact Number: 24258 **SRP Section Number:** 3.8.3

Document: FINAL SER ABWR APPENDIX F
Title: STAFF POSITION ON STEEL EMBEDMENTS

Summary:

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Appendix F of the ABWR FSER presents the staff position on steel embedments and provides staff exceptions to Appendix B to ACI 349.

Potential Impact Number: 24263 **SRP Section Number:** 9.3.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

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.

In its July 21, 1993 Staff Requirements Memorandum, the Commission approved the staff positions to exempt the PASS for ALWRs of PWR design from determining the concentration of hydrogen in the containment atmosphere in accordance with the requirements of 10 CFR 50.34(f)(2)(viii) and Item II.B.3 of NUREG-0737. It also approved extending the time limit for analysis of the coolant for boron and activity to eight hours and 24 hours, respectively. The Commission modified the staff's recommendations regarding evolutionary and passive ALWRs of the PWR type to have the capability to determine the gross amount of dissolved gases (not necessarily pressurized) as a means to meet the intent of 10 CFR 50.34(f)(2)(viii) and Item II.B.3 of NUREG-0737.

In addition, the FSER documents that the PASS will be able to sample and analyze for activity in reactor coolant and the containment atmosphere 24-hours following the accident.

Also, the staff acknowledged that determination of chloride concentrations, although helpful in ensuring that plant personnel take appropriate actions to minimize the likelihood of accelerated primary system corrosion following the accident, is a secondary consideration since long-term samples could likely be taken at a low pressure. It does not constitute, therefore, a mandatory requirement of the PASS. The time for taking these samples can be extended to 24 hours following the accident.

Potential Impact Number: 24264 **SRP Section Number:** 3.8.4

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Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

Appendix 3A in Chapter 3 of the System 80+ FSER presents the staff position on steel embedments and provides staff exceptions to Appendix B to ACI 349.

Potential Impact Number: 24265 **SRP Section Number:** 3.8.4

Document: FINAL SER ABWR APPENDIX F
Title: STAFF POSITION ON STEEL EMBEDMENTS

Summary:

Appendix F of the ABWR FSER presents the staff position on steel embedments and provides staff exceptions to Appendix B to ACI 349.

Potential Impact Number: 24266 **SRP Section Number:** 3.8.4

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

Appendix 3B In Chapter 3 of the System 80+ FSER presents the staff position on the use of standard ANSI/AISC N690-1984 Nuclear Facilities: Steel Safety-Related Structures.

Potential Impact Number: 24267 **SRP Section Number:** 3.8.4

Document: FINAL SER ABWR APPENDIX G

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Potential Impacts

Title: STAFF POSITIONS AND TECHNICAL BASES ON THE USE OF AMERICAN NATIONAL STANDARDS INSTITUTE (ANSI)/AMERICAN INSTITUTE OF STEEL CONSTRUCTION (AISC) N690, 'NUCLEAR FACILITIES - STEEL SAFETY-RELATED STRUCTURES'

Summary:

Appendix G of the ABWR FSER presents the staff position on the use of standard ANSI/AISC N690-1984 Nuclear Facilities: Steel Safety-Related Structures.

Potential Impact Number: 24268 **SRP Section Number:** 3.8.4

Document: FINAL SER ABWR APPENDIX H
Title: DYNAMIC LATERAL SOIL PRESSURES ON EARTH RETAINING WALLS AND EMBEDDED WALLS OF NUCLEAR POWER PLANT STRUCTURES

Summary:

Appendix H of the ABWR FSER presents the staff position on dynamic lateral soil pressures on earth retaining walls and embedded walls of nuclear power plant structures.

Potential Impact Number: 24271 **SRP Section Number:** 9.3.2

Document: NRC GENERIC LETTER 83-37
Title: NUREG-0737 TECHNICAL SPECIFICATIONS

Summary:

Generic Letter 83-37 covers pressurized water reactors and is an identical Generic Letter to Generic Letter 83-36 which covers boiling water reactors. See PI-4396 for details on the impact on the post-accident sampling system.

Potential Impact Number: 24273 **SRP Section Number:** 15.1.1 - 15.1.4

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Potential Impacts

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

TMI Action Plan Items II.E.5.1 and II.E.5.2 deal with design sensitivity of Babcock and Wilcox reactors to feedwater transients, and NRC evaluation of design and procedural improvements.

Potential Impact Number: 24274 **SRP Section Number:** 15.1.1 - 15.1.4

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

The staff's review of acceptability of the ODYNA and REDYA computer codes for ABWR transient analyses is documented in Section 15.1 of the ABWR FSER.

Potential Impact Number: 24275 **SRP Section Number:** 15.1.1 - 15.1.4

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

Tables 15.3 and 15.4 of the System 80+ FSER lists topical reports and letters and NRC approval letters for analyses methods and computer codes used in transient and accident analyses.

Potential Impact Number: 24277 **SRP Section Number:** 9.5.4

Document: NUREG 1430
Title: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS

APPENDIX II
Potential Impacts

Summary:

The Standard Technical Specification (STS) Administrative Control 5.7.2.17 requires that a Diesel Fuel Oil Testing Program be established, and that sampling and testing be performed in accordance with the applicable ASTM standards. Surveillance Requirement 3.8.3.3 requires that periodic testing of fuel oil be performed in accordance with the testing program.

[Note: The STS NUREGs (NUREG-1431, -1432, -1433, -1434) for other designs (i.e., Westinghouse, CE, and GE) contain identical or nearly identical provisions.]

Potential Impact Number: 24278 **SRP Section Number:** 3.5.1.1

Document: NRC NOTICE 91-37
Title: COMPRESSED GAS CYLINDER MISSILE HAZARDS

Summary:

This NRC Notice provides information on compressed gas cylinder missile hazards. This information may be of some use in developing review procedures for 3.5.1.1.

Potential Impact Number: 24279 **SRP Section Number:** 3.5.1.1

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

Once a potential missile is identified, its statistical significance is determined. If the combined probability of missile occurrence, impacting a significant target, and causing significant damage 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 using one or more of the six methods defined in the FSER.

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Potential Impact Number: 24281 **SRP Section Number:** 3.5.1.1

Document: NUREG CR-3551
Title: SAFETY IMPLICATIONS ASSOCIATED WITH IN-PLANT PRESSURIZED GAS STORAGE AND DISTRIBUTION SYSTEMS IN NUCLEAR POWER PLANTS

Summary:

This NUREG documents a study performed by the Oak Ridge National Laboratory to identify safety hazards associated with the storage and handling of compressed gases at nuclear power plants. Recommendations are also made concerning compressed gas cylinder missiles.

Potential Impact Number: 24282 **SRP Section Number:** 3.5.1.1

Document: INSPECTION MANUAL CH 71707
Title: PLANT OPERATIONS

Summary:

The NRC Inspection Manual Chapter 71707 "Plant Operations" details inspection procedures for portable gas cylinders as follows:

Plant procedures should require that portable gas cylinders not be allowed in areas containing safety-related equipment unless: (1) Analysis indicates that portable gas cylinder missiles would not damage safety equipment to the extent that safety functions were compromised. (2) Procedures are developed to protect the cylinders and prevent them from becoming missiles. For portable gas cylinders stored in the plant, the inspector will ensure that at least one of the above conditions is met.

Potential Impact Number: 24285 **SRP Section Number:** 9.5.4

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

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Potential Impacts

The diesel engine vendor has not been selected. Therefore, ABB-CE has not fully defined the auxiliary support systems. Specific design criteria, such as specific tank capacities, pump characteristics, and system flow parameters, were not given in the design description. The selection of these criteria will be included with each application and evaluated on a plant- specific basis.

Potential Impact Number: 24286 **SRP Section Number:** 9.5.4

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The diesel engine vendor has not been selected. Therefore, GE has not fully defined the auxiliary support systems. Specific design criteria, such as specific tank capacities, pump characteristics, and system flow parameters, were not given in the design description. The selection of these criteria will be included with each application and evaluated on a plant- specific basis.

Potential Impact Number: 24287 **SRP Section Number:** 9.5.4

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The ABWR design includes three standby diesel generators. Each diesel engine fuel oil and transfer division is independent and physically separated from the other two divisions, thus a single failure within any one of the divisions will affect only the associated diesel generator.

Potential Impact Number: 24288 **SRP Section Number:** 3.5.1.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

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Summary:

The ABWR discussion on the scope of review in regard to missile protection design features includes those SSC important to safety whose failure could prevent safe shutdown of the facility or result in significant uncontrolled release of radioactivity.

The FSER text reads, "The staff reviewed the ABWR design for protecting SSCs important to safety against internally-generated missiles outside the containment in accordance with SRP Section 3.5.1.1. Specifically, the review included the missile-protection design features for the SSCs whose failure could prevent safe shutdown of the facility or result in significant uncontrolled release of radioactivity." SRP section 3.5.1.1 does not use this terminology throughout its subsections.

Potential Impact Number: 24290 **SRP Section Number:** 3.5.1.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The review documented in the ABWR FSER provides a list of six methods one or more of which is acceptable for providing missile protection for safety-related SSC. This information provides more detailed guidance than is currently documented in SRP section 3.5.1.2.

Potential Impact Number: 24291 **SRP Section Number:** 3.5.1.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

Once a potential missile is identified, its statistical significance is determined. If the combined probability of missile occurrence, impacting a significant target, and causing significant damage 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 using 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.

Potential Impact Number: 24292 **SRP Section Number:** 3.5.1.2

Document: NRC NOTICE 80-21
Title: ANCHORAGE AND SUPPORT OF SAFETY-RELATED ELECTRICAL EQUIPMENT

Summary:

This NRC Notice has a discussion on the control of ancillary equipment control with implications for missile protection. The reviewers observed that non-seismic Category I ancillary items (dollys, gas bottles, block and tackle gear, ductwork, etc.) may be located such that they could potentially dislodge, impact, and damage safety related equipment during an earthquake.

Potential Impact Number: 24293 **SRP Section Number:** 3.5.1.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

As discussed in SSAR Section 3.6.1, redundant divisions of safety-related equipment inside containment are physically separated to the extent practicable to maintain independence in order to prevent loss of safety function due to a single pipe break event. Specifically, redundant divisions of safety-related equipment are widely spaced around containment. A high-energy line separation analysis (HELSA) is used to determine which high-energy lines meet the spatial separation requirements. Although the HELSA evaluation is performed to determine if safety-related systems and components are adequately protected from the effects of pipe breaks, it is also helpful

in determining if adequate protection is provided for these systems and components from missiles which may be generated inside containment.

As part of this evaluation, safety-related systems and components which are more than 9.14 m (30 ft) from any high-energy lines are considered as meeting the spatial separation requirements. Safety systems and components which are less than 9.14 m (30 ft) from high-energy piping are evaluated to see if damage could occur to more than one division. If damage can occur to only one division of safety-related systems, the requirement for separation of redundant equipment is met. If more than one division can be damaged by high-energy piping, barriers, shields, and enclosures will be utilized to protect safety systems and equipment. Based on this information regarding protection of safety systems and components from pipe failures, the staff concludes that this approach is equally effective in protecting these systems and components from missiles generated inside containment.

Potential Impact Number: 24294 **SRP Section Number:** 3.9.5

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

ASME Code, Section III requires that the cumulative damage from fatigue be evaluated for all ASME Code Class 1 piping. The cumulative fatigue usage factor should take into consideration all cyclic effects caused by the plant operating transients for a 60-year design life. Recent test data indicates that the effects of the reactor environment could reduce the fatigue resistance of certain materials. A comparison of the test data with the Code requirements indicates that the margins in the ASME Code fatigue design curves might be less than originally intended. The DFSER reported that the staff was developing an interim position, which would be available at a later date, to account for the environmental effects in the fatigue design of the affected materials. At this time, the staff is assessing the potential generic implication of this issue on all operating plants.

With respect to fatigue considerations for piping design, only the circumferential girth butt welds in piping are considered to be critical by GE and are evaluated for environmental effects. The approach used by GE to account for the environmental effects on the girth butt welds is to modify the local peak stress through (1) the notch factor, (2) the mean stress factor, (3) the environmental correction factor, and (4) the

butt-weld strength reduction factor. SSAR Subsection 3.9.3.1.1.7, "Environmental Effects on Fatigue Evaluation of Carbon Steel Piping," commits to perform additional evaluations for environmental effects on the fatigue design of ASME Code, Section III, Class 1 carbon steel piping in accordance with GE document 408HA414. The SSAR describes the conditions for which these evaluations would be performed and the methodology for performing them. GE's procedure provided supplemental guidelines that enhance the design margin beyond the requirements of the ASME Code, Section III for fatigue evaluation. This is acceptable.

Analyst NOTE: Although the above discussion relates to environmental effects upon the fatigue resistance of certain materials used in piping, and design approaches to ensure conservative fatigue resistance of piping, a similar concern was expressed during reviews of reactor internals in other FSERs applicable to evolutionary plants (e.g. CE System 80+). Although not documented as a concern in this FSER for the ABWR design in conjunction with the staff's review of reactor internals, the environmental parameters and materials of greatest concern with respect to this issue are likely to be more prevalent in BWR designs than in PWR designs.

Potential Impact Number: 24295 **SRP Section Number:** 3.5.1.4

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

Once a potential missile is identified, its statistical significance is determined, if the combined probability of missile occurrence, impacting a significant target, and causing significant damage 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 using one or more of the six methods defined in the FSER.

Potential Impact Number: 24296 **SRP Section Number:** 9.5.5

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

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Summary:

The ABWR design includes three standby diesel generators. Each diesel engine cooling water system is independent and physically separated from the other two divisions, thus a single failure within any one of the divisions will affect only the associated diesel generator.

Potential Impact Number: 24300 **SRP Section Number:** 15.1.5.A

Document: NUREG 1430
Title: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS

Summary:

Technical specification limits on coolant activity limits and primary to secondary coolant leakage limits are used in the radiological consequence analysis for steam line break accidents.

Potential Impact Number: 24301 **SRP Section Number:** 15.1.5.A

Document: NUREG 1432
Title: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS

Summary:

Technical specification limits on coolant activity limits and primary to secondary coolant leakage limits are used in the radiological consequence analysis for steam line break accidents.

Potential Impact Number: 24302 **SRP Section Number:** 15.1.5.A

Document: NUREG 1431
Title: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS

APPENDIX II
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Summary:

Technical specification limits on coolant activity limits and primary to secondary coolant leakage limits are used in the radiological consequence analysis for steam line break accidents.

Potential Impact Number: 24303 **SRP Section Number:** 15.1.5.A

Document: C&S: ICRP 2 1959
Title: PERMISSIBLE DOSE FOR INTERNAL RADIATION

Summary:

The breathing rates and dose conversion factors provided in ICRP 2, dated 1959, are incorporated in Regulatory Guide 1.4. RG 1.4 is used in this SRP section as providing guidance on methodology and assumptions.

Potential Impact Number: 24304 **SRP Section Number:** 6.5.2

Document: NUREG CR-5966
Title: A SIMPLIFIED MODEL OF AEROSOL REMOVAL BY CONTAINMENT SPRAYS

Summary:

NUREG/CR-5966 (microfiche 75407:095) provides a model for analysis of containment spray removal of fission products from containment. The staff used this model in evaluation of the ABB CE System 80+ containment spray system as documented in Chapter 15, Appendix A of the staff's FSER.

Potential Impact Number: 24305 **SRP Section Number:** 15.5.1 - 15.5.2

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

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Summary:

The staff's review of acceptability of the ODYNA and REDYA computer codes for ABWR transient analyses is documented in Section 15.1 of the ABWR FSER.

Potential Impact Number: 24306 **SRP Section Number:** 15.5.1 - 15.5.2

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

Tables 15.3 and 15.4 of the System 80+ FSER lists topical reports and letters and NRC approval letters for analyses methods and computer codes used in transient and accident analyses.

Potential Impact Number: 24308 **SRP Section Number:** 9.5.5

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The SSAR includes a commitment that the operating procedures for the diesel generator will require the loading of the engine up to a minimum of 40-percent of full load (or lower as specified in the manufacturer's recommendation) for 1 hour after up to 8 hours of continuous no-load or light-load operation. Such no-load or light-load conditions would exist for a LOCA with offsite power available. Procedures including this criterion will specifically meet the guidance listed in Item III.7 of SRP Section 9.5.5.

Potential Impact Number: 24309 **SRP Section Number:** 9.5.6

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

APPENDIX II
Potential Impacts

Summary:

Several design parameters have been identified as future requirements to be determined once a DG vendor has been selected. Requirements to be specified include the devices to crank the engine, air start requirements with regard to the duration of the cranking cycle, and the number of engine revolutions per start attempt. These requirements will, in turn, dictate such design parameters as the volume and design pressures of the air receivers (sufficient for five diesel engine start cycles) and compressor size (capacity to recharge the system within 30 minutes). This information should be submitted with the COL application.

Potential Impact Number: 24310 **SRP Section Number:** 9.5.6

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

GE identified several design parameters as future interface requirements to be determined once a diesel generator vendor has been selected. Interface requirements to be specified included the devices to crank the engine, the duration of the cranking cycle, and the number of engine revolutions per start attempt. These interface requirements would dictate design parameters such as the volume and design pressure of the air receivers (sufficient for five start cycles for each receiver) and compressor size (sufficient discharge flow to recharge the system in under 30 minutes). Once established, these interface criteria would provide an adequate basis for selecting component capacities. This was identified as DSER (SECY-91-355) Open Item 89 and incorporation of this information into the SSAR was identified as DFSER Confirmatory Item 9.5.6-2. Upon further evaluation, the staff has determined that this requirement can be accomplished by including a COL action item in the SSAR requiring the COL applicant to submit this information.

Potential Impact Number: 24313 **SRP Section Number:** 9.5.7

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

APPENDIX II
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Summary:

In the ABWR FSER, the staff included a finding stating that the system is designed to provide sufficient lubricating oil to support full-load diesel operation for 7 days.

Potential Impact Number: 24314 **SRP Section Number:** 9.5.5

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

Specific information regarding standpipe capacity, circulation and keep-warm pump characteristics, system volume, flow rates, and heat removal capabilities, plus expected water loss over a 7-day period, are designated as interface issues to be submitted once the DG vendor has been selected and diesel engine data are available. This information must be submitted as part of the application for a combined license. This was designated as DSER COL Action Item 9.5.5-1. ABB-CE addressed this issue in Section 9.5.5.2 of Amendment Q to the CESSAR-DC by stating the COL applicant will submit plant- specific information for the DGECWS. The staff finds this acceptable.

Potential Impact Number: 24315 **SRP Section Number:** 9.5.5

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

In the DSER (SECY-91-355), the staff stated that GE established interface requirements for specific information on the design and capability of the cooling water system and the keep-warm system including the following:

(1) jacket-water circulating pump characteristics (NPSH and motive-power source, i.e., shaft, engine, etc.)

(2) the keep-warm system description (design may or may not include a keep-warm pump)

(3) temperature sensor selection (Amot-type or equivalent)

(4) heat removal capability of system (to be based on maximum permissible diesel engine overload output)

(5) expansion tank size

(6) expected water loss over 7-day period and system volume capacity needed to ensure adequate volume is available to maintain system water level and pump NPSH without refill.

These requirements were identified as DFSER Open Item 9.5.5-1. Upon further evaluation, the staff determined that these requirements do not meet the definition of interface requirements implied in 10 CFR Part 52 and can be accomplished by adding a COL action item in the SSAR requiring the COL applicant referencing the ABWR design to submit this information. GE included this information in the SSAR, which is acceptable.

Potential Impact Number: 24316 **SRP Section Number:** 9.5.7

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

In DSER (SECY-91-355) Open Item 92, the staff requested that GE submit the following specific design criteria: pump flows, operating pressure, temperature differentials, cooling system heat removal capabilities and electric heater characteristics for the DGLS. GE stated that the "lubrication system design criteria will be furnished as an interface criteria after selection of the diesel vendor is finalized." Upon further evaluation, the staff determined that this requirement can be accomplished by identifying a COL action item in the SSAR requiring the COL applicant to submit this information.

Potential Impact Number: 24317 **SRP Section Number:** 9.5.7

Document: FINAL SER CE80 CH 9

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Potential Impacts

Title: AUXILIARY SYSTEMS

Summary:

Specific design criteria, such as pump flows, operating pressure, temperature differentials, cooling system heat removal capability, and electric heater characteristics, are dependent on the selection of a DG vendor. In the DSER, the staff stated that this information must be submitted with the COL application. This was designated as DSER COL Action Item 9.5.7-1. ABB-CE addressed this issue in Section 9.5.7.2 of Amendment Q to the CESSAR-DC by stating the COL applicant will submit specific design criteria for the DGELOS. The staff finds this acceptable.

Potential Impact Number: 24318 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

Unlike earlier BWR designs, the Reactor Core Isolation Cooling (RCIC) system in the advanced BWR is a sub-system of the emergency core cooling system (ECCS).

Potential Impact Number: 24319 **SRP Section Number:** 9.5.7

Document: NUREG 1430
Title: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS

Summary:

The DG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated DG under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. Each engine oil sump contains an inventory capable of supporting a minimum of [7] days of operation. [The onsite storage in addition to the engine oil sump is sufficient to ensure 7 days of continuous operation.] This supply is sufficient to allow the operator to replenish lube oil from outside sources.

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NUREG-1431, 1432, 1433, and 1434 contain the same or similar statements.

Potential Impact Number: 24321 **SRP Section Number:** 9.3.1

Document: NRC GENERIC LETTER 88-14
Title: INSTRUMENT AIR SUPPLY SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

Summary:

The purpose of this generic letter is to request that each licensee/applicant review NUREG-1275, Volume 2, an, perform a design and operations verification of the instrument air system. This verification should include verification that the design of the entire instrument air system including air or other pneumatic accumulators is in accordance with its intended function, including verification by test that air-operated safety-related components will perform as expected in accordance with all design-basis events, including a loss of the normal instrument air system.

Potential Impact Number: 24322 **SRP Section Number:** 9.3.1

Document: RG 1.68.3 **Revision:** 0
Title: PREOPERATIONAL TESTING OF INSTRUMENT AND CONTROL AIR SYSTEMS

Summary:

Tests should be conducted, as appropriate, to demonstrate that plant equipment designated by design to be supplied by the instrument and control air system is not being supplied by other compressed air supplies (such as service air) that may have less restrictive air quality requirements.

Potential Impact Number: 24323 **SRP Section Number:** 9.3.1

Document: NRC GENERIC LETTER 88-14

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Title: INSTRUMENT AIR SUPPLY SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

Summary:

The purpose of this generic letter is to request that each licensee/applicant review NUREG-1275, Volume 2, and, perform a design and operations verification of the instrument air system. This verification should include verification by test that actual instrument air quality is consistent with the manufacturer's recommendations for individual components served.

In addition to the above, each licensee/applicant should provide a discussion of their program for maintaining proper instrument air quality.

Potential Impact Number: 24324 **SRP Section Number:** 9.3.1

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

IE Bulletin No. 80-10 identified an issue concerning the potential contamination of nonradioactive systems which could result in unmonitored, uncontrolled releases of radioactivity to the environment. The ABWR design employs several methods to prevent such an occurrence. The methods include ensuring that no cross-connections exist between nonradioactive and potentially radioactive systems, by providing sampling and monitoring of both radioactive and nonradioactive effluents before discharge, and by separating nonradioactive systems from potentially radioactive systems by barriers, along with radiation monitors to detect leakage across the barrier. These methods are discussed in detail in various SSAR sections and are reviewed in various sections of this report. These methods ensure that all discharges to the environment are monitored and controlled. Based on this information, the staff concludes that the ABWR design has provided sufficient design features and design requirements to ensure that the issues discussed in IE Bulletin No. 80-10 are resolved.

Potential Impact Number: 24325 **SRP Section Number:** 9.3.1

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Document: FINAL SER CE80 CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

The staff has reviewed the System 80+ design regarding the issues identified in NRC Bulletin 80-10, "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release to the Environment."

On the basis of this discussion, the staff considers that CESSAR-DC Sections 9.4, 11.2, 11.3, 11.5, and 12.3 satisfactorily address the concerns raised in NRC Bulletin 80-10 and that the design features in the System 80+ design are 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.

Potential Impact Number: 24326 **SRP Section Number:** 9.3.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.63

Summary:

(a) Requirements.

(1) Each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout as defined in Section 50.2.

(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.

Potential Impact Number: 24328 **SRP Section Number:** 9.3.1

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Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

IE Bulletin No. 80-01 concerns the operability of the ADS pneumatic supply. The bulletin states that the ADS pneumatic supply may not be operable for all possible events because of a combination of misapplication of check valves, a lack of testing of the accumulator system backing up each ADS valve operator, and questions about the continued operability of the pneumatic supply in a seismic event.

Based on the staff's review documented in the ABWR FSER, the staff concludes that the HPIN design is sufficient to ensure that the ADS SRV accumulators will supply adequate nitrogen to the ADS SRVs to ensure proper valve actuation. The system design adequately addresses the issues in IE Bulletin No. 80-01.

Potential Impact Number: 24329 **SRP Section Number:** 9.3.1

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

This section of the CE80+ FSER describes the staff's review of CE's evaluation of Generic Letter 88-14 and the recommendations of NUREG-1275 and documents COL action items that were established to address issues outside the scope of DC.

Potential Impact Number: 24330 **SRP Section Number:** 9.5.8

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

ABB-CE did not specify air flow capacity. As with the other DG auxiliary systems, this design characteristic is dependent on the selection of a DG vendor. In the DSER, the staff stated that this information must be submitted with the COL application. This was

designated as DSER COL Action Item 9.5.8-1. ABB-CE has included a sentence in Section 9.5.8.2.1 of Amendment Q to the CESSAR-DC which addresses the COL applicant specifying the DG air flow capacity. The staff finds this acceptable.

Potential Impact Number: 24331 **SRP Section Number:** 9.5.8

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff designated selection of a combustion air flow capacity sufficient to ensure complete combustion as an interface requirement and as Open Item 95 in the DSER (SECY-91-355). Upon further evaluation, the staff determined that this requirement can be accomplished by including a COL action item in the SSAR requiring the COL applicant to submit this information. The interface was reclassified as DFSER COL Action Item 9.5.8-1. GE included this information in the SSAR, which is acceptable.

Potential Impact Number: 24332 **SRP Section Number:** 3.9.5

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

In accordance with SSAR Table 3.2-1, the core support structures and all other safety-related reactor internals are designed as Safety Class 2 components and to the quality assurance requirements of 10 CFR Part 50, Appendix B. In addition, as discussed in Section 3.9.1, "Special Topics for Mechanical Components," Section 3.9.2.4, "Dynamic System Analysis of Reactor Internals Under Faulted Conditions," and Section 3.9.3.1, "Loading Combinations, Design Transients, and Stress Limits," of this report, the SSAR contains acceptable criteria for the design of all safety-related reactor internals under normal, upset, emergency, and faulted loading conditions. Implementation of these criteria to the design of the reactor internal structures and components provides reasonable assurance that, in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components will not exceed allowable

stresses and deformations under such loading combinations. This provides an acceptable design basis for ensuring that these structures and components will withstand the most adverse loading events that were postulated to occur during their service lifetime without loss of structural integrity or impairment of function.

Potential Impact Number: 24333 **SRP Section Number:** 15.7.4

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

The staff's review of the radiological consequences of a postulated fuel handling accident used the positions given in Regulatory Guide 1.25, as modified by revised source term considerations given in Appendix 15A of the FSER.

Potential Impact Number: 24335 **SRP Section Number:** 5.2.3

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In Chapter 1, paragraph 5.3.1.1.5 of the URD, EPRI states that although the stated controls on grinding on stainless steel may be desirable for use on PWR components, they are not required for avoiding IGSCC. ABB-CE intends to avoid fabrication processes which would severely cold work the surface of austenitic stainless steel components. Grinding is done with resin-bonded or rubber bonded, aluminum oxide or silicon carbide wheels which were not previously used on materials other than austenitic alloys. This will offer reasonable assurance of stainless steel component integrity, and that stainless steel components in contact with reactor coolant are not likely to crack. On this basis, DSER Open Item 5.2.3-11 is resolved.

Potential Impact Number: 24339 **SRP Section Number:** 5.2.3

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Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In response to Q281.46, ABB-CE stated that the CVCS maintains the reactor coolant chemistry within the limits defined in EPRI guidelines (EPRI, "PWR Primary Water Chemistry Guidelines: Revision 2," Report NP-7077, November 1990). In addition, ABB-CE indicated that the reactor coolant chemistry is consistent with EPRI Utility Requirements Document (URD) (EPRI, "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, ALWR Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991). However, the water chemistry stated in CESSAR-DC Section 5.2.3.4.1.2.1 was not consistent with EPRI NP-7077 and EPRI URD.

In a letter of November 24, 1992, ABB-CE identified the chemistry limits in CESSAR-DC Section 5.2.3.4.1.2.1 as applying to cleaning water used on NSSS components fabricated with austenitic stainless steel and, accordingly, would not be consistent with reactor coolant water chemistry limits. These cleaning water chemistry limits provide contamination control during component fabrication, shipment, and storage as required by RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants." Also, to prevent halide-induced intergranular corrosion, which would occur with the presence of significant quantities of oxygen, flushing water is inhibited by additions of hydrazine. The staff finds that the cleaning and flushing water does not have to conform to the chemistry limits of reactor coolant water. In addition, EPRI URD indicates that the guidelines in EPRI NP-7077 should be supplemented as follows for the makeup water storage tank: Conductivity at 25C - 0.2 S/cm; Oxygen - 0.100 ppm (max); Chloride - 0.15 ppm (max); Fluoride - 0.15 ppm (max); and Suspended solids* 1.0 ppm (max). ABB-CE revised the primary water chemistry in CESSAR-DC Table 9.2.3-1 to meet or exceed the guidelines of EPRI URD. On this basis, DSER Open Item 5.2.3-7 is resolved.

*Solids concentration is determined by filtration through a filter having a 0.45-micron pore size.

Potential Impact Number: 24341 **SRP Section Number:** 5.2.3

Document: FINAL SER CE80 CH 5

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Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The following recent fatigue test data indicate that the effects of the environment could significantly reduce the fatigue resistance of materials:

- K. Iida, J. Fukakura, M. Higuchi, H. Kobayashi, S. Miyazono, and M. Nakao, "Survey of Fatigue Strength Data of Nuclear Structural Materials in Japan," Abstract of DBA Committee Report, 1988. (Presented to the American Society of Mechanical Engineers, Subgroup on Fatigue Strength, on December 5, 1988, in New York City.) (Enclosure in letter, from John W. Craig of NRC to Edward Griffing of the Nuclear Management and Resources Council, Washington, DC, July 2, 1991.)

- M. Higuchi and K. Iida, "Fatigue Strength Correction Factors of Carbon and Low Alloy Steels in Oxygen-Containing High-Temperature Water," Nuclear Engineering and Design, Volume 129, 1991, pp. 293-306.

- J. B. Terrell, "Effect of Cyclic Frequency on the Fatigue Life of ASME SA-106-C Piping Steel in PWR Environments," Journal of Materials Engineering, Volume 10, Number 3, 1988, pp. 193-203.

The specific concern relates to the reactor water and temperature environment and its synergistic interactions with strain rate. The recent data indicate that the design fatigue curves in ASME Code Section III Class 1 requirements may not be as conservative as originally intended. The design of System 80+ components will address the potential influence of environmental effects on the fatigue life of materials over the 60-year design life. The issue of environmental effects on fatigue is currently under consideration by a special steering committee for cyclic life and environmental effects in nuclear applications of the pressure vessel research council (PVRC). These activities were initiated in response to requests from the ASME Boiler and Pressure Vessel (B&PV) Code Committee and the Board on Nuclear Codes and Standards (BNCS).

Preliminary recommendations were made to the BNCS in September 1992. The initial findings reported to BNCS were that the current fatigue S/N curves should be appropriate for PWR environments. There was not a complete agreement of the steering committee on this position and the issue is unresolved. ABB-CE will continue to monitor the industry activities on the fatigue curves and fatigue analysis methodology.

System 80+ components will be designed to the ASME B&PV Code requirements.

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ABB-CE addressed the issue of environmental fatigue of Class 1 components in Amendment R to Section 3.9.1 of CESSAR-DC and the staff's evaluation is provided in Section 3.12.5.7 of the FSER. On this basis, DSER Open Item 5.2.3-6 is resolved.

Potential Impact Number: 24343 **SRP Section Number:** 5.2.3

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

ABB-CE took exception to the recommendations in Position C.2 of RG 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels," for controls imposed on preheat temperatures for welding ferritic steels. These controls would 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. ABB-CE, in a letter of November 24, 1992, stated 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 ABB-CE'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. This approach should provide adequate assurance that low-alloy steel weldments will not be developing cracking due to hydrogen. On this basis, DSER Open Item 5.2.3-1 is resolved.

Potential Impact Number: 24344 **SRP Section Number:** 5.2.3

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

The ferrite content limits for austenitic stainless weld metal in the CESSAR-DC were broader than those in industry guidelines (EPRI, "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, ALWR Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991) and staff guidance (USNRC,

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"Technical Report on Materials Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," NUREG-0313, Revision 2, January 1988). In CESSAR-DC Amendment L, ABB-CE revised the CESSAR-DC to be consistent with industry guidelines and staff guidance. The revision should assure an acceptable level of structural integrity for stainless steel welds over the life of the plant. On this basis, DSER Open Item 4.5.1-5 is resolved.

Potential Impact Number: 24345 **SRP Section Number:** 5.2.3

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

The ferrite content limits for austenitic stainless steel castings and weld metal in the System 80+ design are broader than those in industry guidelines Electric Power Research Institute (EPRI), "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, Advanced Light Water Reactor (ALWR) Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991, and staff guidance U.S. Nuclear Regulatory Commission, "Technical Report on Material Selection and Processing Guidelines for (BWR) Coolant Pressure Boundary Piping," NUREG-0313, Rev. 2, January 1988. In Amendment Q to CESSAR-DC, ABB-CE committed to limit the ferrite content of austenitic stainless steel castings to a maximum of 20 percent for components with normal operating temperature exposure above 260 C (500 F). In Amendment L to CESSAR-DC, ABB-CE modified the upper ferrite limit to 15 percent for austenitic stainless steel weld metal. The staff believes that these lower limits at ferrite content 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. On this basis, DSER Open Item 6.1-4 is resolved.

NUREG-0313, which provides guidelines for BWR coolant pressure boundary piping, is also cited in the context of corrosion allowances for the CE System 80+ design. ABB-CE provided acceptable corrosion allowances for a 60-year plant design life and its technical basis. Among the materials of construction for the primary-side components are a variety of nickel alloys, stainless steels, and low alloy steels clad with corrosion resistant material (i.e., nickel alloy or stainless steel). All of these materials are resistant to general corrosion in high temperature aqueous environments, and numerous studies have documented this resistance (EPRI URD and NUREG-0313).

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Potential Impact Number: 24346 **SRP Section Number:** 5.2.3

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

Since hydrogen water chemistry will be used, a hydrogen injection of less than 1 part per million (ppm) in the feedwater will also be used to minimize intergranular stress corrosion cracking (IGSCC) of the reactor internals in the ABWR. To suppress IGSCC, the reactor coolant conductivity will be maintained below 0.3 microsiemen/cm and sufficient hydrogen will be added to the feedwater to reduce the electrochemical potential below -0.23 volt (standard hydrogen electrode). These controls will further ensure that the materials exposed to the reactor coolant will not be subject to IGSCC. This is acceptable.

Potential Impact Number: 24347 **SRP Section Number:** 9.2.6

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The CSS is not a safety-related system since the emergency feedwater system (EFWS) is designated as the safety-grade makeup water source. Accordingly, the CSS is not seismically designed. However, ABB-CE states that leakage from the CSS or failure of the CSTs, or both will not result in unacceptable environmental effects. As noted in ABB-CE's response to RAI Q410.112, the CSTs are located in the yard and are designed in compliance with RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," to preclude the potential for releasing radioactive liquids and flooding of areas surrounding the CST. A seismically designed dike or retention pond prevents runoff in the event of tank overflow/rupture.

Potential Impact Number: 24348 **SRP Section Number:** 9.2.6

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Document: FINAL SER ABWR CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

In the draft safety evaluation report (DSER) (SECY-91-235), the staff stated that the provisions incorporated in the ABWR design to control the release of radioactive materials in liquids resulting from inadvertent tank overflow were consistent with the criteria given in RG 1.143 except for the lack of a local alarm capability for the condensate storage tank (CST). This was identified as Open Item 70 in the DSER (SECY-91-235). GE responded in Amendment 20 to SSAR Section 9.2.9.2, Item (9), by stating: "Instrumentation shall be provided to indicate CST water level in the main control room. High water level will be alarmed both locally and in the main control room." This satisfies the staff concern regarding RG 1.143 requirements for a local high level alarm for CST which will be located in the radwaste building control room as stated in SSAR Section 11.2.1.2.1. Therefore, Open Item 70 is resolved.

Potential Impact Number: 24350 **SRP Section Number:** 9.2.6

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

Normal alignment for removal of decay heat is with the CST. Water for RCIC operation is taken from either the CST or the suppression pool as described in the emergency procedure guidelines of SSAR Appendix 18A. The volume of water in these two sources is sufficient to permit core cooling during SBO for a duration of 8 hours. The switchover from the CST to the suppression pool (or the reverse) is performed using station dc power and is not dependent upon either offsite ac power and systems or onsite emergency power systems. Therefore, the MUWC system complies with the applicable guidance of RG 1.155, "Station Blackout," Revision 0 (August 1988).

Potential Impact Number: 24351 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

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Potential Impacts

Summary:

The emergency core cooling system (ECCS) suction strainers are located in the suppression pool, and their function is to ensure that debris in the suppression pool does not lead to clogging of ECCS pumps, heat exchangers, valves, and spray nozzles. To accomplish this function, debris in the suppression pool will be filtered out on the surface of the suction strainers. An excessive accumulation of debris on the strainer surface could lead to inadequate net positive suction head (NPSH) and failure of the ECCS pumps.

In Amendment 35, GE committed to size the RHR suction strainers 3 times the area that results from RG 1.82 for all breaks. The HPCF and RCIC system suction strainers are sized according to RG 1.82, but with conservatism in the mass of debris assumed to be deposited on the strainers from a design basis aspect. GE also committed to provide a 10 percent margin in the net positive suction head available from the static head of the suppression pool for conservatism. Based on the commitments in Amendment 35, the staff finds this acceptable. This resolved Open Item F6.2.1.9-1.

Potential Impact Number: 24353 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

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. In NUREG-0737, the staff stated that the pipe-break-detection circuitry has resulted in spurious isolation of the HPCI and RCIC systems because of the pressure spike that accompanies the actuation of the systems. This TMI item requires applicants to modify the pipe-break-detection circuitry so that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent system isolation. Spurious trips are avoided because the RCIC has a bypass startup system controlled by valves F037 and F045. Upon receiving RCIC start signals, bypass valve F045 opens to pressurize the line downstream and accelerate the turbine. The bypass line through F045 is small (diameter of 1 in.) and naturally limits the initial flow surge to prevent a differential pressure spike in the upstream pipe. After approximately 5 to 10 seconds, steam supply valve F037 opens to admit full steam flow to the turbine. At this stage, the line downstream is already pressurized. This design feature will reduce the possibility that

a pressure spike would occur during any phase of the normal startup process. In the DFSER, the staff concluded that the ABWR design adequately addresses the requirements of this TMI item. However, the staff stated that the COL applicant should test the RCIC bypass startup system during plant startup.

Potential Impact Number: 24354 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In the ABWR design, system reliability is improved by including a bypass line to the turbine steam inlet valve (F045 dc powered) to provide for a smoother turbine start and reduce the possibility of an overspeed trip.

Potential Impact Number: 24355 **SRP Section Number:** 15.7.5

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

The staff's review of the radiological consequences of a postulated fuel handling accident used the positions given in Regulatory Guide 1.25, as modified by revised source term considerations given in Appendix 15A of the FSER. No calculations were performed for a postulated spent fuel cask drop accident, but it is reasonable to assume that similar assumptions would have been used.

Potential Impact Number: 24356 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

APPENDIX II
Potential Impacts

Summary:

Paragraph (2)(xxvi) of 10 CFR 50.34(f) requires provision of leakage control and detection in the design of systems outside containment that contain (or might contain) TID-14844 source term radioactive materials following an accident. Applicants are required to submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and the public and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency.

Potential Impact Number: 24357 **SRP Section Number:** 13.6

Document: CODE OF FED. REGS 10CFR73
Title: PHYSICAL PROTECTION OF PLANTS AND MATERIALS
Locations: 10 CFR 73.46(b)(4), 10 CFR 73.46(b)(4), 10 CFR 73.46(b)(10), 10 CFR 73.55(c)(7), (8), (9) and (10), 10 CFR 73.1 (a)(1)

Summary:

The following changes to 10 CFR 73 have been made and should be reviewed in relation to existing requirements in the SRP.

The security requirements regarding the training, equipment and qualifications of tactical response team members, armed response persons or guards have been revised and now shall also be trained and qualified in accordance with either paragraphs (b)(10) and (b)(11) or paragraphs (b)(12) of this section.

New sections 10 CFR 73.46(b)(10), (b)(11) and (b)(12) have been added addressing the physical fitness training programs for security personnel.

New sections 10 CFR 73.55(c)(7), (8), (9) and (10) have been added addressing physical barriers and four-wheel drive land vehicle bombs.

These rule changes were published in the Federal Register at 59 FR 38889-38900.

Potential Impact Number: 24358 **SRP Section Number:** 5.4.6

APPENDIX II
Potential Impacts

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Systems that should be leak tested are as follows: Reactor core isolation cooling.

Potential Impact Number: 24361 **SRP Section Number:** 4.4

Document: NRC GENERIC LETTER 94-02
Title: LONG-TERM SOLUTIONS AND UPGRADE OF INTERIM OPERATING RECOMMENDATIONS FOR THERMAL-HYDRAULIC INSTABILITIES IN BOILING WATER REACTORS

Summary:

In June 1991, the BWROG issued NEDO-31960 (Ref. 1) which documented proposed long-term solutions to the stability issue as well as methodologies that have been developed to support the design of these long-term solutions. Supplement 1 to NEDO-31960 (Ref. 2) was issued in March 1992 and contained final methodology details and additional information requested by the NRC. By a July 1993 letter from A. C. Thadani (NRC) to L. A. England (BWROG), the NRC transmitted its safety evaluation report on NEDO-31960 and its Supplement 1 to the BWROG. The SER

describes the results of the staff review of the proposed solution concepts and their associated methodologies.

All licensees of BWRs, except for Big Rock Point, are requested to develop and submit to the NRC a plan for long-term stability corrective actions, including design specifications for any hardware modifications or additions to facilitate manual or automatic protective response needed to ensure that the plant is in compliance with General Design Criteria 10 and 12. An acceptable plan could provide for implementing one of the long-term stability solution options proposed by the BWROG and approved by the NRC.

Potential Impact Number: 24362 **SRP Section Number:** 13.6

Document: RG 5.68 **Revision:** 0
Title: PROTECTION AGAINST MALEVOLENT USE OF VEHICLES AT NUCLEAR POWER PLANTS

Summary:

This regulatory guide has been developed to provide guidance 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). In summary, a vehicle barrier system (VBS) that is capable of preventing forced access of a land vehicle to gain proximity to vital areas should be established at each nuclear power reactor site. The VBS should provide a perimeter around vital areas of the facility such that no location along the perimeter would permit forced entry of a land vehicle. The VBS, regardless of the type of barriers used, should be of a design capable of stopping the forward motion of the design basis land vehicle (DBV). The VBS may be incorporated as part of the protected area perimeter system but should not diminish or remove any requirements established for the protected area.

Reference PI-24357 for related changes to 10 CFR 73.

Potential Impact Number: 24365 **SRP Section Number:** 9.2.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

APPENDIX II
Potential Impacts

Location: Appendix A, General Design Criteria 60

Summary:

Control and releases of radioactive materials to the environment. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental condition can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Potential Impact Number: 24366 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The staff noted that the previously described RCIC system capability, without ac power, of "8 hours" had been changed to "at least 2 hours." The staff raised a concern to GE that changing the capability from 8 hours to 2 hours would result in a measurable increase in the core damage frequency estimate as related to station blackout. To clarify the changed position, GE stated and the staff agreed that an RCIC system capability of up to 8 hours could only be adequately demonstrated during startup tests when plant steam is available after fuel loading. SSAR Section 5.4.6 stated that the RCIC system is designed to perform its function without ac power for at least 2 hours with a capability up to 8 hours. It further stated that the COL applicant will provide analyses for the as-built facility to demonstrate the 8-hour capability.

Potential Impact Number: 24373 **SRP Section Number:** 9.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.34(f)(2)(xxvi)

APPENDIX II
Potential Impacts

Summary:

10 CFR 50.34(f)(2)(xxvi) requires provisions for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal 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.

Potential Impact Number: 24374 **SRP Section Number:** 9.3.2

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

Paragraph (2)(xxvi) of 10 CFR 50.34(f) requires provision of leakage control and detection in the design of systems outside containment that contain (or might contain) TID-14844 source term radioactive materials following an accident. Applicants are required to submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and the public and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency.

Based on its review, the staff found in the advance SER 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 eight listed systems. GE revised and expanded the list of systems in Section 1A.2.34 to include the post-accident sampling systems among others.

In the advance SER, the staff also found that the COL applicant will be required to develop plant procedures that will prescribe the leak testing methods for the above systems and schedule maintenance programs to monitor leakages and reduce detected leakages to lowest practical levels. GE stated that the leak tests will be conducted periodically at each refueling outage. Furthermore, the staff found that GE identified Item 5.5.2.2 in the administrative controls section of the ABWR TS, which

calls for the COL applicant to establish a leakage control program for the subject systems. The TS item further stipulates that the program include preventive maintenance and periodic visual inspection requirements and integrated leak test requirements for each system at refueling cycle intervals or less.

Potential Impact Number: 24375 **SRP Section Number:** 9.3.2

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Potential Impact Number: 24376 **SRP Section Number:** 8.3.1

Document: FINAL SER CE80 CH 8
Title: ELECTRIC POWER SYSTEMS

Summary:

In SECY-91-078, the staff recommended that the Commission approve the staff position that an evolutionary plant design should include at least one offsite circuit to

each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus. In its August 15, 1991, SRM, the Commission approved the staff's position.

In SECY-91-078, the staff recommended that the Commission approve the staff's position that an evolutionary plant design should include an alternate power source to the non-safety-related loads, unless it can be demonstrated that the design margins will result in transients for a loss of non-safety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs. In its August 15, 1991 SRM, the Commission approved the staff's position. The staff, in its safety evaluation report (SER) for the EPRI Evolutionary Utility Requirements Document (URD) clarified the intent of this position: "...an alternate power source be provided to a sufficient string of non-safety loads so that forced circulation could be maintained, and the operator would have available to him the complement of non-safety equipment that would most facilitate his ability to bring the plant to a stable shutdown condition, following a loss of the normal power supply and plant trip."

The staff proposed applicable regulations for electrical distribution, as follows:

(1) at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non- safety buses in such manner that the offsite source can power the safety buses upon failure of any non-safety bus.

(2) an alternate power source that is provided to a sufficient string of non-safety loads so that forced circulation could be maintained, and the operator has available a complement of non-safety equipment that would most facilitate the ability to bring the plant to a stable shutdown condition, following a loss of the normal power supply and plant trip.

The staff has determined that conformance to the applicable GDC and guidelines cited above is a sufficient basis for accepting the electric power systems for the System 80+ design.

Potential Impact Number: 24377 **SRP Section Number:** 8.3.1

Document: FINAL SER ABWR CH 8
Title: ELECTRIC POWER SYSTEMS

APPENDIX II
Potential Impacts

Summary:

In SECY-91-078, the staff recommended that the Commission approve its position that an evolutionary plant design should include an alternate power source to the non-safety loads unless it can be demonstrated that the design margins will result in transients for a loss of non-safety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs. In its August 15, 1991, SRM, the Commission approved the staff's position. The staff in NUREG-1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," Volume 2 (August 1992), clarified the intent of this position. Further, the staff has evaluated the electrical design for the ABWR relative to the provision of an alternate source of power for non-safety loads against the following proposed applicable regulation, which is based on the position included in NUREG-1242: "The electric power system of the standard design must include an alternate power source that is provided to a sufficient string of non- safety loads so that forced circulation could be maintained, and the operator has available the complement of non-safety equipment that would most facilitate the ability to bring the plant to a stable shutdown condition, following a loss of the normal power supply and plant trip."

In SECY-91-078, the staff recommended that the Commission approve this posi- tion for evolutionary plant designs. The evolutionary plant design should include at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus. In its August 15, 1991, SRM, the Commission approved the staff's position.

Therefore, the staff's proposed applicable regulation for offsite power sources is as follows:

The electric power system of standard plant design must include at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus.

Potential Impact Number: 24378 **SRP Section Number:** 8.3.1

Document: NUREG 1242
Title: PROGRAM SUMMARY NRC REVIEW OF EPRI ALWR UTILITY REQUIREMENTS DOCUMENT

APPENDIX II
Potential Impacts

Summary:

This SER, NUREG-1242, Section 11, provides 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 designing for a 60 year life, NRC policy regarding alternate power source for non-safety loads and connection of offsite circuits to safety buses(Secy- 91-078), and station blackout policy requirements (Secy-90-016).

Potential Impact Number: 24380 **SRP Section Number:** 9.3.2

Document: FINAL SER CE80 CH 1
Title: INTRODUCTION AND GENERAL DISCUSSION

Summary:

In accordance with 10 CFR 52.48, the staff used the current regulations in 10 CFR Parts 20, 50, 73, and 100 in reviewing ABB-CE's application for design certification. During this review, the staff recognized that if certain regulations were applied to the System 80+ design, the use of those regulations would not serve the underlying purpose of the rule or would not be needed for achieving the fundamental intent of the rule. These exemptions are listed and included section 9.3.2.2 "Exemption from postaccident sampling."

Potential Impact Number: 24381 **SRP Section Number:** 9.3.2

Document: FINAL SER ABWR CH 1
Title: INTRODUCTION AND GENERAL DISCUSSION

Summary:

In accordance with 10 CFR 52.48, the staff used the applicable regulations in 10 CFR Parts 20, 50, 73, and 100 in performing its review of GE's application for design certification. During this review, the staff identified certain regulations for which application of the regulation to the ABWR design would not serve or was not necessary to achieve the underlying purpose of the rule. These exemptions are listed and included section 9.3.2 "Exemption from post-accident sampling."

APPENDIX II
Potential Impacts

Potential Impact Number: 24394 **SRP Section Number:** 8.3.2

Document: NUREG CR-5414
Title: TECHNICAL FINDINGS FOR PROPOSED INTEGRATED RESOLUTION OF GENERIC ISSUE 128, 'ELECTRIC POWER RELIABILITY'

Summary:

Formalized the d-c system design recommendations contained in USNRC NUREG-0666 and restated them as design guidelines noting that the staff had identified these guidelines as sufficient to resolve the concerns of Generic Issue A-30.

Potential Impact Number: 24395 **SRP Section Number:** 8.3.2

Document: C&S: IEEE 946-85
Title: IEEE RECOMMENDED PRACTICE FOR THE DESIGN OF SAFETY-RELATED DC AUXILIARY POWER SYSTEMS FOR NUCLEAR POWER GENERATING STATIONS

Summary:

IEEE Std 946-1985 supports many of the same recommendations contained in NUREG/CR-5414. This IEEE standard provides guidance and addresses the overall design criteria for a d-c power supply system. If the suggested design criteria were followed, the resultant d-c power system design including operation. indication and controls, and surveillance attributes would agree with the recommendations of past studies for the resolution of GI 128.

Potential Impact Number: 24396 **SRP Section Number:** 8.3.2

Document: FINAL SER EPRI CH 11
Title: EPRI URD-EPD: ELECTRIC POWER SYSTEMS

Summary:

APPENDIX II
Potential Impacts

In the EPRI FSER, the staff stated that plant designers need to 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.

Potential Impact Number: 24397 **SRP Section Number:** 8.3.2

Document: FINAL SER CE80 CH 8
Title: ELECTRIC POWER SYSTEMS

Summary:

The staff asked ABB-CE 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.

Potential Impact Number: 24401 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The HPCF and RCIC systems are located in different rooms of the reactor building for additional protection against common-mode failures. Different energy sources will be used for pump motivation (steam turbine for RCIC pump, electric power for HPCF pumps) and different power systems for control power. This diversity conforms to SRP Section 5.4.6.

Potential Impact Number: 24402 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 7
Title: INSTRUMENTATION AND CONTROLS

APPENDIX II
Potential Impacts

Summary:

The RCIC system is a high-pressure injection system which uses a steam turbine-driven pump. The actuated equipment is a single train system. This system is initiated when either high drywell pressure or low reactor water level (L2) setpoints are met. Each parameter has four sensors which provide input via the EMS to the SSLC. The output is via electrical Division I of the SSLC and the EMS. The RCIC pump turbine is automatically shutdown on turbine overspeed, high turbine exhaust pressure, RCIC auto-isolation signal, low pump suction pressure, reactor water level high (L8), or manual trip if the initiation signal is not present. The RCIC system fails-as-is upon loss of power to the SSLC or loss of input signals. Automatic and manual isolation capability for the RCIC system is provided as part of the leak detection and isolation system. The RCIC system itself is not redundant because the HPCF system provides functionally redundant capability. Some RCIC system valves are Division II components and appropriate separation of these devices and circuits from the Division I equipment is maintained. System tests are accomplished with a division-of-sensors bypass in place, as discussed in Section 7.2 of this report. On-line signal verification is accomplished by the SSLC. System status annunciation and performance indicators are provided in the control room.

Potential Impact Number: 24403 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The system cools the following safety-related equipment areas in the reactor building during accident conditions: the RHR pump rooms, HPCF pump rooms, RCIC pump room, FCS equipment rooms, SGTS equipment rooms, and the CAMS equipment rooms. As stated in Section 9.4.5.1 of this report, the RB secondary equipment HVAC system cools the above areas under normal conditions. Increased and reliable cooling of these rooms will be required during accident conditions when the equipment in the rooms operate. The system is designed to provide a controlled temperature environment in the rooms in accordance with equipment temperature qualification. Separate safety-related fan coil units (FCUs), one for each room serve the 12 rooms that house the redundant components of the five systems mentioned above and the RCIC pump room. The pump room FCUs are automatically initiated when RHR pumps, HPCF pumps, and RCIC turbines are started. The safety-related FCUs in other rooms

are automatically initiated upon isolation of the RB secondary containment HVAC system. The FCS room FCUs are initiated upon receiving a manual FCS start signal. All components of the RB safety-related equipment HVAC system are designed to ESF requirements and are safety-related. The units are open-ended and continuously recirculate cooling air within the space served. Divisional RCW is the cooling medium for the associated cooling coils.

Potential Impact Number: 24404 **SRP Section Number:** 9.5.3

Document: C&S: IESNA HANDBOOK
Title: LIGHTING HANDBOOK REFERENCE AND APPLICATION

Summary:

The Illuminating Engineering Society Lighting Handbook (as related to lighting systems design and illumination levels recommended for industrial facilities) is cited as Acceptance Criteria in SRP Section 9.5.3 and listed as Reference 1. The version/date of this publication is not specified. The latest version of this publication is The Illuminating Engineering Society of North America (IESNA) Lighting Handbook Reference & Application, 8th Edition, published in 1993. Chapter 20, "Industrial Lighting," subsection titled "Electric Generating Stations" of the latest version appears to be the portion of the publication most relevant to this regulatory citation. Recommendations specifically addressing additional unique considerations for nuclear power plants appear to have been added in Chapter 20 in or after 1986. Chapter 20 refers to Chapter 11, "Illuminance Selection," for recommended electric generating station and nuclear power plant area illumination levels. Illumination levels to be maintained for electric generating stations are tabulated for various plant areas and for emergency lighting in Fig. 11-1. The basic lighting design recommendations (including those for emergency lighting) for electric generating stations do not appear to have changed since the last revision date (July 1981) of SRP Section 9.5.3 based upon the endnotes provided at the end of Chapter 20. This publication has not been addressed as a code/standard under the SRP-UDP. Since this citation is limited to lighting system design in accordance with a consensus publication, no standard comparison was performed on the basis of the limited potential safety significance of any differences between the cited and current versions of this publication. The EPRI Evolutionary Plant URD states that illumination will be provided for each area of the plant in accordance with the guidelines of the IES Lighting Handbook, as published by the Illumination Engineering Society at the time the plant is designed. This requirement was apparently

acceptable to the staff based upon the lack of staff exceptions to this requirement in the EPRI Evolutionary Plant FSER.

Potential Impact Number: 24406 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

To protect against an intersystem LOCA (ISLOCA), designers of future ALWR plants should reduce the possibility of a LOCA outside 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 the normal RCS operating pressure.

In SECY-90-016, the staff stated that systems that have not been designed to full RCS pressure must include the following protection measures: 1. the capability for leak testing of the pressure isolation valves, 2. valve position indication that is available in the control room when isolation valve operators are deenergized, and 3. high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of the attached low pressure systems and both isolation valves are not closed.

The design pressure for the low-pressure piping systems that interface with the RCS pressure boundary should be equal to 0.4 times the normal reactor operating pressure of 6,965 kPa (1025 psig), that is, 2,786 kPa (410 psig), and the minimum wall thickness of the low-pressure piping should be no less than that of a standard weight pipe. The staff concluded that this minimum pressure will ensure reasonable protection against burst failure should the low-pressure system be subjected to full pressure.

The staff will continue to require periodic surveillance and leak rate testing of the pressure isolation valves per TS requirements as a part of the ISI program.

Where low-pressure piping interfaces with higher-pressure piping that is connected to the RCS, design pressure values were increased to 2,786 kPa (410 psig). The low-pressure piping boundaries were upgraded to URS pressures and extended to the last closed valve connected to the piping interfacing a low-pressure sink, such as the suppression pool or the condensate storage tank.

APPENDIX II
Potential Impacts

Potential Impact Number: 24414 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In the ABWR the ECCS consists of the following: 1. reactor core isolation cooling (RCIC) system; 2. high-pressure core flooder (HPCF) system; 3. automatic depressurization system (ADS); and 4. low-pressure flooder (LPFL) system.

The initiation logic is diversified by adding a high drywell pressure input as well as maintaining the typical system initiation on reactor pressure vessel (RPV) level 2. The RCIC system is a high-pressure reactor coolant makeup system that will start independently of the ac power supply. The system will provide sufficient water to the reactor vessel to cool the core and to maintain the reactor in a standby condition if the vessel becomes isolated from the main condenser and experiences a loss of feedwater flow. The system also is designed to maintain reactor water inventory, in the event of a loss of normal feedwater flow, while the vessel is depressurized to the point where the residual heat removal (RHR) system can function in the shutdown cooling mode at a reactor pressure of 1,034 kPa (150 psig). The HPCF system will maintain the reactor vessel water level above the top of the active core in the event of a break of a 2.54 cm (1 in.)-diameter pipe or smaller and will provide cooling in the event of large-pipe breaks. Actuation of the HPCF system will not require the depressurization of the reactor vessel. The HPCF system consists of two loops. Each loop includes a single motor-driven centrifugal pump that will take suction from the CST or the primary containment suppression pool. An automatic switching feature is based on indication of low CST level. The HPCF flow rate is dependent on the reactor pressure. The HPCF system is designed to operate from normal offsite auxiliary ac power or from the emergency diesel generators. Each HPCF pump will be powered from a different diesel generator. The system will initiate automatically by either low water level 1.5 or high drywell pressure signals. The system also can be placed in operation manually from the main control room and the remote shutdown panel (loop B only). If the RCIC and HPCF systems do not maintain the reactor water level, the ADS will reduce the reactor pressure so that flow from the RHR system, which will be operating in the LPFL mode, enters the reactor vessel in time to cool the core and limit fuel cladding temperature.

APPENDIX II
Potential Impacts

Potential Impact Number: 24416 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

Confirm Adequacy of Space Cooling for HPCI and RCIC Systems (TMI Item II.K.3(24) Paragraph (1)(ix) of 10 CFR 50.34(f) requires a determination of the adequacy of space cooling for the high pressure coolant injection systems at BWRs. Long-term operation of these systems during a complete loss of offsite power, for up to two hours, may require space cooling to maintain the HPCI pump rooms within allowable temperature limits.

Potential Impact Number: 24424 **SRP Section Number:** 5.4.6

Document: RG 1.82 **Revision:** 1
Title: WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT

Summary:

The containment emergency sump(s) should be designed to provide a reliable, long-term water source for ECCS recirculation pumps. The design of the sumps, and the protective screen assemblies is a critical element in assuring long-term recirculation cooling capability. Therefore, adequate design consideration of: a) sump hydraulic performance, b) evaluation of potential debris generation and associated effects including debris screen blockage, c) RHR and CSS pump performance under postulated post-LOCA conditions is necessary. These design considerations are addressed in Regulatory Guide 1.82.

Potential Impact Number: 24425 **SRP Section Number:** 15.8

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

APPENDIX II
Potential Impacts

Summary:

The ABWR complies with prescriptive design requirements of the ATWS rule and is designed to mitigate the effects of an ATWS event. The ABWR also includes features not explicitly required by 10 CFR 50.62 (e.g. fine motion control rod drives, recirculation runback, feedwater runback, ADS auto inhibit) which further reduce the probability of an ATWS or provide further mitigation capability beyond currently operating BWRs. The staff found the design to be acceptable.

Potential Impact Number: 24426 **SRP Section Number:** 15.8

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

GE submitted the ATWS analysis in SSAR Appendix 15E. GE analyzed the limiting transients identified in NEDO-24222 (proprietary) (a supporting document used for the ATWS rule). GE used the NRC-approved ODYNA and REDYA computer codes for the analysis. GE analyzed cases with the ARI system operational, without the alternate rod insertion (ARI) system but with fine motion control rod drive (FMCRD) system operational, and without the ARI and FMCRD systems but with standby liquid control system (SLCS) operational. GE compared the results with the performance guidelines for fuel integrity (coolable core geometry), containment integrity (45 psi (414 kPa)), suppression pool temperature (207 F (97 C)), primary system pressure (1,500 psig (10446 kPa)) and long-term shutdown cooling and found them acceptable.

The staff performed audit calculations to verify that the ABWR design is satisfactory to mitigate the effects of an ATWS. The study focused on the consequences of manual SLCS actuation and no recirculation pump runback on scram signals other than reactor high pressure and reactor low level. One problem that was identified initially was the potential for fuel damage if only the FMCRD system provides slow scram (motor-driven scram) and there were no other actions to lower power. In this case, the power shifted to the top of the core when the control rods entered from the bottom and, in combination with the delay in shutting down the reactor because of the slowness of the FMCRD system, could lead to higher linear power densities and the potential for fuel damage. If the use of the FMCRD system was always in combination with a reactor internal pump trip or recirculation runback, then overall power would be reduced sufficiently to avoid excessive power densities when the power shifted to the top of the

core. The new design feature of recirculation runback on any scram signals or any ARI/FMCRD run-in signals ensures that there is no potential for any unacceptable power shift to the top of the core.

Another feature included in the initial study was the time available for SLCS actuation. Assuming the failure of ARI and FMCRD, the original design called for manual operation of SLCS; however, if the system was not started within a few minutes, containment integrity would be threatened. GE resolved this issue by changing the ABWR design so that the SLCS is automatically started at 3 minutes into an ATWS event.

The results of calculations to evaluate the thermal hydraulic stability of ABWR under the recirculation runback and feedwater runback conditions associated with ATWS events have been provided for staff review. ABWR design response to ATWS enables avoidance of large oscillations and the staff audit of stability calculations is complete. This is discussed in detail in Section 4.4 of this FSER.

Potential Impact Number: 24428 **SRP Section Number:** 15.8

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

The staff asked ABB-CE to submit an ATWS analysis demonstrating that the System 80+ ATWS response is within the bounds considered by the staff during the deliberations leading to the ATWS rule (10 CFR 50.62). In response, ABB-CE submitted the results of ATWS analyses and subsequently included them in CESSAR-DC Section 19.4.13 for the staff review. These analyses were performed on a best-estimate basis. No credit was taken for reactor trip by the non-Class 1E alternate protection system (APS). ABB-CE's analytical results demonstrated that the maximum peak pressure (resulting from the limiting case, loss of main feedwater without a turbine trip) is less than 2.17 x 104 kPa (3,150 psia), and is within the bounds of ABB-CE's response considered by the staff for the ATWS rule making. The analysis was performed assuming a power level of 3,817 MWt. ABB-CE has subsequently increased the power level by 3 percent. ABB-CE did not, however, reanalyze the ATWS event at the increased power level. ABB-CE's basis for not providing a reanalysis is that the pressurizer safety valve capacity has also been increased by 14 percent. ABB-CE stated that the increase in safety valve capacity will offset the effect

of the power increase. The staff agrees that the increase in safety valve capacity will act to limit the effect of increasing power. Since the System 80+ design complies with the requirements of the ATWS rule specified in 10 CFR 50.62, and the ATWS analysis shows that the maximum peak pressure is comparable to that considered for the ATWS rule making, the staff concludes that the System 80+ design satisfactorily addresses the ATWS concerns and is acceptable.

The original Technical Specifications (TSs) for the System 80+ design allow positive moderator temperature coefficients (MTCs) for operation of System 80+ plants. A positive MTC design is not consistent with the requirements of EPRI's URD, which requires a negative MTC design. Furthermore, ATWS analysis submitted to the staff assumed a negative MTC to calculate the peak pressure to be within the acceptable bound considered for the ATWS rule making. To address the staff's concern, ABB-CE revised TS 3.1.4 to limit MTCs to be nonpositive. Since this approach is consistent with the assumption used in the ATWS analysis, the staff determines that it is acceptable.

Potential Impact Number: 24429 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

SRP Section 3.9.6 provides guidance for review of inservice testing (IST) of certain safety-related pumps and valves typically designated as ASME Code Class 1, 2, or 3. The staff's review acceptance criteria are based on meeting (1) GDC 37 as related to periodic functional testing of the ECCS to assure the leak tight integrity and performance of its active components; (2) GDC 40 as related to periodic functional testing of the containment heat removal system to assure the leak tight integrity and performance of its active components; (3) GDC 43 as related to periodic functional testing of the containment atmospheric cleanup systems to assure the leak tight integrity and the performance of the active components, such as pumps and valves; (4) GDC 46 as related to periodic functional testing of the cooling water system to assure the leak tight integrity and performance of the active components; (5) GDC 54 as related to piping systems penetrating containment being designed with the capability to test periodically the operability of the isolation and determine valve leakage acceptability; and (6) 10 CFR Part 50, Subsection 50.55a(f) as related to including pumps and valves whose function is required for safety in the inservice testing program to verify operational readiness by periodic testing.

APPENDIX II
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Potential Impact Number: 24431 **SRP Section Number:** 5.4.1.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The design overspeed will be at least 10 percent above the highest anticipated overspeed of the pump. The highest anticipated overspeed is predicted on the basis of the largest break size remaining after application of the leak-before-break principle of the piping as described in Section 3.6 of this report. In Amendment T to the CESSAR-DC, ABB-CE identified the largest pipe remaining after leak-before-break as a 4-inch pressure spray line.

Potential Impact Number: 24432 **SRP Section Number:** 5.4.1.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

ABB-CE stated that a typical plate material is A-533, Grade B, Class 1, and a typical forging material is A-508, Class 2 or Class 3. However, the recommended materials for flywheels listed in SRP Section 5.4.1.1 are ASME SA-533-B Class 1, SA-508 Class 2, and SA-516 Grade 65. The SA-508 Class 3 material is not on this recommended list. In a letter of November 24, 1993, ABB-CE justified the use of SA-508 Class 3 material. SA-508 Class 3 is a quenched and tempered vacuum-treated carbon and alloy steel forging material commonly used for pressure vessel construction. The material is a high-strength ductile material produced in different classes from Class 1 to Class 5 by varying the chemical composition and heat treatment temperatures. SA-508 Class 2 and SA-508 Class 3 are essentially the same material. Both materials have the same heat treatment temperatures and mechanical properties. The only differences are slight variations in the carbon content (0.27 max. versus 0.25 max.) and the alloy compositions as shown in the SA-508 material specification. Figure A-4200-1 in ASME Section XI shows the same lower bound fracture toughness curve for SA-508 Class 2 and SA-508 Class 3 materials. ABB-CE considers SA-508 Class 3 to be an acceptable flywheel forging material and requests that it be added to SRP Section 5.4.1.1 to allow

for future use of other acceptable materials. The staff believes this material should provide adequate materials properties to ensure a 60-year life of the reactor coolant pump flywheel.

Potential Impact Number: 24433 **SRP Section Number:** 15.1.1 - 15.1.4

Document: NRC GENERIC LETTER 81-08
Title: ODYN CODE (GENERIC LETTER 81-08)

Summary:

This Generic Letter transmits information on the NRC review, approval and requirements for use of the ODYN code utilized by BWR licensees in the evaluation of accidents and transients to satisfy the necessary reload analyses.

The information provided in this Generic Letter and the NRC staffs SER should be considered for incorporation as applicable reference material.

Potential Impact Number: 24434 **SRP Section Number:** 15.2.1 - 15.2.5

Document: NRC GENERIC LETTER 81-08
Title: ODYN CODE (GENERIC LETTER 81-08)

Summary:

This Generic Letter transmits information on the NRC review, approval and requirements for use of the ODYN code utilized by BWR licensees in the evaluation of accidents and transients to satisfy the necessary reload analyses.

The information provided in this Generic Letter and the NRC staffs SER should be considered for incorporation as applicable reference material.

Potential Impact Number: 24437 **SRP Section Number:** 15.3.3 - 15.3.4

Document: NRC GENERIC LETTER 81-08

APPENDIX II
Potential Impacts

Title: ODYN CODE (GENERIC LETTER 81-08)

Summary:

This Generic Letter transmits information on the NRC review, approval and requirements for use of the ODYN code utilized by BWR licensees in the evaluation of accidents and transients to satisfy the necessary reload analyses.

The information provided in this Generic Letter and the NRC staffs SER should be considered for incorporation as applicable reference material.

Potential Impact Number: 24440 **SRP Section Number:** 15.1.1 - 15.1.4

Document: NRC GENERIC LETTER 81-37
Title: ODYN CODE REANALYSIS REQUIREMENTS

Summary:

This Generic Letter transmits information on the NRC review, approval and requirements for use of the ODYN code utilized by BWR licensees in the evaluation of accidents and transients to satisfy the necessary reload analyses.

The information provided in this Generic Letter and the NRC staffs SER should be considered for incorporation as applicable reference material.

Potential Impact Number: 24441 **SRP Section Number:** 15.2.1 - 15.2.5

Document: NRC GENERIC LETTER 81-37
Title: ODYN CODE REANALYSIS REQUIREMENTS

Summary:

This Generic Letter transmits information on the NRC review, approval and requirements for use of the ODYN code utilized by BWR licensees in the evaluation of accidents and transients to satisfy the necessary reload analyses.

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The information provided in this Generic Letter and the NRC staffs SER should be considered for incorporation as applicable reference material.

Potential Impact Number: 24444 **SRP Section Number:** 15.3.3 - 15.3.4

Document: NRC GENERIC LETTER 81-37
Title: ODYN CODE REANALYSIS REQUIREMENTS

Summary:

This Generic Letter transmits information on the NRC review, approval and requirements for use of the ODYN code utilized by BWR licensees in the evaluation of accidents and transients to satisfy the necessary reload analyses.

The information provided in this Generic Letter and the NRC staffs SER should be considered for incorporation as applicable reference material.

Potential Impact Number: 24449 **SRP Section Number:** 6.2.3

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In its calculation of the short-term and long-term containment pressure- temperature response to postulated high-energy line breaks, GE used the same analytical models and conservative assumptions that were presented and reviewed for the Mark III containment in General Electric Standard Safety Analysis Report (GESSAR) II (NUREG-0979). The staff found these to be acceptable, using independent confirmatory analyses with the CONTEMPT-LT28 computer code. These models and assumptions are discussed in the ABWR SSAR and NEDO-20533 and its Supplement 1, "The G.E. Mark III Pressure Suppression Containment Analytical Model." In response to the staff's request for additional information (RAI), GE stated that the analytical models described in NEDO-20533 are appropriate to calculate the ABWR containment responses to postulated accidents. Although originally written for predicting of Mark III transients, these models, which simulate the transient conditions in the containment, can be adapted for the ABWR containment configuration. These

models simulate the drywell, vent systems, and wetwell (suppression pool and airspace). They are, therefore, adaptable to other containment configurations having the same basic components.

Potential Impact Number: 24452 **SRP Section Number:** 13.2.1

Document: NRC GENERIC LETTER 82-18
Title: Reactor Operator and Senior Reactor Operator Requalification Examinations

Summary:

This generic letter transmitted information on the elements of the NRC administered requalification examinations.

In addition to the information on requalification examinations, in response to SECY 82-232 the Commission removed the requirement for NRC conducted simulator examinations for those plants that do not have plant specific simulators.

Potential Impact Number: 24453 **SRP Section Number:** 13.2.1

Document: NRC GENERIC LETTER 84-14
Title: REPLACEMENT AND REQUALIFICATION TRAINING PROGRAM

Summary:

One of the most important follow-up actions that has been identified as part of the operator licensing program reviews is to ensure that there is an accurate description of each licensee's requalification training program and replacement operator training program. This generic letter requested information needed to ensure that candidates for operator licensing examinations have completed the necessary qualifications and training prior to examination and to ensure that requalification program audits by the Regions are based on the requalification training program as implemented.

The regulations, 10 CFR Part 50.54(i-1), require that licensees have a program in effect which meets the requirements of Appendix A to Part 55 and that changes to the approved program that decrease the scope, time allotted for the program or frequency in conducting different parts be approved.

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Potential Impact Number: 24456 **SRP Section Number:** 14.2

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In Section 5.4.3 of the FSER, the staff applied the positions of Branch Technical Position (BTP) RSB 5-1, "Design Requirements of the Residual Heat Removal System," including position E of the BTP, to the review of the Shutdown Cooling System.

Potential Impact Number: 24457 **SRP Section Number:** 14.2

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In Section 5.4.7 of the FSER, the staff applied Branch Technical Position (BTP) RSB 5-1, "Design Requirements of the Residual Heat Removal System," during review of the RHR system and found that the RHR system meets the BTP.

Potential Impact Number: 24459 **SRP Section Number:** 6.2.1.1.C

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(3)(v)

Summary:

10 CFR 50.34(f)(3)(v) states that containment integrity will be maintained by meeting applicable ASME code requirements 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.

Potential Impact Number: 24462 **SRP Section Number:** 6.2.1.1.A

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(3)(v)

Summary:

10 CFR 50.34(f)(3)(v) states that containment integrity will be maintained by meeting applicable ASME code requirements 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.

Potential Impact Number: 24463 **SRP Section Number:** 6.2.1.1.B

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(3)(v)

Summary:

10 CFR 50.34(f)(3)(v) states that containment integrity will be maintained by meeting applicable ASME code requirements 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.

Potential Impact Number: 24464 **SRP Section Number:** 6.1.2

Document: FINAL SER CE80 CH 6

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Title: ENGINEERED SAFETY FEATURES

Summary:

The quality assurance standards of American National Standards Institute (ANSI) N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities" have been superseded. The requirements of ANSI N101.2 are met by selecting coatings based on ASTM D-3842, "Standard Guide for Selection of The Methods for Coatings for Use in Light Water Nuclear Power Plants," ASTM D-3911 "Standard Test Method for Evaluating Coatings Used in Light Water Nuclear Power Plants at Simulated Design Basis Accident Conditions," and ASTM D-3843 "Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities. These ASTM specifications represent the current industry technology and experience and are intended to replace the ANSI standards referenced in ANSI N101.2.

Potential Impact Number: 24472 **SRP Section Number:** 6.2.1.1.A

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.34(f)(3)(v)(B)(1)

Summary:

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.

Potential Impact Number: 24474 **SRP Section Number:** 17.1

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 52.63

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Summary:

10 CFR 52.63 provides requirements with respect to changes to design certifications.

Potential Impact Number: 24475 **SRP Section Number:** 17.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.55(f)

Summary:

10 CFR 50.55 (f) provides requirements with respect to changes in the quality assurance program for construction permit holders.

Potential Impact Number: 24476 **SRP Section Number:** 5.4.6

Document: NRC BULLETIN 88-04
Title: POTENTIAL SAFETY-RELATED PUMP LOSS

Summary:

The NRC staff believes that these issues may be relevant to all water-cooled reactor designs, regardless of the pump application or the NSSS manufacturer. This is based on the belief that miniflow lines have traditionally been designed for only 5% to 15% of pump design flow. Some pump manufacturers now 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.

3. Evaluate the adequacy of the minimum flow bypass lines for safety-related centrifugal pumps with respect to damage resulting from operation and testing in the minimum flow mode. This evaluation should include consideration of the effects of cumulative operating hours in the minimum flow mode over the lifetime of the plant and during the postulated accident scenario involving the largest time spent in this mode. The evaluation should be based on best current estimates of potential pump damage from operation of the specific pump models involved, derived from pertinent test data

and field experience on pump damage. The evaluation should also include verification from the pump suppliers that current miniflow rates (or any proposed modifications to miniflow systems) are sufficient to ensure that there will be no pump damage from low flow operation. If the test data do not justify the existing capacity of the bypass lines (e.g., if the data do not come from flows comparable to the current capacity) or if the pump supplier does not verify the adequacy of the current miniflow capacity, the licensee should provide a plan to obtain additional test data and/or modify the miniflow capacity as needed.

Potential Impact Number: 24477 **SRP Section Number:** 5.4.6

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

To protect the RCIC pump from overheating, the RCIC system contains a miniflow line that will discharge into the suppression pool when the line to the reactor vessel is isolated. When sufficient flow to the vessel is achieved, a valve in the miniflow line will automatically close, thus directing all flow to the reactor.

Potential Impact Number: 24479 **SRP Section Number:** 6.2.1.1.B

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.34(f)(3)(v)(B)(1)

Summary:

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.

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Potential Impact Number: 24480 **SRP Section Number:** 6.2.1.1.C

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.34(f)(3)(v)(B)(1)

Summary:

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.

Potential Impact Number: 24481 **SRP Section Number:** 17.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: 50.34(f)(3)(ii), 50.34(f)(3)(ii), 50.34(f)(3)(iii)

Summary:

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.

Potential Impact Number: 24483 **SRP Section Number:** 17.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.120

Summary:

10 CFR 50.120 provides NRC requirements on training and qualification of nuclear power plant personnel.

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Potential Impact Number: 24485 **SRP Section Number:** 13.3

Document: NRC GENERIC LETTER 91-14
Title: EMERGENCY TELECOMMUNICATIONS

Summary:

The purpose of this Generic Letter is to alert reactor power plant licensees to the forthcoming NRC effort to implement an upgrade to its emergency telecommunications system to the FTS-2000 system. Some level of licensee effort and cooperation will be required for successful implementation of this program. Licensee support actions necessary to upgrade voice and data (Emergency Response Data System required by an August 13, 1991, amendment to 10 CFR 50.72, aka. nuclear data link) emergency communications networks with NRC are identified. No changes to plant parameters, data, or instrumentation readings currently required to be monitored/transmitted to NRC are identified.

As documented in 6/9/94 memorandum from J. Wermiel to A. Gody the instrumentation and controls branch (HICB) disapproved this potential impact for inclusion in Chapter 7. The impact of this generic letter on the review of the ERDS in SRP Section 13.3 must be evaluated.

Potential Impact Number: 24486 **SRP Section Number:** 13.3

Document: NRC GENERIC LETTER 93-01
Title: EMERGENCY RESPONSE DATA SYSTEM TEST PROGRAM

Summary:

This generic letter recommends a schedule for efficiently accomplishing the presently required quarterly testing of the Emergency Response Data System (ERDS), using available telephone lines. The NRC is also issuing this generic letter to recommend an acceptable testing program to addressees, based on experience of both the industry and the NRC in setting up ERDS. The letter is largely based upon industry and NRC experience with ERDS implementation as described in NUREG-1394 Rev.1.

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Potential Impacts

As documented in 6/9/94 memorandum from J. Wermiel to A. Gody the instrumentation and controls branch (HICB) disapproved this potential impact for inclusion in Chapter 7. The impact of this generic letter on the review of the ERDS in SRP Section 13.3 must be evaluated.

Potential Impact Number: 24488 **SRP Section Number:** 4.4

Document: NRC GENERIC LETTER 92-04 Revision Date: 8/19/92
Title: RESOLUTION OF THE ISSUES RELATED TO REACTOR VESSEL WATER LEVEL INSTRUMENTATION IN BWRs PURSUANT TO 10 CFR 50.54(F)

Summary:

This Generic Letter addresses level errors that could result from the effects of noncondensible gas which may prevent the level instrumentation systems in BWRs from operating properly. This Generic Letter requested action and presented guidance to assure that facilities are in compliance with existing regulatory requirements.

Potential Impact Number: 24489 **SRP Section Number:** 4.4

Document: NRC GENERIC LETTER 84-23
Title: REACTOR VESSEL WATER LEVEL INSTRUMENTATION IN BWRs

Summary:

The letter discusses two required design improvements (and a third still under consideration) for BWR reactor level instrumentation in order to satisfy NUREG- 0737 TMI Action Plan Requirement II.F.2. The improvements discussed are: 1. (Required) Prevention of reference leg overheating or reduction of the vertical level instrument line drops in the drywell (as measured from the condensation pot to the drywell exit point). 2. (Required) Those plants that use mechanical level instrumentation should replace the mechanical level equipment with analog level transmitters and trip units unless operating experience confirms high reliability. 3. (Under consideration) Changes in protection system logic to minimize the need for operator action in response to events involving a reference leg break and a single failure in a channel with an intact reference leg. These changes would involve additional transmitters to satisfy the single failure

criterion. No position requiring item 3 has been issued in subsequent NRC documents reviewed under SRP-UDP.

Potential Impact Number: 24490 **SRP Section Number:** 4.4

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

The staff reviewed the ABWR design for the reactor vessel water level measurement system and found that it meets the requirements specified in GL 84-23, "Reactor Vessel Water Level Instrumentation in BWRs," dated October 26, 1984. To ensure high functional reliability of the instrumentation, the staff issued GL 92-04, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)," dated August 19,1992, and Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs," dated May 28, 1993, requesting hardware modifications for operating reactors. In response to these generic communications, all BWR licensees have committed to implement hardware modifications to their level instrumentation systems. Similar cold-leg instruments are used in the ABWR design.

In the DFSER, the staff stated that the design of the ABWR reactor vessel water level measurement system met the requirements specified in GL 84-23. However, the staff required GE to discuss GL 92-04 in the SSAR and include any design changes necessary to preclude the potential for false reactor coolant level readings. The staff also required GE to determine if compliance with this TMI requirement was affected by any design changes. This was identified in the DFSER as Open Item 20.3-8. In response to these generic communications, GE changed the design of the ABWR RPV level instrumentation system. GE incorporated a backfill modification system that will constantly purge the reference leg with a very low flow rate of water supplied by the CRD system. The constant flow of water up the reference leg will prevent dissolved gases from migrating down the reference leg.

The ABWR FSER also documents an evaluation of the need for instrumentation diversity. The ACRS did not support the staff recommendation on diversity. Based upon the ACRS deliberations and GE's position, the staff reconsidered the need for the requirement for instrumentation diversity.

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Even though it may be desirable to provide instrumentation diversity in the ABWR design, there is not sufficient basis to postulate an unidentified potential common-mode failure. Further, diverse level measurement devices have not been demonstrated to be adequate. In light of the enhanced LOCA response in the ABWR and the guidance provided in the ABWR EPGs to address the use of the RPV instrumentation, the staff concludes that reactor vessel level instrumentation diversity is not required for the ABWR. On the basis of the above discussion, Open Item F20.5.30-1 (DFSER Open Item 20.3-8) is resolved. Based on the above discussion, the staff also concludes that the ABWR RPV level instrumentation will provide adequate indication of ICC as required by TMI action plan item II.F.2.

Potential Impact Number: 24491 **SRP Section Number:** 4.4

Document: NRC GENERIC LETTER 82-28
Title: INADEQUATE CORE COOLING INSTRUMENTATION SYSTEM

Summary:

Provides guidance for Westinghouse and CE PWR Inadequate Core Cooling (ICC) instrumentation systems and requests information to enable staff review of plant specific ICC instrumentation system proposals. The letter indicates that the Commission has determined that an ICC instrumentation system consisting of upgraded subcooling margin monitors, core-exit thermocouples, and a reactor coolant inventory tracking system is required for the operation of pressurized water reactor facilities. The Combustion Engineering Heated Junction Thermocouple (HJTC) system and the Westinghouse Reactor Vessel Level Instrumentation System (RVLIS) are acceptable for tracking reactor coolant system inventory and provide an enhanced ICC instrument package when used in conjunction with core exit thermocouple systems and subcooling margin monitors designed in accordance with NUREG-0737 and operated within approved Emergency Operating Procedure Guidelines. The details of the NRC Staff review of these generic systems are reported in NUREG/CR-2627 and NUREG/CR-2628 for the Combustion Engineering and Westinghouse systems, respectively. Other differential pressure (d/p) measurement techniques for reactor coolant system inventory tracking are acceptable provided that they meet NUREG-0737 design requirements and monitor the coolant inventory over the range from the vessel upper head to the bottom of the hot leg as a minimum.

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Potential Impact Number: 24492 **SRP Section Number:** 6.2.1.1.C

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In its calculation of the short-term and long-term containment pressure- temperature response to postulated high-energy line breaks, GE used the same analytical models and conservative assumptions that were presented and reviewed for the Mark III containment in General Electric Standard Safety Analysis Report (GESSAR) II (NUREG-0979). The staff found these to be acceptable, using independent confirmatory analyses with the CONTEMPT-LT28 computer code. These models and assumptions are discussed in the ABWR SSAR and NEDO-20533 and its Supplement 1, "The G.E. Mark III Pressure Suppression Containment Analytical Model." In response to the staff's request for additional information (RAI), GE stated that the analytical models described in NEDO-20533 are appropriate to calculate the ABWR containment responses to postulated accidents. Although originally written for predicting of Mark III transients, these models, which simulate the transient conditions in the containment, can be adapted for the ABWR containment configuration. These models simulate the drywell, vent systems, and wetwell (suppression pool and airspace). They are, therefore, adaptable to other containment configurations having the same basic components.

Potential Impact Number: 24493 **SRP Section Number:** 6.2.1.1.C

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

Environmental qualification of safety-related mechanical and electrical equipment is described in SSAR Section 3.11. ABWR Class 1E electrical equipment, including pump and valve motors and electrical accessories, will be environmentally qualified by the methods documented in the NRC-approved report, NEDE-24326-1-P, "General Electric Environmental Qualification Program." These methods are in accordance with the guidance of IEEE 323-1974, NUREG-0588, RG 1.89 (Rev. 1), and the generic requirements of 10 CFR 50.49.

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The proposed qualification program for ABWR electrical equipment is acceptable as discussed further in Sections 3.10 and 3.11 of this report. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

Potential Impact Number: 24494 **SRP Section Number:** 6.2.1.1.C

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

GE submitted proprietary SSAR Appendix 3B to address the issue of suppression pool dynamic loads for the ABWR. The scope of Appendix 3B encompasses SRV actuation and LOCA phenomena, SRV discharge loads, LOCA loads, submerged structure loads, and load combinations. Although similar to the Mark III containment design, the ABWR has the following distinctive features that affect suppression pool dynamic loads: wetwell airspace pressurization, an LD volume, a smaller number of horizontal vents (30 in the ABWR compared with 120 in the Mark III), horizontal vent extension into the pool, vent submergence, and suppression pool width.

Potential Impact Number: 24495 **SRP Section Number:** 6.2.1.1.C

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

SRP 3.6.2 contains design, testing, and examination guidelines for guard pipes in the containment penetration areas. SSAR Section 3.6.2.4 states that the ABWR primary containment does not require guard pipes. This is because the ABWR design does not contain guard pipes as defined in Section 3.6.2.4 of RG 1.70, "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants," Revision 3: "a guard pipe is a device to limit pressurization of the space between dual barriers of certain containments to acceptable levels." The staff notes that SRP 3.6.2 uses the term "guard pipe" in a broader context than that in RG 1.70 to include all applicable sleeves in the containment penetration area. Section 3.6.2.1.4.2(6) in the SSAR provides

design, testing, and examination requirements for such sleeves. These requirements are consistent with the guidelines in SRP 3.6.2 and are acceptable.

Potential Impact Number: 24496 **SRP Section Number:** 6.2.1.1.C

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

Of the Mark I, II, and III containments, the ABWR containment design is most similar to the Mark II in size and wetwell-to-drywell volume ratio. Therefore, the staff concludes that the acceptance criteria in SRP Section 6.2.1.1.C for Mark II containments are the most applicable. These acceptance criteria specify (1) an effective steam bypass capability for small breaks (A/K1/2) of 46.5 cm2 (0.05 ft2) (2) automatic initiation of wetwell sprays, (3) preoperational high-pressure leak test, (4) periodic low-pressure leak tests, (5) periodic visual inspection of the vacuum relief system, (6) redundant position indicators and alarms on all vacuum breakers, and (7) monthly operability test of vacuum breakers.

Potential Impact Number: 24497 **SRP Section Number:** 6.2.1.1.C

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

In SSAR Section 15.6.5.5.1, GE assumed that the primary containment leak rate into the reactor building through penetrations and ESF system components will not be greater than an equivalent release of 0.5 percent by weight per day of the primary containment free air weight for the first 24 hours after a LOCA and half of that value (0.25 percent/day) after 24 hours. GE based its assumption of a reduced leak rate on the finding that the primary containment pressure is reduced by more than a factor of 1/2 during the 12 hours following initiation of a LOCA; therefore, the driving force for leakage through the pathway is correspondingly reduced.

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RG 1.3 (Rev. 2) assumes a constant containment leak rate for the duration of a LOCA, although it permits a reduced leak rate with supporting justification. Two rationales support a constant containment leak rate: (1) the pressure profile for a BWR does remain high for a long period of time and (2) for most plants, the leakage is only measured at the maximum value in accordance with Appendix J to 10 CFR Part 50. Therefore, the primary containment leak rate was identified as Open Item 143 in the DSER (SECY-91-355). Subsequently, GE revised the primary containment leak rate of 0.5 percent by weight per day for the entire duration of a LOCA, accepting the staff's position in a draft revision to the SSAR. This was identified as DFSER Confirmatory Item 15.4.4.1-1. In Amendment 24 to the SSAR Section 15.6.5.5.1, GE revised the leak rate to 0.5 percent by weight per day for the duration of a LOCA and this resolved DFSER Confirmatory Item 15.4.4.1-1.

Potential Impact Number: 24498 **SRP Section Number:** 6.2.1.1.C

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In the DSER (SECY-91-355), the staff identified Open Item 4 regarding the adequacy of the vacuum breaker design. Specifically, GE needed to clarify the arrangement of vacuum breakers (e.g., LD or UD, two valves in series for bypass single-failure protection) and the test program that would demonstrate the performance of the vacuum breaker system.

DSER Open Item 4 is resolved because the design of the wetwell-drywell vacuum breakers has been found to be acceptable in accordance with SRP Section 6.2.1, and because the system will be tested as specified in the TS for the ABWR.

Potential Impact Number: 24499 **SRP Section Number:** 6.2.1.1.C

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

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GE documented in its submittal of June 1, 1992, that the same suppression pool temperature limits will be used for the ABWR as those used in current Mark I, II, and III plants to ensure steady condensation of the SRV discharge. These criteria are documented in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments."

Potential Impact Number: 24500 **SRP Section Number:** 6.2.1.1.C

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

The maximum drywell-to-wetwell differential pressure occurs during the blowdown phase (short term) of a LOCA. GE has performed analyses of various postulated primary system breaks, including a double-ended rupture of the main feedwater line, a double-ended rupture of the main steamline, and small-break accidents. The analyses show that the main feedwater line break (FWLB) yields the limiting drywell-to-wetwell differential pressure and peak drywell and wetwell pressure and is, therefore, the design-basis accident for the drywell and wetwell.

Potential Impact Number: 24502 **SRP Section Number:** 4.4

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

ABB-CE did not initially describe the software design of the Core Protection Calculator (CPC)/Control Element Assembly Calculator (CEAC), and to complete its review, the staff requested ABB-CE to (1) describe the CPC/CEAC software design and protection algorithm; (2) identify all the differences from the previously approved designs and evaluate how the differences will affect the CPC/CEAC performance and safety functions, and (3) discuss the verification program for implementing the CPC/CEAC software during the certification process. As an alternative, ABB-CE could have discussed the applicability of the previously submitted and approved documents for the System 80 CPC/CEAC design to the System 80+ design. This was designated as

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DSER Open Item 4.4.4-1. In response, ABB-CE provided a list of design documents, which are included, along with NRC approval documents, in Section 4.4.4 of the FSER. In addition, the FSER states the COL applicant and ABB-CE will follow change control procedures and guidelines listed in the FSER for any changes to the algorithms and data for the CPC and CEAC. The staff found the documentation and change control procedures to be acceptable based on previous approval of the documents.

Potential Impact Number: 24503 **SRP Section Number:** 4.4

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

The overall CPC/CEAC software implementation, by which the system functional requirements will be translated into modules of machine executable code and the modules integrated into a real-time software system, will be verified through the Phase I and Phase II software verification test. The testing will generate a plant-specific data-base document, appropriate test cases, and acceptable criteria and be documented in test reports. Phase I testing will be performed on the DNBR/LPD calculation systems to verify that CPC/CEAC system software modifications have been properly implemented. Phase II testing on the CPC/CEAC system will verify that CPC and CEAC software modifications have been properly integrated with the CPC and CEAC software and system hardware and to confirm that the static and dynamic operation of the integrated system as modified bears out the predictions of the design analyses.

Testing of the CPC/CEAC software for each license applicant referencing the System 80+ design certification will be considered complete with the formal issuance of (1) a CPC/CEAC database document, (2) the Phase I test report, and (3) the Phase II test report. These documents are plant-specific and will be reviewed individually for each license application referencing the System 80+ design certification. Therefore, DSER Open Item 4.4.4-1 is reclassified as COL Action Item 4.4.4-1.

Potential Impact Number: 24504 **SRP Section Number:** 17.2

Document: CODE OF FED. REGS 10CFR50

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Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.120

Summary:

10 CFR 50.120 provides NRC requirements on training and qualification of nuclear power plant personnel.

Potential Impact Number: 24505 **SRP Section Number:** 15.3.3 - 15.3.4

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

ABB-CE topical reports describing analytical methods for transient and accident analyses and the associated NRC approval letters are listed in Table 15.3 of the FSER.

Potential Impact Number: 24506 **SRP Section Number:** 15.3.3 - 15.3.4

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

In conducting its evaluation on the radiological consequences of a locked rotor event, ABB-CE used appropriate guidance from draft NUREG-1465 on the revised source term.

Potential Impact Number: 24507 **SRP Section Number:** 6.1.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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Summary:

Section 5.2.3 of the SER provides a discussion related to reactor coolant chemistry and material selection for compatibility. ABB-CE stated that the Chemical and Volume Control System (CVCS) maintains the reactor coolant chemistry within the limits defined in Electric Power Research Institute (EPRI) guidelines (EPRI, "PWR Primary Water Chemistry Guidelines: Revision 2," Report NP-7077, November 1990). In addition, ABB-CE indicated that the reactor coolant chemistry is consistent with EPRI Utility Requirements Document (URD) (EPRI, "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, ALWR Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991). However, the water chemistry stated in CESSAR-DC Section 5.2.3.4.1.2.1 was not consistent with EPRI NP-7077 and EPRI URD.

ABB-CE subsequently identified the chemistry limits in CESSAR-DC Section 5.2.3.4.1.2.1 as applying to cleaning water used on NSSS components fabricated with austenitic stainless steel and, accordingly, would not be consistent with reactor coolant water chemistry limits. These cleaning water chemistry limits provide contamination control during component fabrication, shipment, and storage as required by RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants." Also, to prevent halide-induced intergranular corrosion, which would occur with the presence of significant quantities of oxygen, flushing water is inhibited by additions of hydrazine. The staff finds that the cleaning and flushing water does not have to conform to the chemistry limits of reactor coolant water.

In addition, EPRI URD supplements the guidelines in EPRI NP-7077 as follows for the makeup water storage tank:

Conductivity at 25C-0.2 S/cm
Oxygen-0.100 ppm (max)
Chloride-0.15 ppm (max)
Fluoride-0.15 ppm (max)
Suspended solids*-1.0 ppm (max)

*Solids concentration is determined by filtration through a filter having a 0.45-micron pore size.

ABB-CE revised the primary water chemistry in CESSAR-DC Table 9.2.3-1 to meet or exceed the guidelines of EPRI URD. On this basis, DSER Open Item 5.2.3-7 is resolved.

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Section 5.4.2 of the SER provides a discussion related to secondary-side chemistry and material selection for compatibility. ABB-CE stated that the secondary water chemistry complies with EPRI Report NP-6239, "PWR Secondary Water Chemistry Guidelines," December 1988, Revision 2. In order to comply with the secondary water chemistry guidelines, ABB-CE has incorporated the following design features and controls:

monitoring the feedwater to limit the amount of impurities that can be introduced into the Steam Generator (SG);

continuous blowdown of the SG to reduce the concentrating effects of the SG;

chemical addition points for establishing and maintaining an environment that minimizes system corrosion;

preoperational cleaning of the feedwater system; and

minimizing feedwater oxygen content before entry into the SG.

Furthermore, ABB-CE has proposed several design and operational features in order to control feed train and SG corrosion.

The staff found the proposed SG materials, water chemistry controls, and related practices to be acceptable.

Potential Impact Number: 24508 **SRP Section Number:** 3.9.4

Document: NRC GENERIC LETTER 86-01
Title: SAFETY CONCERNS ASSOCIATED WITH PIPE BREAKS IN THE BWR SCRAM SYSTEM

Summary:

Generic Letters 81-34 and 81-35 transmitted NUREG- 0803, Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping, dated August 1981 to BWR licensees and applicants. NUREG-0803 provided the staff's guidance and schedule for implementation. The letters also indicated that the issue does not represent a significant safety issue (thus no response required) for Mark III containment plants. Generic Letter 86-01 transmits another staff SER which provides final resolution of the issue. Based on its review of BWROG and General Electric Company (GE) supplied

generic information (NED0-22209, BWROG-8325 and BWROG-8420) and staff generic analyses of the SDV piping system integrity, the staff has concluded that breaks and through wall cracks in the SDV piping need not be postulated. The SER also concludes that water hammer is not considered a contributing factor in potential SDV pipe breaks based upon BWR operating experience.

Potential Impact Number: 24509 **SRP Section Number:** 3.9.4

Document: NRC GENERIC LETTER 81-20
Title: SAFETY CONCERNS ASSOCIATED WITH PIPE BREAKS IN THE BWR SCRAM SYSTEM

Summary:

This letter summarizes an NRC AEOD report which describes a potential sequence of events which could result from a break in the BWR scram discharge piping during a scram condition concurrent with an inability to reclose the scram outlet valves. Concerns identified include the quality of the scram discharge volume piping, the ability to detect and isolate such a break, and potential water and steam degradation of available ECCS equipment as a result of the break. The letter requests considerable information, analyses, etc. from BWR licensees. Estimates of probabilities of failures of scram discharge piping, consequences of such events, evaluation of the AEOD report recommendations, evaluation of ability to isolate such breaks, analyses of ECCS capabilities to mitigate such events (considering accident environmental and flooding effects) in compliance with 10 CFR 50.46, and evaluations of scram discharge volume system conformance to GDC 14, GDC 35, GDC 55, 50.2(v), 50.55a (including footnote 2), and 50.46 of the Commission's regulations are requested.

Potential Impact Number: 24510 **SRP Section Number:** 3.9.4

Document: NRC GENERIC LETTER 81-35
Title: SAFETY CONCERNS ASSOCIATED WITH PIPE BREAKS IN THE BWR SCRAM SYSTEM

Summary:

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The letter transmits NUREG-0803, Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping, dated August 1981 to BWR applicants for CPs and OLs, and holders of CPs. The staff has concluded that the SDV piping system design is acceptable, provided certain conditions are satisfied on a plant specific basis. The staff further concluded that the safety concerns associated with a postulated failure of the SDV piping system do not represent a dominant contribution to the risk of core melt, provided certain assumptions used in the risk assessment are validated on a plant specific basis. NUREG-0803 provides the staff's guidance and schedule for implementation. The letter also indicates that the issue does not represent a significant safety issue (thus no response required) for Mark III containment plants.

Potential Impact Number: 24511 **SRP Section Number:** 3.9.4

Document: NRC GENERIC LETTER 81-33
Title: SAFETY CONCERNS ASSOCIATED WITH PIPE BREAKS IN THE BWR SCRAM SYSTEM

Summary:

This generic letter forwards NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," dated August 1981. Section 5 of NUREG 0803 presents the staff's generic conclusions on this issue.

Potential Impact Number: 24512 **SRP Section Number:** 3.9.4

Document: NRC GENERIC LETTER 81-34
Title: SAFETY CONCERNS ASSOCIATED WITH PIPE BREAKS IN THE BWR SCRAM SYSTEM

Summary:

The letter transmits NUREG-0803, Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping, dated August 1981 to BWR licensees. The staff has concluded that the SDV piping system design is acceptable, provided certain conditions are satisfied on a plant specific basis. The staff further concluded that the safety concerns associated with a postulated failure of the SDV piping system do not represent a dominant contribution to the risk of core melt, provided certain assumptions

used in the risk assessment are validated on a plant specific basis. NUREG-0803 provides the staff's guidance and schedule for implementation.

Potential Impact Number: 24513 **SRP Section Number:** 13.4

Document: NRC GENERIC LETTER 83-28
Title: REQUIRED ACTIONS BASED ON GENERIC IMPLICATIONS OF SALEM ATWS EVENTS

Summary:

Generic Letter 83-28 provides staff positions on the program, procedures and data collection capability to assure that the causes for unscheduled reactor shutdowns are fully understood prior to plant restart.

Potential Impact Number: 24514 **SRP Section Number:** 15.3.3 - 15.3.4

Document: NRC GENERIC LETTER 83-10A
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

Generic Letter 83-10A discusses staff guidance on the resolution of TMI Action Item II.K.3.5.

Potential Impact Number: 24515 **SRP Section Number:** 4.6

Document: NRC BULLETIN 80-14
Title: DEGRADATION OF BWR SCRAM DISCHARGE VOLUME CAPABILITY

Summary:

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This bulletin provides information on two events at BWRs that raised concerns that the scram discharge volume (SDV) function may be degraded by the undetected presence of fluid in the SDV. One event was significant as it indicated the potential for a common cause failure (a faulty solenoid) to result in operation of the SDV in a manner which could defeat both the level switch function and the SDV draining function.

Potential Impact Number: 24516 **SRP Section Number:** 4.6

Document: NRC GENERIC LETTER 80-66
Title: FAILURE OF 76 OF 185 CONTROL RODS TO FULLY INSERT DURING A SCRAM AT A BWR

Summary:

This generic letter transmitted Supplement 1 of NRC IE Bulletin No. 80-17 to BWR licensees for action. The generic letter identifies additional NRC concerns surrounding this event and specifies actions in addition to those specified in NRC IE Bulletin 80-17 that were to be taken by BWR licensees.

Potential Impact Number: 24517 **SRP Section Number:** 4.6

Document: NRC GENERIC LETTER 80-111 Revision Date: 12/18/80
Title: IE SUPPLEMENT 4 TO BULLETIN N. 80017: FAILURE OF CONTROL RODS TO INSERT DURING A SCRAM AT A BWR

Summary:

This generic letter transmitted Supplement 4 of NRC IE Bulletin No. 80-17 to BWR licensees for action. The generic letter identifies additional NRC concerns surrounding this event and specifies actions in addition to those specified in NRC IE Bulletin 80-17 that were to be taken by BWR licensees. The purpose of these actions was to provide assurance that the continuous monitoring system was tested to demonstrate operability as installed, remained operable during plant operation, and was periodically surveillance tested to demonstrate continued operability.

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Potential Impact Number: 24520 **SRP Section Number:** 5.4.2.2

Document: NUREG 0844
Title: NRC INTEGRATED PROGRAM FOR RESOLUTION OF UNRESOLVED SAFETY ISSUES A-3, A-4 AND A-5 REGARDING STEAM GENERATOR TUBE INTEGRITY

Summary:

Report presents results of NRC integrated program for resolution of USIs A-3, A-4, A-5 regarding steam generator tube integrity. A generic risk assessment is provided and indicates that risk from SGTR events is not a significant contributor to total risk. Report indicates staff recommended actions that can further improve the effectiveness of licensee programs. GL 85-02 was issued as part of the integrated program. GL 85-02 encourages licensees to upgrade their programs, to meet intent of staff-recommended actions; however, such actions do not establish NRC requirements. The report evaluates benefits from recommended actions. The staff concludes that with final publication of this report, USIs A-3, A-4, A-5 are technically resolved.

Staff actions (SA) and staff recommended actions (SRA):

SRA Visual inspection of secondary side, QA/QC work procedures
SA Supplemental tube inspections
SRA Full-length tube inspections
SA Denting inspections
SRA Steam generator inservice inspection interval (defined is TS, but 72 months max)
SA Improved eddy-current testing techniques
SRA Secondary water chemistry program
SRA Condenser inservice inspection program
SRA Primary to secondary leakage limits
SRA Coolant iodine activity limit
SA Reactor coolant pressure control
SRA Safety injection signal reset

Potential Impact Number: 24521 **SRP Section Number:** 13.3

Document: NUREG 0654 SUP 1 **Revision:** 1 , 8/1/88
Title: CRITERIA FOR PREPARATION AND EVALUATION OF RADIOLOGICAL EMERGENCY RESPONSE PLANS AND PREPAREDNESS IN SUPPORT OF

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NUCLEAR POWER PLANTS: CRITERIA FOR UTILITY ONSITE PLANNING AND PREPAREDNESS

Summary:

This document was developed for use in reviewing and evaluating utility-prepared offsite emergency planning and preparedness for those situations where State and/local governments decline to participate in emergency planning.

Potential Impact Number: 24523 **SRP Section Number:** 15.3.1 - 15.3.2

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

NRC staff approved analysis methods for transient and accident analyses are discussed in Section 15.1 of the System 80+ FSER.

Potential Impact Number: 24524 **SRP Section Number:** 15.3.1 - 15.3.2

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

NRC staff review of the ODYNA and REDYA computer codes is discussed in Section 15.1 of the ABWR FSER.

Potential Impact Number: 24525 **SRP Section Number:** 15.3.1 - 15.3.2

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

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Summary:

In Section 15.2 of the ABWR FSER, the staff used a special category for a postulated trip of all of the reactor internal pumps with offsite power available with a common-mode failure of adjustable speed drives, and for a pressure regulator down-scale failure. The special category of anticipated transients involving a common-mode software failure used a special acceptance criterion for radiological dose calculations. The FSER stated that the two postulated events will be treated as special cases, applicable only for the ABWR.

Potential Impact Number: 24526 **SRP Section Number:** 15.4.1

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

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. This was designated as DSER Open Item 15.1-5. By letter dated December 18, 1992, ABB-CE agreed to comply with GDC 17 which requires that the LOOP not be treated as a single failure. ABB-CE reported the results in the CESSAR-DC (Amendment R) Section 6.3.3 and Chapter 15. The staff reviewed the submittal and found that ABB-CE considered the LOOP in all the events analyzed and applied the acceptance criteria specified in the related SRP sections for the event with or without LOOP. (CE 15.1)

During the review of the CE System 80+ FSER, the Staff required that a loss of offsite power (LOOP), must be assumed in addition to the limiting single failure event, for the analysis of all anticipated operational occurrence.

Potential Impact Number: 24527 **SRP Section Number:** 15.4.1

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Document: RG 1.70 **Revision:** 3
Title: STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS

Summary:

Guidance in Regulatory Guide 1.70 addresses that "Uncontrolled control rod assembly withdrawal from a subcritical or low power start up condition (assuming the most unfavorable reactivity conditions of the core and reactor coolant system) including control rod or temporary control device removal error during refueling" be considered.

Potential Impact Number: 24534 **SRP Section Number:** 9.5.1

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

If the primary source is a volume-limited supply such as a tank, a minimum of 455,000 liters (120,000 gal.) must be passively reserved for use by the seismically-designed portion of the fire suppression system. This reserve will supply two manual hose reels for 2 hours.

Potential Impact Number: 24535 **SRP Section Number:** 9.5.1

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

If the primary source is a volume-limited supply, such as a tank, a minimum of 450,000 L (120,000 gal) must be passively reserved for use by the seismically designed portion of the suppression system. This reserve will supply two manual hose reels for two hours.

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Potential Impact Number: 24540 **SRP Section Number:** 13.2.1

Document: CODE OF FED. REGS 10CFR55
Title: OPERATORS' LICENSES
Location: 10 CFR 55.45 (b) (1)

Summary:

10 CFR 55.45 (b) (1) establishes a requirement that the operating test for operator and senior operator licenses be administered in a simulation facility.

Potential Impact Number: 24541 **SRP Section Number:** 13.2.2

Document: CODE OF FED. REGS 10CFR26
Title: FITNESS-FOR-DUTY PROGRAM
Location: Section 26.21 and 26.22

Summary:

Requirements for policy communications and awareness training, and training of supervisors and escorts for fitness for duty programs are stated.

Potential Impact Number: 24542 **SRP Section Number:** 6.2.7

Document: NUREG 0800
Title: STANDARD REVIEW PLAN

Summary:

Table 3.2.1-1 of SRP Section 3.2.2 lists ASME Code Subsection NE as the applicable code for metal containment components.

Potential Impact Number: 24545 **SRP Section Number:** 3.7.4

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Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

Design certification, using a single-earthquake SSE design, is predicated on the adequacy of pre-earthquake planning and post-earthquake damage inspections that are to be implemented by the COL applicant. The COL applicant will be required to demonstrate to the NRC staff as a part of its application the procedures it plans to use for pre-earthquake planning and post-earthquake actions. For design certification, the NRC staff reviewed the criteria developed by the EPRI in EPRI Reports 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 for ensuring that these actions can be taken. The staff finds from its review of EPRI NP-5930 that this report is adequate for and may be used by the COL applicant with the following exceptions: [5 specific items are listed, all of which constitute requirements that the seismic instrumentation should meet.]

Potential Impact Number: 24547 **SRP Section Number:** 3.7.4

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

Design certification, using a single-earthquake SSE design, is predicated on the adequacy of pre-earthquake planning and post-earthquake damage inspections that are to be implemented by the COL applicant. The COL applicant will be required to demonstrate to the NRC staff as a part of its application the procedures it plans to use for pre-earthquake planning and post-earthquake actions. For design certification, the NRC staff reviewed the criteria developed by the EPRI in EPRI Reports 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 for ensuring that these actions can be taken. The staff finds from its review of EPRI NP-5930 that this report is adequate for and may be used by the COL applicant with the following exceptions: [5 specific items are listed, all of which constitute requirements that the seismic instrumentation should meet.]

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For the System 80+, the following criteria for structures are used to ensure that when the OBE is eliminated from design, the structures will continue to be designed appropriately for earthquake effects: ... (2) CESSAR-DC Sections 3.7.4 states that seismic instrumentation will be installed so that prompt action can be taken when the response spectra level and the cumulative absolute velocity (CAV) experienced at the site exceed the shutdown level. It also gives information on pre-earthquake planning and post-earthquake operator action. The acceptability review of seismic instrumentation is discussed in Section 3.7.4 of this report.

Potential Impact Number: 24548 **SRP Section Number:** 10.3.6

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

GE has committed in SSAR Section 6.6.7.2 to a monitoring program for erosion-corrosion that provides assurances that procedures or administrative controls are in place to ensure that the NUMARC program (or another equally effective program) is implemented and the structural integrity of all high-energy (two-phase as well as single-phase) carbon-steel systems 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.

Potential Impact Number: 24549 **SRP Section Number:** 10.3.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

NRC (NRR) considered the issue of erosion/corrosion to be resolved based on: (1) guidelines on erosion/corrosion in single-phase piping, as developed by NUMARC and found acceptable by the staff; (2) participation in a timely way by all 113 LWR plants; (3) acceptable analytical procedures for the evaluation and selection of piping to be inspected; (4) replacement of components as needed; and (5) a long-term as well as a

short-term program for continuing evaluation and inspection of both single-phase and two-phase piping.

Potential Impact Number: 24550 **SRP Section Number:** 3.7.4

Document: DRAFT RG DG-1016
Title: NUCLEAR POWER PLANT INSTRUMENTATION FOR EARTHQUAKES

Summary:

NOTE: Draft Regulatory Guide DG-1016 is the second proposed revision 2 to Regulatory Guide 1.12. As stated in the proposed rule at 59 FR 52255, the staff is developing the third proposed revision 2 to Regulatory Guide 1.12 as Draft Regulatory Guide DG-1033.

Paragraph IV(a)(4) of Proposed Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50 would require that suitable instrumentation be provided so that the seismic response of nuclear power plant features important to safety can be evaluated promptly. Paragraph IV(a)(3) of Proposed Appendix S to 10 CFR Part 50 would require shutdown of the nuclear power plant if vibratory ground motion exceeding that of the operating basis earthquake ground motion (OBE) occurs.

This guide is being developed to describe seismic instrumentation acceptable to the NRC staff for satisfying the requirements of Parts 20 and 50 and the Proposed Appendix S to Part 50.

When an earthquake occurs, it is important to immediately assess the effects of the earthquake at the nuclear power plant. State-of-the-art solid-state digital time-history accelerographs installed at appropriate locations will provide time-history data on the seismic response of the free-field, containment structure, and other Category I structures. The instrumentation should be located so that a comparison and evaluation of such response may be made with the design basis and so that occupational radiation exposures are maintained as low as reasonably achievable (ALARA).

The guidance being developed for pre-earthquake planning and post-earthquake actions is based on the assumption that the nuclear power plant has operable seismic instrumentation, including the equipment and software required to process the data within 4 hours after an earthquake. This is necessary because the decision to shut down the plant will be made, in part, by comparing the recorded data against OBE

exceedence criteria. The decision to shut down the plant is also based on the results of the operator walkdown inspections that take place within 8 hours of the event.

Instrumentation characteristics, installation, activation, remote indication, and maintenance are described in this guide to help ensure (1) that the data provided are comparable with the data used in the design of the nuclear power plant, (2) that exceedence of the OBE can be determined, and (3) that the equipment will perform as required.

Potential Impact Number: 24557 **SRP Section Number:** 10.3.6

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In Chapter 1, paragraph 5.3.1.1.5 of the URD, EPRI states that although the stated controls on grinding on stainless steel may be desirable for use on PWR components, they are not required for avoiding IGSCC. ABB-CE intends to avoid fabrication processes which would severely cold work the surface of austenitic stainless steel components. Grinding is done with resin-bonded or rubber bonded, aluminum oxide or silicon carbide wheels which were not previously used on materials other than austenitic alloys. This will offer reasonable assurance of stainless steel component integrity, and that stainless steel components in contact with reactor coolant are not likely to crack. On this basis, DSER Open Item 5.2.3-11 is resolved.

Potential Impact Number: 24559 **SRP Section Number:** 10.3.6

Document: NUREG 0313 **Revision:** 2 , 1/1/88
Title: TECHNICAL REPORT ON MATERIAL SELECTION AND PROCESSING GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING. FINAL REPORT

Summary:

This report updates and supersedes the technical recommendations of NUREG-0313, Technical Report on Material Selection and Processing Guidelines for BWR Coolant

Pressure Boundary Piping, published in July 1977, and its subsequent revision published in July 1980. This report provides the technical bases for the NRC staff's revised recommended methods to control the intergranular stress corrosion cracking susceptibility of BWR piping. For piping that does not fully comply with the material selection, testing, and processing guideline combinations of this document, varying degrees of augmented inservice inspection are recommended. This revision also includes guidance and NRC staff recommendations (not requirements) regarding crack evaluation and weld overlay repair methods for long-term operation or for continuing interim operation of plants until a more permanent solution is implemented.

Analyst Notes:

Reg. Guide 1.31 position C.4 specifies ferrite content limits between 5 and 20% (Ferrite Numbers from 5 to 20) for stainless steel weld metal filler materials. NUREG-0313, Revision 2 identifies a lower ferrite content limit of 7.5% which is more restrictive than the lower limit specified in Reg. Guide 1.31.

NUREG-0313, Revision 2 guidelines for prevention of IGSCC in BWR RCPB piping are in many cases different (including stated carbon and ferrite content limits) than the current guidance of Reg. Guide 1.44 for control of the use of sensitized stainless steel. NUREG-0313, Revision 2 guidelines are in some cases more restrictive and/or more comprehensive (provide more initial material selection options and more post weld treatment/repair options to reduce susceptibility to IGSCC) than Reg. Guide 1.44 guidance.

Microfiche No: 44241:127

Potential Impact Number: 24560 **SRP Section Number:** 10.3.6

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

The ferrite content limits for austenitic stainless weld metal in the CESSAR-DC were broader than those in industry guidelines (EPRI, "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, ALWR Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991) and staff guidance (USNRC, "Technical Report on Materials Selection and Processing Guidelines for BWR Coolant

Pressure Boundary Piping," NUREG-0313, Revision 2, January 1988). In CESSAR-DC Amendment L, ABB-CE revised the CESSAR-DC to be consistent with industry guidelines and staff guidance. The revision should assure an acceptable level of structural integrity for stainless steel welds over the life of the plant. On this basis, DSER Open Item 4.5.1-5 is resolved.

Potential Impact Number: 24561 **SRP Section Number:** 10.3.6

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

The ferrite content limits for austenitic stainless steel castings and weld metal in the System 80+ design are broader than those in industry guidelines Electric Power Research Institute (EPRI), "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, Advanced Light Water Reactor (ALWR) Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991, and staff guidance U.S. Nuclear Regulatory Commission, "Technical Report on Material Selection and Processing Guidelines for (BWR) Coolant Pressure Boundary Piping," NUREG-0313, Rev. 2, January 1988. In Amendment Q to CESSAR-DC, ABB-CE committed to limit the ferrite content of austenitic stainless steel castings to a maximum of 20 percent for components with normal operating temperature exposure above 260 C (500 F). In Amendment L to CESSAR-DC, ABB-CE modified the upper ferrite limit to 15 percent for austenitic stainless steel weld metal. The staff believes that these lower limits at ferrite content 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. On this basis, DSER Open Item 6.1-4 is resolved.

NUREG-0313, which provides guidelines for BWR coolant pressure boundary piping, is also cited in the context of corrosion allowances for the CE System 80+ design. ABB-CE provided acceptable corrosion allowances for a 60-year plant design life and its technical basis. Among the materials of construction for the primary-side components are a variety of nickel alloys, stainless steels, and low alloy steels clad with corrosion resistant material (i.e., nickel alloy or stainless steel). All of these materials are resistant to general corrosion in high temperature aqueous environments, and numerous studies have documented this resistance (EPRI URD and NUREG-0313)

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Potential Impact Number: 24569 **SRP Section Number:** 9.5.3

Document: FINAL SER ABWR CH 8
Title: ELECTRIC POWER SYSTEMS

Summary:

GE identified an exception to the requirement that all Class 1E equipment is seismically qualified. GE indicated that the safety-related dc standby lighting system is powered from the Class 1E dc system and that the lighting system circuits from the Class 1E dc system power source to the lighting fixtures will be treated as Class 1E circuits (that is, these circuits will be classified as associated) and will be routed in seismic Category I raceways. The lighting fixtures themselves will not be seismically qualified, but will be seismically supported. The bulbs cannot be seismically qualified. To justify this exception, GE indicated that bulbs can only fail open, and, therefore, do not represent a hazard to the Class 1E power source. Based on subsequent discussions with GE, the staff determined that lighting fixtures will be seismically qualified. Light bulbs may fail during and/or following a seismic event thereby extinguishing the light; however, the light bulbs will be replaceable. In addition, bulbs will not fail in a manner that could cause failure of other safety-related systems and will not become a hazard to personnel or safety-related equipment during or following a seismic event.

The staff concludes that lighting circuits that are treated as Class 1E except for the seismic qualification of light bulbs and which include seismically qualified light bulb fixtures will provide protection to Class 1E systems such that there will be reasonable assurance that the lighting systems will not prevent the Class 1E systems from performing their required function.

GE further indicated that overcurrent protective devices and their coordination would provide protection and isolation of the Class 1E power supply from possible failure of non-seismically qualified fixtures during a seismic event. The staff concluded that Class 1E systems will be adequately protected from failure of the non-seismically qualified fixtures as well as the non-seismically qualified light bulbs during a seismic event. The staff concludes that lighting circuits that are treated as Class 1E except for seismic qualification of light bulbs and fixtures and include protective devices which meet safety system requirements will provide protection to Class 1E systems such that there will be reasonable assurance that the lighting systems will not prevent the Class 1E systems from performing their required function.

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Potential Impact Number: 24570 **SRP Section Number:** 15.8

Document: SECY 83-293
Title: AMENDMENTS TO 10 CFR 50 RELATED TO ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) EVENTS

Summary:

During the ATWS rulemaking process, the NRC established a task force and a steering committee on ATWS. The task force presented its recommendations to the Commission in SECY 83-293. The Commission approved a final rule (implementing most of the task force recommendations) on ATWS as 10 CFR 50.62.

The task force recommendations are presented in Enclosure D of SECY 83-293. The prescriptive requirements of the ATWS rule were determined primarily through value/impact evaluations of estimated risk reduction benefits associated with several ATWS-related modification alternatives. The task force separately evaluated GE BWRs and Westinghouse PWRs. CE and B&W plants were evaluated together as a third class of plants due to ATWS behavior similarities. The frequency of ATWS events which result in unacceptable plant conditions was estimated for base cases (plants configured in accordance with pre-ATWS rule requirements) and for evaluation cases (plants configured in accordance with alternative ATWS rule requirements) to determine incremental ATWS risk reduction effects of each alternative. The base cases assumed implementation of several "pre-ATWS rule commitments" which included BWR recirculation pump trips, BWR scram discharge volume upgrades, BWR emergency procedure guidelines applicable to ATWS, PWR scram breaker trip circuit modifications related to the Salem ATWS events, improved PWR procedures/practices based upon staff positions related to the Salem ATWS events, and upgraded auxiliary feedwater systems meeting the requirements of NUREG-0737. Frequencies of transients resulting in reactor protection system (RPS) demands were estimated based upon operating experience data. The task force assumed that the frequency of a failure to scram (for all classes of plants evaluated) was 3 E-5/demand. Since ATWS event sequences are dependent upon the RPS failure mode, the task force further assumed that 2/3 of failures were electrical and 1/3 were mechanical. The task force modeled ATWS event sequences using simplified event trees which reflect probabilistic treatment of mitigating operator actions, mitigating system effectiveness, and mitigating system reliability.

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The unacceptable plant conditions defined by the task force were: BWRs-exceeding the 200 degree F suppression pool temperature limit and PWRs-exceeding ASME Service Level C (approximately 3200 psi) pressure in the reactor coolant system. Events resulting in unacceptable plant conditions were assumed to progress to core melt, containment failure, and large radioactive releases. As cost bases for value/impact evaluations, the task force used cost estimates furnished by a Utility Group for each modification alternative considered. Release consequences were valued at $10,000,000,000 (1 E+7 man-rem maximum estimated ATWS consequences valued at $1,000/man-rem). For plants configured in accordance with the task force's ATWS rule recommendations, the highest frequency of an ATWS event resulting in unacceptable plant conditions was calculated (for CE and B&W plants) to be 2.2 E-5/yr. The task force set a goal (which could not be practically treated as a requirement) for estimated core melt frequency due to ATWS events at about 1 E-5/yr.

Potential Impact Number: 24576 **SRP Section Number:** 5.4.12

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

Issue II.B.1, in NUREG-0933, addressed the requirements in 10 CFR Part 50 and NUREG-0737 to install reactor vessel and RCS high-point vents. These vents are designed to release non-condensable gases from the RCS to avoid loss of core cooling during natural circulation. The design of these vents must conform to the applicable GDC requirements of 10 CFR Part 50 (Appendix A), which are listed below, and meet the applicable codes and standards for the RCS pressure boundary.

ABB-CE states in CESSAR-DC Section 20.2.101 that the System 80+ design includes a SDS that performs the RCGV function to meet the requirements in NUREG-0737. The RCGV system is described in CESSAR-DC Section 6.7.1.2.2. The staff has reviewed the RCGV system design and concludes, in Section 6.7.1 of this report, 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).

Therefore, Issue II.B.1 is resolved for the System 80+ design.

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Potential Impact Number: 24577 **SRP Section Number:** 5.4.12

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

The RCGVS can also remove noncondensable gases or steam from the reactor vessel upper head through a vent line to the RDT through a flow-restricting orifice and one or both of the parallel isolation valves. The RCGVS isolation valves are closed during normal reactor operation. During shutdown or transient conditions, the operator will follow operating procedures to manually actuate (open) the RCGVS valves from the MCR in order to vent the reactor vessel upper head or the pressurizer steam space if the operator determines that noncondensable gases have collected in the reactor vessel upper head or in the pressurizer steam space. The RCGVS will have the capability to be manually actuated, monitored, and controlled from the control room, as required by GDC 19.

The parallel isolation valves are powered by a normal ac power source and an emergency ac power source. In CESSAR-DC Table 6.7-3, ABB-CE documents a failure modes and effects analysis (FMEA) demonstrating that the RCGVS will maintain a vent path after a single failure of either of the valves or of the power source. This demonstration complies with the requirements of GDC 17 and 34.

To satisfy the requirements for natural circulation cooldown specified in SRP, BTP RSB 5-1, the RCGVS can remove the steam bubble and depressurize the RCS for the shutdown cooling mode of operation. This is discussed further in Section 5.4.3-1 of this report. The RCGVS is designed to seismic Category I requirements in compliance with RG 1.29.

The RCGVS is designed to permit periodic inspection in accordance with Section XI of the ASME Boiler and Pressure Vessel Codes. This constitutes compliance with GDC 36.

Potential Impact Number: 24578 **SRP Section Number:** 3.6.2

Document: FINAL SER CE80 CH 3

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Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

In earlier amendments to CESSAR-DC, ABB-CE specified the pipe-break criteria for outside containment penetration areas, which conforms to SRP Section 3.6.2, Revision 1. In response to the preceding staff requests, ABB-CE adopted the staff position for postulation of pipe breaks for ASME Class 1 piping as delineated in SRP Section 3.6.2, BTP MEB 3-1 Revision 2. However, ABB-CE later reinstated the pipe break criteria for Class 1 piping outside the containment penetration areas as delineated in SRP Section 3.6.2, Revision 1. The staff's review of this reinstated position is as follows.

Before issuance of Revision 2 of BTP MEB 3-1 in June 1987, breaks were postulated at intermediate locations between terminal ends of a pipe run if the maximum stress range as calculated by Eq. (10) is greater than 2.4 Sm and if the maximum stress range as calculated by either Eqs. (12) or (13) is greater than 2.4 Sm, where Eqs. (10), (12), and (13) and Sm are as defined in ASME Section III, Subsection NB 3653. In Revision 2 of BTP MEB 3-1, the same criteria are maintained for break exclusion in the containment penetration areas. However, for other areas, the criteria were revised to require that breaks be postulated at any intermediate locations when only Eq. (10) predicts stresses of greater than 2.4 Sm. The use of Eqs. (12) and (13) was eliminated. This resulted in an inconsistency in the Revision 2 criteria in that they allow higher limits in the containment penetration areas than in other areas. The break exclusion areas should provide a margin greater than (or at least equal to) the margin for areas outside the break exclusion area. To determine the impact of this inconsistency, the staff audited several calculations for both boiling-water reactors (BWRs) and PWRs which compared the number of postulated pipe breaks resulting from the use of Revision 1 and 2 criteria. These analyses indicated that the Revision 2 criteria result in a significant increase in the number of postulated breaks, which may be counter-productive in terms of enhancing plant safety. Therefore, the staff has recommended that SRP Section 3.6.2 be revised to reinstate that portion of the Revision 1 criteria related to allowing the use of Eqs. (12) and (13) for the postulation of intermediate pipe breaks in ASME Class 1 piping system. On the basis of the above evaluation, the staff concludes that it is acceptable to reinstate the Revision 1 criteria related to allowing the use of Eqs. (12) and (13).

Potential Impact Number: 24579 **SRP Section Number:** 3.6.2

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Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

For ASME Class 1 piping, the staff position for postulating pipe breaks is delineated in SRP 3.6.2, BTP MEB 3-1. Before the NRC issued Revision 2 of BTP MEB 3-1 in June 1987, breaks were postulated at intermediate locations between terminal ends of a pipe run if the maximum stress range as calculated by Eq.(10) > 2.4 Sm and if the maximum stress range as calculated by either Eq. 12) or (13) > 2.4 Sm, where Eqs. (10), (12), and (13) and Sm are as defined in ASME Section III, Subsection NB 3653. In Revision 2 of BTP MEB 3-1, the same criteria were maintained for break exclusion in the containment penetration areas. However, for other areas, the criteria were revised to require that breaks be postulated at any intermediate locations when only Eq. (10) > 2.4 Sm. The use of Eqs. (12) and (13) was eliminated. This resulted in an inconsistency in the Revision 2 criteria in that they allow higher limits in the containment penetration areas than in other areas. The break exclusion areas should provide a margin greater than (or at least equal to) the margin for areas outside the break exclusion area. To determine the impact of this inconsistency, the staff obtained several independent analyses for both BWRs and PWRs that compared the number of postulated pipe breaks resulting from the use of Revisions 1 and 2 criteria. These analyses indicated that the Revision 2 criteria will result in a significant increase in the number of postulated breaks, which may be counter productive in terms of improving plant safety. Therefore, the staff recommended that SRP 3.6.2 be revised to reinstate the Revision 1 criteria related to allowing the use of Eqs. (12) and (13) for the postulation of intermediate pipe breaks in ASME Class 1 piping systems. Sections 3.6-1 and 3.6-2 in the SSAR contains criteria that are consistent with the Revision 1 criteria as previously discussed. Therefore, the staff concludes that this is an acceptable deviation from SRP 3.6.2, Revision 2.

Potential Impact Number: 24581 **SRP Section Number:** 9.5.3

Document: NUREG 0700
Title: GUIDELINES FOR CONTROL ROOM DESIGN REVIEWS

Summary:

These guidelines for control room design reviews provide the NRC staff guidance that should be followed in accomplishing the detailed control room design review described

in Task I.D.1 of NUREG-0660, the NRC Action Plan. Guidance includes recommendations for planning the review process, review procedures, the assessment of identified human engineering discrepancies to select those that should be corrected, implementation of design improvements, and reporting. Detailed human engineering guidelines are included which provide benchmarks for the human factors quality of specific control room design features.

Sections 6.1.5.3 and 6.1.5.4 provide criteria for lighting systems, illumination, and related design features of the control room to support reliable human performance.

Microfiche No. 09940:204

Potential Impact Number: 24585 **SRP Section Number:** 15.4.1

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

GE examined the design of the rod control system to ascertain if a single failure can lead to the uncontrolled withdrawal of a control rod during refueling and during startup and low power operation. During refueling operations, interlocks insure that all control rods are inserted while fuel is being handled over the core. When no fuel is being handled, a maximum of one rod may be withdrawn. However, the control system is designed (see Section 4.3.2 of this report) so that the core is subcritical with the highest worth rod withdrawn. Finally, a control rod cannot physically be removed (from the top) without removing the four fuel assemblies that surround the rod. therefore, GE has not provided an analysis of control rod removal error during refueling. This is in accord with approvals for current BWRs and is acceptable. (ABWR 15.1(4)(a))

Potential Impact Number: 24586 **SRP Section Number:** 15.4.1

Document: NRC NOTICE 83-35
Title: FUEL MOVEMENT WITH CONTROL RODS WITHDRAWN AT BWRS

Summary:

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BWR Technical Specifications allow the defeating of the one rod refueling interlock to allow multiple control rod drives to be removed for maintenance. This practice is pervasive, and is implemented at virtually every domestic BWR. Further, the Standard BWR Technical Specifications allow fuel movement to take place during control rod removal if certain conditions are met.

Under these conditions, a single error could cause a control rod to be withdrawn that was not planned to be withdrawn. Further, fuel could be loaded into cells in which the control rod has been removed and replaced, but not yet been inserted, which creates the same conditions as an unplanned control rod removal.

Information Notice 83-35 briefly describes two events of this type that occurred at Brunswick Unit 1 and at Duane Arnold. It is clear that under current practices and current licensing positions, an event that establishes the conditions of a control rod removal error can occur.

Potential Impact Number: 24588 **SRP Section Number:** 15.4.6

Document: NRC NOTICE 83-77
Title: AIR/GAS ENTRAINMENT EVENTS RESULTING IN SYSTEM FAILURES

Summary:

This information notice is provided as notification of events that rendered redundant safety systems inoperable because air or gas entrainment caused pump cavitation. The notice describes an event involving a loss of high head safety injection emergency boration and reactor coolant makeup capability. Hydrogen from the positive displacement pump (PDP) suction dampener entered the common suction of the charging system, causing both centrifugal charging pumps and the PDP to be inoperable. This event was described in more detail in IE Information Notice 82-19.

Potential Impact Number: 24589 **SRP Section Number:** 15.4.6

Document: NRC NOTICE 86-63
Title: LOSS OF SAFETY INJECTION CAPABILITY

Summary:

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This notice is to alert recipients to a potentially significant problem pertaining to the loss of safety injection (SI) capability as a result of common mode failure of SI pumps from crystallization of boric acid. Three such events are described involving multiple inoperable safety injection (SI) pumps apparently caused by boric acid crystallization blocking pump suction and/or possible gas binding of the pumps.

Potential Impact Number: 24590 **SRP Section Number:** 15.4.6

Document: NRC NOTICE 91-54
Title: FOREIGN EXPERIENCE REGARDING BORON DILUTION

Summary:

This information notice alerts addressees (PWR license and permit holders) to a potential sequence of events that may result in a rapid injection of unborated reactor coolant water into the reactor core. The adverse events involve the accumulation of a significant volume of unborated water in a part of the primary coolant system during plant shutdown. Various actions performed during plant shutdown could result in such an accumulation. Once the unborated water is accumulated, the startup of an idle reactor coolant pump could send the unborated water into the reactor core causing a significant and unplanned reactivity insertion.

The following sequence of events is evaluated in the foreign study. The sequence is initiated by the loss of one of two offsite electrical power sources. This loss of power occurs while the reactor coolant water is being diluted for reactor startup. At this time, decay heat is insufficient to cause natural circulation of the reactor coolant. It is assumed that the operating reactor coolant pump(s) are powered from the failed offsite power source and cease running. However, the charging pumps, powered from the alternate power source, continue to operate and a significant volume of unborated water is accumulated in the reactor coolant system. If, at this point, an idle reactor coolant pump is started, the accumulated volume of weakly borated or unborated water would be injected into the reactor core. The safety analyses for most U.S. nuclear plants do not address this explicit sequence of events. However, the staff is aware of one representative analysis of a similar sequence for a U.S. PWR. In that analysis, the results indicate that such an accident could cause some core damage but that breach of the reactor coolant system would be unlikely. The NRC is continuing to study other possible sequences that could cause a similar unplanned injection of unborated water. The above postulated sequence of events would occur as a result of actions performed

during plant restart from a shutdown condition and as such is being considered as part of the NRC's review of shutdown risk.

Training and procedures that emphasize the need to ensure uniform boron concentration in the reactor coolant system at all times and the implementation of appropriate caution in starting an idle reactor coolant loop can reduce the probability of occurrence of such an event.

Potential Impact Number: 24591 **SRP Section Number:** 3.5.1.4

Document: NRC MEMORANDUM 03/18/94, from G. Bagchi to L. Cunningham
Title: REVISION OF RG 1.76

Summary:

The Civil Engineering and Geosciences Branch reviewed the tornado pressure drops specified in RG 1.76 and recommended revising the pressure drops for Region II and III.

Potential Impact Number: 24592 **SRP Section Number:** 15.4.6

Document: NRC NOTICE 93-32
Title: NONCONSERVATIVE INPUTS FOR BORON DILUTION EVENT ANALYSIS

Summary:

This information notice informs addressees (PWR licensees and permit holders) that nonconservative assumptions have been identified in the analysis of licensing basis boron dilution events.

Westinghouse-designed plants have used a boron dilution mitigation system (mitigation system) to detect and mitigate a boron dilution event in Modes 3 (hot standby), 4 (hot shutdown), and 5 (cold shutdown). The mitigation system is designed to terminate automatically an inadvertent boron dilution event occurring with the plant in these operating modes before the reactor coolant system is diluted sufficiently to result in a total loss of shutdown margin. A boron dilution event is detected by this system by monitoring the signals of the source range monitors to determine if the neutron flux has

increased by a specified multiplication factor over a prescribed time period. When a dilution event is detected, the mitigation system isolates known dilution paths to the reactor coolant system and realigns the reactor makeup water system to the refueling water storage tank to initiate a re-boration.

If operator action is required during Modes 3, 4, and 5, NRC criteria (see SRP Section 15.4.6) specify at least 15 minutes notice be provided to operators from the time at which an alarm announces an unplanned dilution to the time at which shutdown margin would be eliminated (criticality).

A generic curve is used for the relationship between the inverse count rate ratio and the reactor coolant system boron concentration, and was used in the Westinghouse boron dilution event analysis. The curve is used to predict the time at which a source range flux-doubling signal would occur as criticality is approached during a boron dilution event. This curve and a source range flux multiplication setpoint of 2.0 (flux-doubling) are the bases for detecting and mitigating an inadvertent boron dilution event. The generic curve might not be bounding and that the analysis did not include instrument uncertainties. These deficiencies could cause operation outside the licensing bases for dilution events. The boron dilution event analysis shows that the boron dilution mitigation system response will prevent a return to criticality.

Using actual plant inverse count rate ratio data, a licensee determined that the flux-doubling signal and alarm detected by the mitigation system may not meet acceptance criteria for the boron dilution event since an appreciable amount of dilution would already have occurred before the signal was generated.

The licensee also found that the flux multiplication setpoint of 2.0 does not include the instrument uncertainties that should be applied to determine the flux multiplication setpoint. These uncertainties include the statistical variation of neutron flux, the variations in the linearity of the log level amplifier as a function of counts per second, and uncertainties in numerous amplifiers and the multiplexer. Thus, an actual flux signal could exceed the flux-doubling setpoint used to analyze the boron dilution event. This condition could result in a loss of shutdown margin before the mitigation system terminates an inadvertent boron dilution event. (This problem may be applicable not only to plants that rely on an automatic mitigation system, but also to those that rely on a flux-doubling signal to generate an alarm for operator intervention.) Alternatively, setting the setpoint too conservatively could unnecessarily allow the mitigation system to be activated and the reactor coolant system to be borated. This sequence of events would have detrimental effects on operation.

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The staff met with Westinghouse and licensees of Westinghouse-designed plants which use the mitigation system. It was concluded that the mitigation system was unreliable because of the large uncertainties associated with the indication of a true flux doubling by the nuclear instruments. Therefore, this problem may be common to all PWRs that rely on a flux-doubling signal to mitigate a boron dilution event before criticality can occur.

Potential Impact Number: 24594 **SRP Section Number:** 3.7.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY 90-016 the staff discussed its proposal to decouple the analytical concepts of operating basis earthquake (OBE) and safe shutdown earthquake (SSE). The staff recommended that the Commission approve a review approach which allows decoupling the OBE from the SSE on a design-specific basis for evolutionary designs. In Staff Requirement Memorandum (SRM) for SECY 90-016 the Commission approved the staff's recommendation in this regard.

In SECY 93-087 the staff proposed additional guidance and regulatory positions for the design of structures, systems and components (SSCs) in cases where the OBE is eliminated from the design analysis. The staff evaluated the effect on safety of eliminating the OBE contribution from the analytical design load combinations for selected SSCs and has developed proposed criteria for an analysis using the SSE only.

In the SRM responding to SECY 93-087 the Commission approved specific staff positions and criteria for elimination of the OBE from the design analysis. The specific approved positions and regulatory guidance can be found in section I.M of SECY 93-087 and in the SRM responding to SECY 93-087.

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 a seismic event (so called inspection level earthquake).

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Potential Impact Number: 24595 **SRP Section Number:** 3.7.2

Document: FINAL SER CE80 CH 1
Title: INTRODUCTION AND GENERAL DISCUSSION

Summary:

In accordance with 10 CFR 52.48, the staff used the current regulations in 10 CFR Parts 20, 50, 73, and 100 in reviewing ABB-CE's application for design certification. During this review, the staff recognized that if certain regulations were applied to the System 80+ design, the use of those regulations would not serve the underlying purpose of the rule or would not be needed for achieving the fundamental intent of the rule. These exemptions are discussed in the sections of this report listed below.

In the SRM pertaining to SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor Designs," the Commission approved the staff's recommendation to proceed with design-specific rulemakings as part of individual design certifications to resolve selected technical and severe accident issues as "applicable regulations" for the advanced boiling water reactor and System 80+ standard designs. Among these issues are staff positions that deviate from or are not embodied in current regulations applicable to the System 80+ design.

These issues were addressed in various Commission papers, such as SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced LWR designs." The "applicable regulations" that are specific to the System 80+ design are identified and evaluated in this report. The proposed design certification rule will then include these "applicable regulations" for the System 80+ design for the purposes of 10 CFR 52.48, 52.54, 52.59, and 52.63. These "applicable regulations" are discussed in the sections of this report, including Section 3.1.1, "Exemption from operating-basis earthquake design requirement."

Potential Impact Number: 24596 **SRP Section Number:** 3.7.2

Document: FINAL SER ABWR CH 1
Title: INTRODUCTION AND GENERAL DISCUSSION

Summary:

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In the SRM pertaining to SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor Designs," the Commission approved the staff's recommendation to proceed with design-specific rulemakings as part of design certification rulemakings to establish selected technical and severe accident issues as "applicable regulations" for the ABWR and System 80+ standard designs. These issues included staff positions that deviated from or were not embodied in current regulations applicable to the standard designs.

These issues were addressed in various Commission papers, such as SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolution and Advanced LWR Designs." The "applicable regulations" that are specific to the ABWR design are identified and evaluated in this report, including Section 3.1.1, Exemption from operating basis earthquake design requirement.

Potential Impact Number: 24597 **SRP Section Number:** 5.3.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

ABB-CE takes exception to the recommendations in Position C.2 of RG 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels," for controls imposed on preheat temperatures for welding ferritic steels. These controls would provide reasonable assurance that components made from low-alloy steels will not crack during fabrication and minimize the possibility of subsequent hydrogen cracking in the weldment. In a letter of November 24, 1992, ABB-CE stated 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 NRC evaluated and accepted this report. This report presents three acceptable alternatives, and ABB-CE's position is that a particular alternative will be specified based upon the particulars of the configuration or a particular facilities capabilities. This approach should offer adequate assurance that low-alloy steel weldments will not developing cracking due to hydrogen.

Potential Impact Number: 24598 **SRP Section Number:** 5.3.1

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Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

When components of austenitic stainless steels are welded, ASME Code controls will be supplemented by conformance with the recommendations of RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," Revision 3, and NUREG-0313, Revision 2. These controls will provide reasonable assurance that the welds will not contain microcracks and will be resistant to IGSCC and satisfy GDC 1, 14, and 30 and 10 CFR 50.55a as related to the integrity of the RCPB.

Potential Impact Number: 24599 **SRP Section Number:** 5.3.1

Document: C&S: ASME III
Title: CLASS 1 AND 2 SYSTEMS

Summary:

SRP Section 5.3.1 cites the following Article, Appendix, and Paragraph to the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1:

Article NA-8000

Appendix IX-6000

Paragraph NB-2322a

The latest version of the ASME B&PV Code is the 1992 Edition through the 1994 Addenda. There is a Paragraph NB-2322 in the latest version of the Code but it does not include an "a" designator. Neither of the other two citations appear in the latest version of the code.

Potential Impact Number: 24600 **SRP Section Number:** 5.3.1

Document: C&S: ASTM A262-70

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Title: DETECTING SUSCEPTIBILITY TO INTERGRANULAR ATTACK IN STAINLESS STEEL

Summary:

Regulatory Guide 1.44 is cited in SRP Section 5.3.1 and endorses the use of ASTM A-262 1970. ASTM A-262 has been incorporated into the SRP under Integrated Impact No.973. The latest version of ASTM A-262 is dated 1993. A side-by-side comparison has been performed between the cited and latest version of the standard.

Potential Impact Number: 24601 **SRP Section Number:** 19.2

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY-90-016 the staff requested that the Commission approve certain performance goals related to the ability of the containment to withstand credible severe accident challenges. The position indicates that evaluations of evolutionary designs should use a conditional containment failure probability (CCFP) goal of 0.1 or a deterministic containment performance goal that offers comparable protection. SECY-90-016 provides a description of 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 ALWRs but indicated that this objective should not be imposed as a requirement in and of itself. The Commission also indicated that the use of the CCFP should not discourage accident prevention and the staff should review suitable alternative, deterministically-established, containment performance objectives providing comparable mitigation capability if submitted by applicants.

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Potential Impact Number: 24602 **SRP Section Number:** 19.2

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY-90-016 the staff requested that the Commission approve certain positions related to core debris coolability in the event of a severe accident in which the core melts through the reactor vessel. The position indicates that evolutionary and passive ALWR designs should provide; (1) sufficient reactor cavity floor space to enhance debris spreading, and (2) provide for quenching debris in the reactor cavity.

The position proposed by the staff was approved by the Commission in an SRM dated June 26, 1990. However, the Commission indicated in the SRM that the staff should not limit licensees to only one method for addressing containment responses to severe accident events but also permit other technically justified means for demonstrating adequate containment response.

Potential Impact Number: 24603 **SRP Section Number:** 19.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY-93-087 the staff requested that the Commission approve certain positions related to ensuring adequate containment response to core debris challenges in the event of a severe accident in which the core melts through the reactor vessel. The position indicates that evolutionary and passive ALWR designs should; (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 for steel containments or Factored Load Category for concrete containments, for approximately 24 hours and ensure that the containment capability

has margin to accommodate uncertainties in the environmental conditions from core-concrete interactions. Items (1) and (2) are equivalent to postions originally proposed in SECY-90-016, but items (3) and (4) were added by this SECY paper.

The positions proposed by the staff were approved by the Commission in an SRM dated July 21, 1993. However, the Commission indicated in the SRM that the staff should not limit licensees to only one method for addressing containment responses to severe accident events but also permit other technically justified means for demonstrating adequate containment response.

Potential Impact Number: 24604 **SRP Section Number:** 19.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY-93-087 the staff restated their original positions concerning high pressure core melt ejections from SECY-90-016 (i.e., a reliable vessel depressurization system and cavity design features to decrease the amount of ejected core debris that reaches the upper containment) and recommended that they also be applied to passive plants. The position proposed by the staff was approved by the Commission in an SRM dated July 21, 1993.

Potential Impact Number: 24605 **SRP Section Number:** 19.2

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY-90-016 the staff requested that the Commission approve the use of a containment over-pressure protection system to meet containment vent requirements in the ABWR design certification process. The intent of this design feature is to avoid a

gross containment failure resulting from a postulated slow rising over-pressure severe accident scenario. In an SRM dated June 26, 1990, the Commission approved the staff's recommended use of the containment over-pressure protection system, subject to the results of a comprehensive regulatory review which should fully weigh the potential "downside" risks with the mitigation benefits of the system. The Commission also indicated that the staff should ensure that full capability to maintain control over the venting process is provided.

Potential Impact Number: 24606 **SRP Section Number:** 19.2

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY-90-016, the staff recommended that features provided for severe-accident protection (prevention and mitigation) only (not required 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. The reason for this judgement is that the staff does not believe that severe core damage accidents should be design basis accidents (DBA) in the traditional sense that DBAs have been treated in the past.

Notwithstanding that judgement 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.

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Potential Impact Number: 24607 **SRP Section Number:** 19.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 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. The staff also noted that this position was approved by a June 26, 1990 SRM.

SECY-93-087 also recommends that a policy similar to the SECY-90-016 position be applied to passive plants. The new position for passive plants is the same as the earlier 90-016 position except that it addresses only mitigation equipment. The 90-016 position applies to prevention and mitigation equipment. The passive plant position proposed by the staff was approved by the Commission in an SRM dated July 21, 1993.

Potential Impact Number: 24608 **SRP Section Number:** 19.2

Document: FINAL SER CE80 CH 19
Title: SEVERE ACCIDENTS

Summary:

The NRC has issued guidance for addressing severe accidents. This guidance is in (1) the NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, (2) the NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants, (3) the NRC Policy Statement on Nuclear Power Plant Standardization, (4) 10 CFR Part 52, "Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants," (5) SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current

Regulatory Requirements," and the corresponding staff requirements memorandum (SRM) dated June 26, 1990, and (6) SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," and the corresponding SRM dated July 21, 1993. Whereas, the first three documents provide guidance as to the appropriate course for addressing severe accidents, 10 CFR Part 52 contains general requirements for addressing severe accidents, and the SRMs relating to SECY-90-016 and SECY-93-087 give Commission-approved positions for implementing features for preventing severe accidents and mitigating their effects. In SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor (LWR) Designs," the staff discussed two options for proceeding with severe-accident rulemaking for the evolutionary LWR designs through the individual design certification process or generic rulemaking. In an SRM dated January 28, 1992, the Commission approved the staff's recommendation to proceed with design-specific rulemakings through individual design certifications to resolve selected technical and severe- accident issues. The effect of these actions on the System 80+ is that the criteria specified for resolving severe-accident issues in SECY-90-016 and SECY-93-087 will be incorporated into the System 80+ design certification rulemaking as applicable regulations.

Potential Impact Number: 24609 **SRP Section Number:** 19.2

Document: FINAL SER ABWR CH 19
Title: SEVERE ACCIDENTS

Summary:

As part of the ABWR design certification application, GE submitted the ABWR PRA in response to 10 CFR 52.47, the Commission's Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, described in FR Vol. 50, No. 153, dated August 8, 1988, p. 32138 dated August 8, 1991, and the ABWR Licensing Review Bases. The staff's assessment included the traditional evaluation of events that could lead to core damage and offsite consequences as well as an evaluation of what the ABWR PRA revealed about the ABWR design.

In an SRM, dated January 28, 1992, on SECY-91-262, the Commission approved the staff's recommendation to proceed with design-specific rulemakings through individual design certifications to resolve selected technical and severe accident issues. The effect of these actions on the ABWR is that the criteria specified for resolution of severe

accident issues in SECY-90-016 and SECY-93-087 will be incorporated into the ABWR design certification rulemaking as applicable regulations.

Potential Impact Number: 24610 **SRP Section Number:** 19.2

Document: NUREG 1070
Title: NRC POLICY ON FUTURE REACTOR DESIGNS

Summary:

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.

Potential Impact Number: 24611 **SRP Section Number:** 19.2

Document: NRC POLICY STATEMENT 50 FR 32138
Title: POLICY STATEMENT ON SEVERE REACTOR ACCIDENTS REGARDING FUTURE DESIGNS AND EXISTING PLANTS

Summary:

This statement describes the policy the Commission intends to use to resolve safety issues related to reactor accidents more severe than design basis accidents. Its main focus is on the criteria and procedures the Commission intends to use to certify new designs for nuclear power plants.

Policy for New Plant Applications
1. Introduction - No new commercial nuclear reactors have been ordered in the United States since December 1978. However, the Commission has received several applications for reference design approvals that are currently under review. A reference design is one of the options in the Commission's standardization policy. When approved by the NRC staff, a reference design could be incorporated by reference in a new CP application and, ultimately, in an Operating License (OL) application. During the corresponding CP and OL reviews, the NRC staff would not duplicate that portion of its review encompassed by its reference design approval. Therefore, even in the absence of new CP applications, in order to provide guidelines for the current reference

design reviews, the Commission has recognized the need to promptly establish the criteria by which new designs can be shown to be acceptable in meeting severe accident concerns. The Commission now believes that there exists an adequate basis from which to establish an appropriate set of criteria. This belief is supported by current operating reactor experience, ongoing severe accident research, and insights from a variety of risk analyses. The resultant criteria and procedural requirements are listed below.

2. Criteria and Procedural Requirements - The Commission believes that a new design for a nuclear power plant (as well as a proposed custom 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 as reflected in the CP Rule (10 CFR 50.34(f)):

b. Demonstration of technical resolution of all applicable Unresolved Safety Issues and the medium- and high-priority Generic Safety Issues, including a special focus on assuring the reliability of decay heat removal systems and the reliability of both AC and DC electrical supply systems;

c. Completion of a Probabilistic Risk Assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety; 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.

This policy statement also declares the following: "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."

Potential Impact Number: 24617 **SRP Section Number:** 3.9.3

Document: NRC GENERIC LETTER 80-95
Title: FINAL EDITION OF NUREG-0619, 'BWR FEEDWATER NOZZLE AND CONTROL ROD DRIVE RETURN LINE NOZZLE CRACKING'

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Summary:

The Generic Letter forwarded the final edition of NUREG- 0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking." The report provides the staff's resolution of the NRC's Generic Technical Activity A-10, which is an "Unresolved Safety Issue" pursuant to Section 210 of the Energy Reorganization Act of 1974. Public comments received in response to the "For Comment" edition of April 1980, have been incorporated where applicable. Appendix E of NUREG-0619 discusses the comments and the staff's disposition of these comments. The generic study resulted from the inservice discovery of cracking in feedwater nozzles and control rod drive return line nozzles. It is expected that requirements contained in the NUREG will be met.

Potential Impact Number: 24618 **SRP Section Number:** 3.9.3

Document: NRC GENERIC LETTER 81-11
Title: BWR FEEDWATER NOZZLE CRACKING

Summary:

By letter dated November 13, 1980 licensees were a forwarded a copy of NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking" (November 1980). An error has been found in footnote 2 to Table 2 on page 18 of NUREG-0619. Because of the deletion of leak-testing requirements, footnote 2 should now read: "To be performed even if UT results are satisfactory."

NUREG-0619 was forwarded to PWR licensees and applicants for their information only, in consideration of the fact that the PWR steam generator feedwater line cracking has been attributed to the same phenomenon of thermal fatigue as existed in the BWR feedwater nozzles. No response from PWR licensees and applicants is required.

Potential Impact Number: 24619 **SRP Section Number:** 3.9.3

Document: NUREG 0619
Title: BWR FEEDWATER NOZZLE AND CONTROL ROD DRIVE RETURN LONE NOZZLE CRACKING

Summary:

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The report provides the staff's resolution of the NRC's Generic Technical Activity A-10, which is an "Unresolved Safety Issue" pursuant to Section 210 of the Energy Reorganization Act of 1974. Public comments received in response to the "For Comment" edition of April 1980, have been incorporated where applicable. Appendix E of NUREG-0619 discusses the comments and the staff's disposition of these comments.

The generic study resulted from the inservice discovery of cracking in feedwater nozzles and control rod drive return line nozzles. NUREG-0619 describes the technical issues, the technical studies and 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.

Microfiche No. 07219-166

Potential Impact Number: 24620 **SRP Section Number:** 3.6.1

Document: RG 1.153 **Revision:** 0
Title: CRITERIA FOR POWER, INSTRUMENTATION, AND CONTROL

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with respect to the design, reliability, qualification, and testability of the power, instrumentation, and control portions of safety-related systems.

Potential Impact Number: 24623 **SRP Section Number:** 9.2.2

Document: RG 1.153 **Revision:** 0
Title: CRITERIA FOR POWER, INSTRUMENTATION, AND CONTROL

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with respect to the design, reliability, qualification, and testability of the power, instrumentation, and control portions of safety-related systems.

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Potential Impact Number: 24637 **SRP Section Number:** 13.1.2 - 13.1.3

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor (STA) may serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units.

Clarification

The letter of October 30, 1979 clarified the short-term STA requirements. That letter indicated that the STAs must have completed all training by January 1, 1981. This paper confirms these requirements and requests additional information.

The need for the STA position may be eliminated when the qualifications of the shift supervisors and senior operators have been upgraded and the man-machine interface in the control room has been acceptably upgraded. However, until those long-term improvements are attained, the need for an STA program will continue.

Applicability

This requirement applies to all licensees of operating reactors and applicants for operating licenses.

Potential Impact Number: 24638 **SRP Section Number:** 13.2.2

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

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Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor (STA) may serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units.

Clarification

The letter of October 30, 1979 clarified the short-term STA requirements. That letter indicated that the STAs must have completed all training by January 1, 1981. This paper confirms these requirements and requests additional information.

The need for the STA position may be eliminated when the qualifications of the shift supervisors and senior operators have been upgraded and the man-machine interface in the control room has been acceptably upgraded. However, until those long-term improvements are attained, the need for an STA program will continue.

Applicability

This requirement applies to all licensees of operating reactors and applicants for operating licenses.

Potential Impact Number: 24639 **SRP Section Number:** 13.1.2 - 13.1.3

Document: NRC GENERIC LETTER 86-04
Title: POLICY STATEMENT ON ENGINEERING EXPERTISE ON SHIFT

Summary:

The letter informs licensees and applicants of a Policy Statement regarding Engineering Expertise on Shift.

The Policy Statement offers the licensees two options for meeting the current requirements for providing engineering expertise on shift (NUREG-0737, Item I.A.1.1) and meeting licensed operator staffing requirements (10 CFR 50.54 (m)(2)). Option 1 provides for elimination of the separate Shift Technical Advisor (STA) position by allowing licensees to combine one of the required Senior Reactor Operator (SRO) positions with the STA position into a dual-role (SRO/STA) position. Option 2 states that a licensee may continue to use an NRC-approved STA program while meeting licensed operator staffing requirements. Licensees may use either option on each shift.

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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. However, the Commission realizes that some licensees may prefer the dedicated STA position for a number of reasons. If this latter option is chosen, then the Commission encourages licensees to work towards having the dedicated STA assume an active role in shift activities by having the STAs rotate with a shift and by including responsibilities to review plant logs, participate in shift turnover activities and training, and maintain an awareness of plant configuration and status.

Potential Impact Number: 24640 **SRP Section Number:** 13.2.2

Document: NRC GENERIC LETTER 86-04
Title: POLICY STATEMENT ON ENGINEERING EXPERTISE ON SHIFT

Summary:

The letter informs licensees and applicants of a Policy Statement regarding Engineering Expertise on Shift.

The Policy Statement offers the licensees two options for meeting the current requirements for providing engineering expertise on shift (NUREG-0737, Item I.A.1.1) and meeting licensed operator staffing requirements (10 CFR 50.54 (m)(2)). Option 1 provides for elimination of the separate Shift Technical Advisor (STA) position by allowing licensees to combine one of the required Senior Reactor Operator (SRO) positions with the STA position into a dual-role (SRO/STA) position. Option 2 states that a licensee may continue to use an NRC-approved STA program while meeting licensed operator staffing requirements. Licensees may use either option on each shift.

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. However, the Commission realizes that some licensees may prefer the dedicated STA position for a number of reasons. If this latter option is chosen, then the Commission encourages licensees to work towards having the dedicated STA assume an active role in shift activities by having the STAs rotate with a shift and by including responsibilities to review plant logs, participate in shift turnover activities and training, and maintain an awareness of plant configuration and status.

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Potential Impact Number: 24653 **SRP Section Number:** 13.1.2 - 13.1.3

Document: NRC POLICY STATEMENT 50 FR 43621
Title: COMMISSION POLICY STATEMENT ON ENGINEERING EXPERTISE ON SHIFT

Summary:

This Policy Statement presents the policy of the Nuclear Regulatory Commission (NRC) with respect to ensuring that adequate engineering and accident assessment expertise is possessed by the operating staff at a nuclear power plant. This Policy Statement offers licensees two options for providing engineering expertise on shift and meeting licensed operator staffing requirements. Option 1 provides for elimination of the separate Shift Technical Advisor (STA) position by allowing licensees to combine one of the required Senior Reactor Operator (SRO) positions with the STA position into a dual-role (SRO/ STA) position. Option 2 provides that a licensee may Continue to use an NRC- approved STA program, with certain modifications, while meeting licensed operator staffing requirements.

Potential Impact Number: 24654 **SRP Section Number:** 13.2.2

Document: NRC POLICY STATEMENT 50 FR 43621
Title: COMMISSION POLICY STATEMENT ON ENGINEERING EXPERTISE ON SHIFT

Summary:

This Policy Statement presents the policy of the Nuclear Regulatory Commission (NRC) with respect to ensuring that adequate engineering and accident assessment expertise is possessed by the operating staff at a nuclear power plant. This Policy Statement offers licensees two options for providing engineering expertise on shift and meeting licensed operator staffing requirements. Option 1 provides for elimination of the separate Shift Technical Advisor (STA) position by allowing licensees to combine one of the required Senior Reactor Operator (SRO) positions with the STA position into a dual-role (SRO/ STA) position. Option 2 provides that a licensee may Continue to use an NRC- approved STA program, with certain modifications, while meeting licensed operator staffing requirements.

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Potential Impact Number: 24657 **SRP Section Number:** 15.2.8

Document: NRC GENERIC LETTER 80-33
Title: ACTIONS REQUIRED FROM OPERATING LICENSE APPLICANTS OF NUCLEAR STEAM SUPPLY SYSTEMS DESIGNED BY BABCOCK AND WILCOX RESULTING FROM THE NRC BULLETINS AND ORDERS TASK FORCE REVIEW REGARDING THE THREE MILE ISLAND UNIT 2 ACCIDENT

Summary:

This letter was sent to B&W operating license applicants.

The letter outlines actions to be taken by B&W applicants to address TMI Action Plan item II.E.1.1. The letter indicates that the auxiliary feedwater reliability analysis and reliability enhancement recommendations of NUREG 0611, that addresses Westinghouse designed PWRs, are applicable to B&W plants.

Potential Impact Number: 24658 **SRP Section Number:** 15.1.5

Document: NRC GENERIC LETTER 80-33
Title: ACTIONS REQUIRED FROM OPERATING LICENSE APPLICANTS OF NUCLEAR STEAM SUPPLY SYSTEMS DESIGNED BY BABCOCK AND WILCOX RESULTING FROM THE NRC BULLETINS AND ORDERS TASK FORCE REVIEW REGARDING THE THREE MILE ISLAND UNIT 2 ACCIDENT

Summary:

This letter was sent to B&W operating license applicants.

The letter outlines actions to be taken by B&W applicants to address TMI Action Plan item II.E.1.1. The letter indicates that the auxiliary feedwater reliability analysis and reliability enhancement recommendations of NUREG 0611, that addresses Westinghouse designed PWRs, are applicable to B&W plants.

Potential Impact Number: 24659 **SRP Section Number:** 15.2.6

Document: NRC GENERIC LETTER 80-33

APPENDIX II
Potential Impacts

Title: ACTIONS REQUIRED FROM OPERATING LICENSE APPLICANTS OF NUCLEAR STEAM SUPPLY SYSTEMS DESIGNED BY BABCOCK AND WILCOX RESULTING FROM THE NRC BULLETINS AND ORDERS TASK FORCE REVIEW REGARDING THE THREE MILE ISLAND UNIT 2 ACCIDENT

Summary:

This letter was sent to B&W operating license applicants.

The letter outlines actions to be taken by B&W applicants to address TMI Action Plan item II.E.1.1. The letter indicates that the auxiliary feedwater reliability analysis and reliability enhancement recommendations of NUREG 0611, that addresses Westinghouse designed PWRs, are applicable to B&W plants.

Potential Impact Number: 24660 **SRP Section Number:** 15.2.7

Document: NRC GENERIC LETTER 80-33
Title: ACTIONS REQUIRED FROM OPERATING LICENSE APPLICANTS OF NUCLEAR STEAM SUPPLY SYSTEMS DESIGNED BY BABCOCK AND WILCOX RESULTING FROM THE NRC BULLETINS AND ORDERS TASK FORCE REVIEW REGARDING THE THREE MILE ISLAND UNIT 2 ACCIDENT

Summary:

This letter was sent to B&W operating license applicants.

The letter outlines actions to be taken by B&W applicants to address TMI Action Plan item II.E.1.1. The letter indicates that the auxiliary feedwater reliability analysis and reliability enhancement recommendations of NUREG 0611, that addresses Westinghouse designed PWRs, are applicable to B&W plants.

Potential Impact Number: 24662 **SRP Section Number:** 15.2.8

Document: NRC GENERIC LETTER 90-04
Title: REQUEST FOR INFORMATION ON THE STATUS OF LICENSEE IMPLEMENTATION OF GENERIC SAFETY ISSUES RESOLVED WITH IMPOSITION OF REQUIREMENTS OR CORRECTIVE ACTIONS

APPENDIX II
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Summary:

This letter discusses the resolution of GSI 124, 'Auxiliary Feedwater System Reliability', that was established as a follow-on activity after implementation of upgrades to the auxiliary feedwater systems under TMI Action Plan Items II.E.1.1 and I.E.1.2. Based on evaluation of AFW system reliability studies for various plants, the staff subsequently determined that three-pump AFW systems demonstrated significantly greater reliability than did two-pump systems and, therefore, limited this issue to two-pump plants.

The staff performed plant-specific reviews of the reliability of the AFW systems in the above plants, including assessments of the system design, operating experience, and emergency procedures. From these reviews, the staff determined whether additional means of secondary decay heat removal capability was necessary. No further hardware modification was determined to be required for ANO-1 and Prairie Island 1 & 2 on the basis of the startup feedwater pump and AFW system sharing capability, respectively. The licensees for Rancho Seco, Crystal River-3 and Ft. Calhoun committed to install additional means of secondary decay heat removal, thereby resolving the issue. The staff issued a plant-specific backfit analysis for ANO-2 requiring the addition of a third train of secondary decay heat removal.

Potential Impact Number: 24663 **SRP Section Number:** 15.1.5

Document: NRC GENERIC LETTER 90-04
Title: REQUEST FOR INFORMATION ON THE STATUS OF LICENSEE IMPLEMENTATION OF GENERIC SAFETY ISSUES RESOLVED WITH IMPOSITION OF REQUIREMENTS OR CORRECTIVE ACTIONS

Summary:

This letter discusses the resolution of GSI 124, 'Auxiliary Feedwater System Reliability', that was established as a follow-on activity after implementation of upgrades to the auxiliary feedwater systems under TMI Action Plan Items II.E.1.1 and I.E.1.2. Based on evaluation of AFW system reliability studies for various plants, the staff subsequently determined that three-pump AFW systems demonstrated significantly greater reliability than did two-pump systems and, therefore, limited this issue to two-pump plants.

The staff performed plant-specific reviews of the reliability of the AFW systems in the above plants, including assessments of the system design, operating experience, and emergency procedures. From these reviews, the staff determined whether additional

means of secondary decay heat removal capability was necessary. No further hardware modification was determined to be required for ANO-1 and Prairie Island 1 & 2 on the basis of the startup feedwater pump and AFW system sharing capability, respectively. The licensees for Rancho Seco, Crystal River-3 and Ft. Calhoun committed to install additional means of secondary decay heat removal, thereby resolving the issue. The staff issued a plant-specific backfit analysis for ANO-2 requiring the addition of a third train of secondary decay heat removal.

Potential Impact Number: 24664 **SRP Section Number:** 15.2.6

Document: NRC GENERIC LETTER 90-04
Title: REQUEST FOR INFORMATION ON THE STATUS OF LICENSEE IMPLEMENTATION OF GENERIC SAFETY ISSUES RESOLVED WITH IMPOSITION OF REQUIREMENTS OR CORRECTIVE ACTIONS

Summary:

This letter discusses the resolution of GSI 124, 'Auxiliary Feedwater System Reliability', that was established as a follow-on activity after implementation of upgrades to the auxiliary feedwater systems under TMI Action Plan Items II.E.1.1 and I.E.1.2. Based on evaluation of AFW system reliability studies for various plants, the staff subsequently determined that three-pump AFW systems demonstrated significantly greater reliability than did two-pump systems and, therefore, limited this issue to two-pump plants.

The staff performed plant-specific reviews of the reliability of the AFW systems in the above plants, including assessments of the system design, operating experience, and emergency procedures. From these reviews, the staff determined whether additional means of secondary decay heat removal capability was necessary. No further hardware modification was determined to be required for ANO-1 and Prairie Island 1 & 2 on the basis of the startup feedwater pump and AFW system sharing capability, respectively. The licensees for Rancho Seco, Crystal River-3 and Ft. Calhoun committed to install additional means of secondary decay heat removal, thereby resolving the issue. The staff issued a plant-specific backfit analysis for ANO-2 requiring the addition of a third train of secondary decay heat removal.

Potential Impact Number: 24665 **SRP Section Number:** 15.2.7

APPENDIX II
Potential Impacts

Document: NRC GENERIC LETTER 90-04
Title: REQUEST FOR INFORMATION ON THE STATUS OF LICENSEE IMPLEMENTATION OF GENERIC SAFETY ISSUES RESOLVED WITH IMPOSITION OF REQUIREMENTS OR CORRECTIVE ACTIONS

Summary:

This letter discusses the resolution of GSI 124, 'Auxiliary Feedwater System Reliability', that was established as a follow-on activity after implementation of upgrades to the auxiliary feedwater systems under TMI Action Plan Items II.E.1.1 and I.E.1.2. Based on evaluation of AFW system reliability studies for various plants, the staff subsequently determined that three-pump AFW systems demonstrated significantly greater reliability than did two-pump systems and, therefore, limited this issue to two-pump plants.

The staff performed plant-specific reviews of the reliability of the AFW systems in the above plants, including assessments of the system design, operating experience, and emergency procedures. From these reviews, the staff determined whether additional means of secondary decay heat removal capability was necessary. No further hardware modification was determined to be required for ANO-1 and Prairie Island 1 & 2 on the basis of the startup feedwater pump and AFW system sharing capability, respectively. The licensees for Rancho Seco, Crystal River-3 and Ft. Calhoun committed to install additional means of secondary decay heat removal, thereby resolving the issue. The staff issued a plant-specific backfit analysis for ANO-2 requiring the addition of a third train of secondary decay heat removal.

Potential Impact Number: 24667 **SRP Section Number:** 15.2.8

Document: NUREG 0611
Title: Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants

Summary:

The purpose of this report is to summarize the results of the Bulletins & Orders Task Force generic review of feedwater transients, small-break LOCAs, and other Three Mile Island Unit 2 types of events in Westinghouse-designed operating plants and to confirm the bases for their continued operation. The results of this evaluation are presented in this report in the form of a set of findings and recommendations in each of the principal review areas.

APPENDIX II
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Potential Impact Number: 24668 **SRP Section Number:** 15.1.5

Document: NUREG 0611
Title: Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants

Summary:

The purpose of this report is to summarize the results of the Bulletins & Orders Task Force generic review of feedwater transients, small-break LOCAs, and other Three Mile Island Unit 2 types of events in Westinghouse-designed operating plants and to confirm the bases for their continued operation. The results of this evaluation are presented in this report in the form of a set of findings and recommendations in each of the principal review areas.

Potential Impact Number: 24669 **SRP Section Number:** 15.2.6

Document: NUREG 0611
Title: Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants

Summary:

The purpose of this report is to summarize the results of the Bulletins & Orders Task Force generic review of feedwater transients, small-break LOCAs, and other Three Mile Island Unit 2 types of events in Westinghouse-designed operating plants and to confirm the bases for their continued operation. The results of this evaluation are presented in this report in the form of a set of findings and recommendations in each of the principal review areas.

Potential Impact Number: 24670 **SRP Section Number:** 15.2.7

Document: NUREG 0611
Title: Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants

APPENDIX II
Potential Impacts

Summary:

The purpose of this report is to summarize the results of the Bulletins & Orders Task Force generic review of feedwater transients, small-break LOCAs, and other Three Mile Island Unit 2 types of events in Westinghouse-designed operating plants and to confirm the bases for their continued operation. The results of this evaluation are presented in this report in the form of a set of findings and recommendations in each of the principal review areas.

Potential Impact Number: 24672 **SRP Section Number:** 15.2.8

Document: NUREG 0635
Title: Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Combustion Engineering Designed Operating Plants

Summary:

This report summarizes the results of our generic evaluation of feedwater transients, small-break LOCAs, and other TMI-2-related events for the Combustion Engineering-designed operating plants, and confirms the bases for their continued operation. The results of this evaluation are presented in this report in the form of a set of findings and recommendations in each of the principal review areas.

Potential Impact Number: 24673 **SRP Section Number:** 15.1.5

Document: NUREG 0635
Title: Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Combustion Engineering Designed Operating Plants

Summary:

This report summarizes the results of our generic evaluation of feedwater transients, small-break LOCAs, and other TMI-2-related events for the Combustion Engineering-designed operating plants, and confirms the bases for their continued operation. The results of this evaluation are presented in this report in the form of a set of findings and recommendations in each of the principal review areas.

APPENDIX II
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Potential Impact Number: 24674 **SRP Section Number:** 15.2.6

Document: NUREG 0635
Title: Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Combustion Engineering Designed Operating Plants

Summary:

This report summarizes the results of our generic evaluation of feedwater transients, small-break LOCAs, and other TMI-2-related events for the Combustion Engineering-designed operating plants, and confirms the bases for their continued operation. The results of this evaluation are presented in this report in the form of a set of findings and recommendations in each of the principal review areas.

Potential Impact Number: 24675 **SRP Section Number:** 15.2.7

Document: NUREG 0635
Title: Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Combustion Engineering Designed Operating Plants

Summary:

This report summarizes the results of our generic evaluation of feedwater transients, small-break LOCAs, and other TMI-2-related events for the Combustion Engineering-designed operating plants, and confirms the bases for their continued operation. The results of this evaluation are presented in this report in the form of a set of findings and recommendations in each of the principal review areas.

Potential Impact Number: 24677 **SRP Section Number:** 15.2.8

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(1)(ii)

APPENDIX II
Potential Impacts

Summary:

This item requires an evaluation of the proposed auxiliary feedwater system that includes 1) a system reliability analysis, 2) a design review of AFW, and 3) an evaluation of AFWS flow design bases.

Potential Impact Number: 24678 **SRP Section Number:** 15.1.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(1)(ii)

Summary:

This item requires an evaluation of the proposed auxiliary feedwater system that includes 1) a system reliability analysis, 2) a design review of AFW, and 3) an evaluation of AFWS flow design bases.

Potential Impact Number: 24679 **SRP Section Number:** 15.2.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(1)(ii)

Summary:

This item requires an evaluation of the proposed auxiliary feedwater system that includes 1) a system reliability analysis, 2) a design review of AFW, and 3) an evaluation of AFWS flow design bases.

Potential Impact Number: 24680 **SRP Section Number:** 15.2.7

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

APPENDIX II
Potential Impacts

Location: 50.34(f)(1)(ii)

Summary:

This item requires an evaluation of the proposed auxiliary feedwater system that includes 1) a system reliability analysis, 2) a design review of AFW, and 3) an evaluation of AFWS flow design bases.

Potential Impact Number: 24682 **SRP Section Number:** 15.2.8

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

The Office of Nuclear Reactor Regulation is requiring reevaluation of the auxiliary feedwater (AFW) systems for all PWR operating plant licensees and operating license applications. This action includes:

(1) Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater-transient conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities, and test and maintenance outages;

(2) Perform 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 as principal guidance; and

(3) Reevaluate the AFW system flowrate design bases and criteria.

Applicability

This requirement applies to all PWR operating plants and applicants for operating licenses.

APPENDIX II
Potential Impacts

Potential Impact Number: 24683 **SRP Section Number:** 15.1.5

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

The Office of Nuclear Reactor Regulation is requiring reevaluation of the auxiliary feedwater (AFW) systems for all PWR operating plant licensees and operating license applications. This action includes:

(1) Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater-transient conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities, and test and maintenance outages;

(2) Perform 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 as principal guidance; and

(3) Reevaluate the AFW system flowrate design bases and criteria.

Applicability

This requirement applies to all PWR operating plants and applicants for operating licenses.

Potential Impact Number: 24684 **SRP Section Number:** 15.2.6

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

APPENDIX II
Potential Impacts

The Office of Nuclear Reactor Regulation is requiring reevaluation of the auxiliary feedwater (AFW) systems for all PWR operating plant licensees and operating license applications. This action includes:

(1) Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater-transient conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities, and test and maintenance outages;

(2) Perform 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 as principal guidance; and

(3) Reevaluate the AFW system flowrate design bases and criteria.

Applicability

This requirement applies to all PWR operating plants and applicants for operating licenses.

Potential Impact Number: 24685 **SRP Section Number:** 15.2.7

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

The Office of Nuclear Reactor Regulation is requiring reevaluation of the auxiliary feedwater (AFW) systems for all PWR operating plant licensees and operating license applications. This action includes:

(1) Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater-transient conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities, and test and maintenance outages;

(2) Perform 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 as principal guidance; and

(3) Reevaluate the AFW system flowrate design bases and criteria.

Applicability

This requirement applies to all PWR operating plants and applicants for operating licenses.

Potential Impact Number: 24686 **SRP Section Number:** 13.1.2 - 13.1.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue called for a decision on the need for engineering expertise on shift. This decision was to be based in part on the functions and duties required by using the results of the job/task analysis and evaluating the STA experience. A policy statement on engineering expertise on shift was to be issued and its effectiveness evaluated. This issue was identified in Table 7 of the NRC 1985 Annual Report as Item 1.1 of the HFPP but was made Item HF1.2 in June 1986 . The final policy statement was approved by the Commission on September 12, 1985 and was published in the Federal Register on October 28, 1985 (50 FR 43621).

This issue has been RESOLVED and no new requirements were established.

Potential Impact Number: 24687 **SRP Section Number:** 13.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

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Potential Impacts

This issue called for a decision on the need for engineering expertise on shift. This decision was to be based in part on the functions and duties required by using the results of the job/task analysis and evaluating the STA experience. A policy statement on engineering expertise on shift was to be issued and its effectiveness evaluated. This issue was identified in Table 7 of the NRC 1985 Annual Report as Item 1.1 of the HFPP but was made Item HF1.2 in June 1986 . The final policy statement was approved by the Commission on September 12, 1985 and was published in the Federal Register on October 28, 1985 (50 FR 43621).

This issue has been RESOLVED and no new requirements were established.

Potential Impact Number: 24690 **SRP Section Number:** 13.2.1

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Pending accreditation of training institutions, licensees and applicants for operating licenses will assure that training center and facility instructors who teach systems, integrated responses, transient, and simulator courses demonstrate senior reactor operator (SRO) qualifications and be enrolled in appropriate requalification programs.

I.A.2.3 of NUREG-0737 refers to Letter from H. R. Denton, NRC, to All Power Reactor Applicants and Licensees (Generic Letter 80-26), dated March 28, 1980, that provides a number of positions related to Operator qualifications and training. A copy of that letter is included in NUREG-0737 under item I.A.2.1.

Clarification

The above position is a short-term position. In the future, accreditation of training institutions will include review of the procedure for certification of instructors.

Applicability

This requirement applies to all operating reactors and operating license applicants.

APPENDIX II
Potential Impacts

Potential Impact Number: 24691 **SRP Section Number:** 13.2.1

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Simulator examinations will be included as part of the licensing examinations.

Applicability

This requirement applies to all applicants for operator and senior operator licenses at power reactors.

Potential Impact Number: 24692 **SRP Section Number:** 13.2.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Summary:

Establishes requirements for operator qualifications including requirements related to simulators.

Potential Impact Number: 24693 **SRP Section Number:** 13.2.1

Document: NUREG 1258
Title: Evaluation Procedure for Simulation Facilities Certified Under 10 CFR 55

Summary:

APPENDIX II
Potential Impacts

This document describes the procedure to be followed by NRC for the inspection of simulation facilities certified under 10 CFR 55. Inspections are divided into four areas based on types of evaluations conducted: 1) performance testing; 2) physical fidelity/human factors; 3) control capabilities; and 4) design, updating, modification, and testing. NRC staff representing several disciplines including license examiners, operations specialists, and human factors experts, under direction of a team leader, will perform these inspections. A simulation facility inspection may include on- site and off-site phases. The off-site phase will consist of an examination of simulation facility documentation and an identification of those operations and procedures which may be considered for use in performance testing during the on-site phase. In the on-site review, the staff will work closely with the facility licensee to conduct a sound and fair inspection and to evaluate the results of those tests that are conducted. Inspection findings will be based on staff judgment of the simulation facility's compliance w/10 CFR 55.45. Findings may range from 'no adverse impact on the conduct of operating tests' through degrees of 'adverse impact requiring correction' to 'adverse impacts so serious that the simulation facility may not be used in the conduct of operating tests until the discrepancies are corrected and the simulation facility is recertified to the NRC.'

Potential Impact Number: 24694 **SRP Section Number:** 13.2.1

Document: RG 1.149 **Revision:** 1
Title: NUCLEAR POWER PLANT SIMULATION FACILITIES FOR USE IN OPERATOR LICENSE EXAMINATIONS

Summary:

This regulatory guide describes a method acceptable to the NRC staff for complying with those portions of the Commission's regulations regarding (1) certification of a simulation facility consisting solely of a plant referenced simulator and (2) application for prior approval of a simulation facility.

Potential Impact Number: 24695 **SRP Section Number:** 13.1.1

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

APPENDIX II
Potential Impacts

Position

Each applicant for an operating license shall establish an onsite independent safety engineering group (ISEG) to perform independent reviews of plant operations.

The principal function of the ISEG is to examine plant operating characteristics, NRC issuances, Licensing Information Service advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety. The ISEG is to perform independent review and audits of plant activities including maintenance, modifications, operational problems, and operational analysis, and aid in the establishment of programmatic requirements for plant activities. Where useful improvements can be achieved, it is expected that this group will develop and present detailed recommendations to corporate management for such things as revised procedures or equipment modifications.

Another function of the ISEG is to maintain surveillance of plant operations and maintenance activities to provide independent verification that these activities are performed correctly and that human errors are reduced as far as practicable. ISEG will then be in a position to advise utility management on the overall quality and safety of operations. ISEG need not perform detailed audits of plant operations and shall not be responsible for sign-off functions such that it becomes involved in the operating organization.

Applicability

This requirement applies to all applicants for operating license.

Potential Impact Number: 24696 **SRP Section Number:** 15.0

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

In letters of September 13 and 27, October 10 and 30, and November 9, 1979, the Office of Nuclear Reactor Regulation required licensees of operating plants, applicants for operating licenses and licensees of plants under construction to perform analyses of

transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, including procedures for operating with natural circulation conditions, and to conduct operator retraining (see also item I.A.2.1). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed.

Applicability

This requirement applies to all operating reactors and applicants for operating license.

Potential Impact Number: 24697 **SRP Section Number:** 15.0

Document: NUREG 0737 SUPPLEMENT 1 Revision Date: 1/1/83
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS - REQUIREMENTS FOR EMERGENCY RESPONSE CAPABILITY

Summary:

Section 7 of NUREG-0737, Supplement 1, describes criteria for the development of emergency operating procedures and supporting technical guidelines and writers guide documents.

Potential Impact Number: 24699 **SRP Section Number:** 5.4.7

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necessary, 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. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations and maintenance

activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both. Implementation of automatic status monitoring if required will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such verification in all instances. The procedures adopted by the licensees may consist of two phases--one before and one after installation of automatic status monitoring equipment, if required, in accordance with item I.D.3.

Applicability

This requirement applies to all operating reactor and operating license applicants.

Potential Impact Number: 24700 **SRP Section Number:** 6.3

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necessary, 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. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both. Implementation of automatic status monitoring if required will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such verification in all instances. The procedures adopted by the licensees may consist of two phases--one before and one after installation of automatic status monitoring equipment, if required, in accordance with item I.D.3.

Applicability

This requirement applies to all operating reactor and operating license applicants.

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Potential Impacts

Potential Impact Number: 24701 **SRP Section Number:** 14.2

Document: NRC GENERIC LETTER 80-57
Title: FURTHER COMMISSION GUIDANCE FOR POWER REACTOR OPERATING LICENSES

Summary:

This letter informs operating license applicants of the development of NUREG 0694 and makes the following statement regarding this document:

On June 16, 1980, the Nuclear Regulatory Commission issued its policy statement regarding the requirements to be met for current operating license applications. The requirements are derived from the NRC's Action Plan (NUREG-0660) and are found in NUREG-0694, "TMI-Related Requirements for New Operating Licenses". They are deemed to be necessary and sufficient for responding to the TMI-2 accident, and current operating license applications should be measured against the NRC regulations a augmented by these requirements.

Potential Impact Number: 24703 **SRP Section Number:** 14.2

Document: NUREG 0694
Title: TMI - RELATED REQUIREMENTS FOR NEW OPERATING LICENSES

Summary:

This report summarizes the several parts of the list of TMI-related requirements approved by the Commission for new operating licenses and the time NUREG-0694 was published. Item I.C.7 is identified in this report.

Potential Impact Number: 24722 **SRP Section Number:** 6.3

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

APPENDIX II
Potential Impacts

Position

With the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems. Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or postaccident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

Applicability

This requirement applies to all operating reactors and applicants for an operating license.

Potential Impact Number: 24723 **SRP Section Number:** 5.4.7

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

With the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in

which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems. Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or postaccident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

Applicability

This requirement applies to all operating reactors and applicants for an operating license.

Potential Impact Number: 24726 **SRP Section Number:** 13.2.1

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Licensees are required 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. They must then implement the training program.

Applicability

This requirement applies to all operating reactors and operating license applicants.

Potential Impact Number: 24727 **SRP Section Number:** 5.2.2

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

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Pressurized-water reactor and boiling-water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents.

Applicability

This requirement applies to all operating reactors and operating license applicants.

Potential Impact Number: 24728 **SRP Section Number:** 5.2.2

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Reactor coolant system relief and safety valves shall 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.

Applicability

This requirement applies to all reactor licenses and applicants for operating license.

Potential Impact Number: 24730 **SRP Section Number:** 15.1.5

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system (AFWS), the following requirements shall be implemented in the short term:

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(1) The design shall provide for the automatic initiation of the AFWS.
(2) The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of AFWS function.
(3) Testability of the initiating signals and circuits shall be a feature of the design. (4) The initiating signals and circuits shall be powered from the emergency buses. (5) Manual capability to initiate the AFWS from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.
(6) The ac motor-driven pumps and valves in the AFWS shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.
(7) The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

Applicability

This requirement applies to all PWR operating reactors and applicants for operating license.

Potential Impact Number: 24731 **SRP Section Number:** 15.2.6

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system (AFWS), the following requirements shall be implemented in the short term:

(1) The design shall provide for the automatic initiation of the AFWS.
(2) The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of AFWS function.

(3) Testability of the initiating signals and circuits shall be a feature of the design. (4) The initiating signals and circuits shall be powered from the emergency buses. (5) Manual capability to initiate the AFWS from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.
(6) The ac motor-driven pumps and valves in the AFWS shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.
(7) The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

Applicability

This requirement applies to all PWR operating reactors and applicants for operating license.

Potential Impact Number: 24732 **SRP Section Number:** 15.2.7

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system (AFWS), the following requirements shall be implemented in the short term:

(1) The design shall provide for the automatic initiation of the AFWS.
(2) The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of AFWS function.
(3) Testability of the initiating signals and circuits shall be a feature of the design. (4) The initiating signals and circuits shall be powered from the emergency buses. (5) Manual capability to initiate the AFWS from the control room shall be retained and shall

be implemented so that a single failure in the manual circuits will not result in the loss of system function.
(6) The ac motor-driven pumps and valves in the AFWS shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.
(7) The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

Applicability

This requirement applies to all PWR operating reactors and applicants for operating license.

Potential Impact Number: 24733 **SRP Section Number:** 15.2.8

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system (AFWS), the following requirements shall be implemented in the short term:

(1) The design shall provide for the automatic initiation of the AFWS.
(2) The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of AFWS function.
(3) Testability of the initiating signals and circuits shall be a feature of the design. (4) The initiating signals and circuits shall be powered from the emergency buses. (5) Manual capability to initiate the AFWS from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

(6) The ac motor-driven pumps and valves in the AFWS shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.
(7) The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

Applicability

This requirement applies to all PWR operating reactors and applicants for operating license.

Potential Impact Number: 24734 **SRP Section Number:** 8.3.1

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

(1) The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

(2) Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

(3) The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

(4) Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

Applicability

This requirement applies to all PWR operating reactors and applicants for operating license.

Potential Impact Number: 24744 **SRP Section Number:** 6.2.1.1.A

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Describes additional accident-monitoring instrumentation in the following areas:

(1) Noble gas effluent radiological monitor;
(2) Provisions for continuous sampling of plant effluents for postaccident releases of radioactive iodines and particulates and onsite laboratory capabilities;
(3) Containment high-range radiation monitor;
(4) Containment pressure monitor;
(5) Containment water level monitor; and
(6) Containment hydrogen concentration monitor.

Applicability

This requirement applies to all operating reactors and applicants for operating license.

Potential Impact Number: 24745 **SRP Section Number:** 6.2.1.1.B

Document: NUREG 0737

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Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Describes additional accident-monitoring instrumentation in the following areas:

(1) Noble gas effluent radiological monitor;
(2) Provisions for continuous sampling of plant effluents for postaccident releases of radioactive iodines and particulates and onsite laboratory capabilities;
(3) Containment high-range radiation monitor;
(4) Containment pressure monitor;
(5) Containment water level monitor; and
(6) Containment hydrogen concentration monitor.

Applicability

This requirement applies to all operating reactors and applicants for operating license.

Potential Impact Number: 24746 **SRP Section Number:** 6.2.1.1.C

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Describes additional accident-monitoring instrumentation in the following areas:

(1) Noble gas effluent radiological monitor;
(2) Provisions for continuous sampling of plant effluents for postaccident releases of radioactive iodines and particulates and onsite laboratory capabilities;
(3) Containment high-range radiation monitor;
(4) Containment pressure monitor;
(5) Containment water level monitor; and
(6) Containment hydrogen concentration monitor.

Applicability

This requirement applies to all operating reactors and applicants for operating license.

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Potential Impact Number: 24752 **SRP Section Number:** 8.3.1

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10 CFR Part 50 for the event of loss-of-offsite power, the following positions shall be implemented:

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

(1) Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

(2) Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

(3) Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

(4) The pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

Potential Impact Number: 24756 **SRP Section Number:** 6.3

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

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Summary:

Position

Although the staff believed that the potential for slug flow was not great in Babcock and Wilcox (B&W) plants because of the venting path provided by the internal vent valves, the staff required that a confirmatory evaluation of the effects of slug flow on steam generator tubes be performed by the licensees to assure that the tubes could withstand any mechanical loading which could result from slug flow.

Applicability

This requirement applies to all B&W operating reactors and operating license applicants.

Potential Impact Number: 24758 **SRP Section Number:** 15.1.5

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Evaluate 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.

Applicability

This requirement applies to all B&W operating reactors and operating license applicants.

Potential Impact Number: 24762 **SRP Section Number:** 15.1.5

Document: NUREG 0737

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Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating-current (ac) power for at least 2 hours. Adequacy of the seal design should be demonstrated.

Application

This requirement applies to all BWR, Westinghouse and Combustion Engineering operating reactors and applicants for operating license.

Potential Impact Number: 24765 **SRP Section Number:** 15.0

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Analyze the potential for voiding in the reactor coolant system (RCS) during anticipated transients.

Applicability

This requirement applies to all PWR operating reactors and operating license applicants.

Potential Impact Number: 24766 **SRP Section Number:** 15.2.7

Document: NUREG 0737

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Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Provide a benchmark analysis of sequential auxiliary feedwater (AFW) flow to the steam generators following a loss of main feedwater.

Applicability

This requirement applies to all PWR operating reactors and applicants for operating licenses.

Potential Impact Number: 24771 **SRP Section Number:** 15.1.5

Document: NRC GENERIC LETTER 83-10A
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

We have completed our evaluation of your analyses of LOFT Test L3-6 and conclude that the evaluations acceptably predict the test results. Therefore, we find the currently approved B&W evaluation model for small break LOCAs in continued conformance with Appendix K to 10 CFR 50 for the case of limited RCP operation after reactor trip and for the range of licensed B&W reactor designs.

We have reviewed industry analyses and performed our own analyses to determine whether RCP trip is necessary during LOCAs, and evaluated the desirability of continued RCP operation during non-LOCA transients and accidents, including steam generator tube ruptures. We have 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, some of the calculations show that for certain small break LOCAs, primarily those with only one of the two High Pressure Safety Injection (HPSI) Pumps assumed available, continued operation of the RCPs or continued operation of the RCPs followed by delayed RCP trip could lead to core damage.

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In summary, we have concluded that the need for RCP trip following a transient or accident should be determined by each licensee on a case-by-case basis, considering the Owners Group input. However, the staff must 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 licensees to comply with the Commission's rules and regulations.

We recognize that possible differences exist between the requirements of 10 CFR 50.46, which assure ample core cooling capacity, and the approaches described in the enclosure which are based upon assuring proper operator/ system response under conditions that may be faced during accidents and transients. Accordingly, in such cases, we will consider a request for exemption from specific requirements of 10 CFR 50.46 pursuant to 10 CFR 50.12.

For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual licensees, we strongly urge that licensees work collectively with owners of similar plants (i.e., owners Group) and propose setpoints and methods consistent with other licensees.

However, once the technical bases for the decision are established, we intend to conduct inspections of individual licensees led by Regional personnel. During these inspections, we will examine the translation of the 10 CFR 50, Appendix K, and RCP operation mode evaluations into plant procedures.

Potential Impact Number: 24772 **SRP Section Number:** 15.2.8

Document: NRC GENERIC LETTER 83-10A
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

We have completed our evaluation of your analyses of LOFT Test L3-6 and conclude that the evaluations acceptably predict the test results. Therefore, we find the currently approved B&W evaluation model for small break LOCAs in continued conformance with Appendix K to 10 CFR 50 for the case of limited RCP operation after reactor trip and for the range of licensed B&W reactor designs.

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We have reviewed industry analyses and performed our own analyses to determine whether RCP trip is necessary during LOCAs, and evaluated the desirability of continued RCP operation during non-LOCA transients and accidents, including steam generator tube ruptures. We have 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, some of the calculations show that for certain small break LOCAs, primarily those with only one of the two High Pressure Safety Injection (HPSI) Pumps assumed available, continued operation of the RCPs or continued operation of the RCPs followed by delayed RCP trip could lead to core damage.

In summary, we have concluded that the need for RCP trip following a transient or accident should be determined by each licensee on a case-by-case basis, considering the Owners Group input. However, the staff must 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 licensees to comply with the Commission's rules and regulations.

We recognize that possible differences exist between the requirements of 10 CFR 50.46, which assure ample core cooling capacity, and the approaches described in the enclosure which are based upon assuring proper operator/ system response under conditions that may be faced during accidents and transients. Accordingly, in such cases, we will consider a request for exemption from specific requirements of 10 CFR 50.46 pursuant to 10 CFR 50.12.

For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual licensees, we strongly urge that licensees work collectively with owners of similar plants (i.e., owners Group) and propose setpoints and methods consistent with other licensees.

However, once the technical bases for the decision are established, we intend to conduct inspections of individual licensees led by Regional personnel. During these inspections, we will examine the translation of the 10 CFR 50, Appendix K, and RCP operation mode evaluations into plant procedures.

Potential Impact Number: 24776 **SRP Section Number:** 15.1.5

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

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Summary:

Position

Tripping of the reactor coolant pumps in case of a loss-of-coolant accident (LOCA) is not an ideal solution. Licensees should consider other solutions to the small-break LOCA problem (for example, an increase in safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small-break LOCA. The signals designated to initiate the pump trip are discussed in NUREG-0623.

Potential Impact Number: 24777 **SRP Section Number:** 15.2.8

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Tripping of the reactor coolant pumps in case of a loss-of-coolant accident (LOCA) is not an ideal solution. Licensees should consider other solutions to the small-break LOCA problem (for example, an increase in safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small-break LOCA. The signals designated to initiate the pump trip are discussed in NUREG-0623.

Potential Impact Number: 24781 **SRP Section Number:** 15.1.5

Document: NRC GENERIC LETTER 83-10B
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

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This letter provides guidance for a Combustion Engineering NSSS for the development of either satisfactory setpoints for RCP trip or the technical bases for allowing continued RCP operation in the event of a SBLOCA and other events.

Potential Impact Number: 24782 **SRP Section Number:** 15.2.8

Document: NRC GENERIC LETTER 83-10B
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

This letter provides guidance for a Combustion Engineering NSSS for the development of either satisfactory setpoints for RCP trip or the technical bases for allowing continued RCP operation in the event of a SBLOCA and other events.

Potential Impact Number: 24786 **SRP Section Number:** 15.1.5

Document: NRC GENERIC LETTER 83-10C
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS' (GENERIC LETTER NO. 83-l0c)

Summary:

This letter provides guidance for a Westinghouse NSSS for the development of either satisfactory setpoints for RCP trip or the technical bases for allowing continued RCP operation in the event of a SBLOCA and other events.

Potential Impact Number: 24787 **SRP Section Number:** 15.2.8

Document: NRC GENERIC LETTER 83-10C
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS' (GENERIC LETTER NO. 83-l0c)

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Summary:

This letter provides guidance for a Westinghouse NSSS for the development of either satisfactory setpoints for RCP trip or the technical bases for allowing continued RCP operation in the event of a SBLOCA and other events.

Potential Impact Number: 24791 **SRP Section Number:** 15.1.5

Document: NRC GENERIC LETTER 83-10D
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

We have completed our evaluation of your analyses of LOFT Test L3-6 and conclude that the evaluations acceptably predict the test results. Therefore, we find the currently approved B&W evaluation model for small break LOCAs in continued conformance with Appendix K to 10 CFR 50 for the case of limited RCP operation after reactor trip and for the range of licensed B&W reactor designs.

We have reviewed industry analyses and performed our own analyses to determine whether RCP trip is necessary during LOCAs, and evaluated the desirability of continued RCP operation during non-LOCA transients and accidents, including steam generator tube ruptures. We have 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, some of the calculations show that for certain small break LOCAs, primarily those with only one of the two High Pressure Safety Injection (HPSI) Pumps assumed available, continued operation of the RCPs or continued operation of the RCPs followed by delayed RCP trip could lead to core damage.

In summary, we have concluded that the need for RCP trip following a transient or accident should be determined by each licensee on a case-by-case basis, considering the Owners Group input. However, the staff must 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 licensees to comply with the Commission's rules and regulations.

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We recognize that possible differences exist between the requirements of 10 CFR 50.46, which assure ample core cooling capacity, and the approaches described in the enclosure which are based upon assuring proper operator/ system response under conditions that may be faced during accidents and transients. Accordingly, in such cases, we will consider a request for exemption from specific requirements of 10 CFR 50.46 pursuant to 10 CFR 50.12.

For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual licensees, we strongly urge that licensees work collectively with owners of similar plants (i.e., owners Group) and propose setpoints and methods consistent with other licensees.

However, once the technical bases for the decision are established, we intend to conduct inspections of individual licensees led by Regional personnel. During these inspections, we will examine the translation of the 10 CFR 50, Appendix K, and RCP operation mode evaluations into plant procedures.

Potential Impact Number: 24792 **SRP Section Number:** 15.2.8

Document: NRC GENERIC LETTER 83-10D
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

We have completed our evaluation of your analyses of LOFT Test L3-6 and conclude that the evaluations acceptably predict the test results. Therefore, we find the currently approved B&W evaluation model for small break LOCAs in continued conformance with Appendix K to 10 CFR 50 for the case of limited RCP operation after reactor trip and for the range of licensed B&W reactor designs.

We have reviewed industry analyses and performed our own analyses to determine whether RCP trip is necessary during LOCAs, and evaluated the desirability of continued RCP operation during non-LOCA transients and accidents, including steam generator tube ruptures. We have 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, some of the calculations show that for certain small break LOCAs, primarily those with only one of the two High Pressure Safety Injection (HPSI) Pumps assumed available, continued operation of the

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RCPs or continued operation of the RCPs followed by delayed RCP trip could lead to core damage.

In summary, we have concluded that the need for RCP trip following a transient or accident should be determined by each licensee on a case-by-case basis, considering the Owners Group input. However, the staff must 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 licensees to comply with the Commission's rules and regulations.

We recognize that possible differences exist between the requirements of 10 CFR 50.46, which assure ample core cooling capacity, and the approaches described in the enclosure which are based upon assuring proper operator/ system response under conditions that may be faced during accidents and transients. Accordingly, in such cases, we will consider a request for exemption from specific requirements of 10 CFR 50.46 pursuant to 10 CFR 50.12.

For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual licensees, we strongly urge that licensees work collectively with owners of similar plants (i.e., owners Group) and propose setpoints and methods consistent with other licensees.

However, once the technical bases for the decision are established, we intend to conduct inspections of individual licensees led by Regional personnel. During these inspections, we will examine the translation of the 10 CFR 50, Appendix K, and RCP operation mode evaluations into plant procedures.

Potential Impact Number: 24796 **SRP Section Number:** 15.1.5

Document: NRC GENERIC LETTER 83-10E
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

We have completed our evaluation of your analyses of LOFT Test L3-6 and conclude that the evaluations acceptably predict the test results. Therefore, we find the currently approved B&W evaluation model for small break LOCAs in continued conformance with

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Appendix K to 10 CFR 50 for the case of limited RCP operation after reactor trip and for the range of licensed B&W reactor designs.

We have reviewed industry analyses and performed our own analyses to determine whether RCP trip is necessary during LOCAs, and evaluated the desirability of continued RCP operation during non-LOCA transients and accidents, including steam generator tube ruptures. We have 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, some of the calculations show that for certain small break LOCAs, primarily those with only one of the two High Pressure Safety Injection (HPSI) Pumps assumed available, continued operation of the RCPs or continued operation of the RCPs followed by delayed RCP trip could lead to core damage.

In summary, we have concluded that the need for RCP trip following a transient or accident should be determined by each licensee on a case-by-case basis, considering the Owners Group input. However, the staff must 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 licensees to comply with the Commission's rules and regulations.

We recognize that possible differences exist between the requirements of 10 CFR 50.46, which assure ample core cooling capacity, and the approaches described in the enclosure which are based upon assuring proper operator/ system response under conditions that may be faced during accidents and transients. Accordingly, in such cases, we will consider a request for exemption from specific requirements of 10 CFR 50.46 pursuant to 10 CFR 50.12.

For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual licensees, we strongly urge that licensees work collectively with owners of similar plants (i.e., owners Group) and propose setpoints and methods consistent with other licensees.

However, once the technical bases for the decision are established, we intend to conduct inspections of individual licensees led by Regional personnel. During these inspections, we will examine the translation of the 10 CFR 50, Appendix K, and RCP operation mode evaluations into plant procedures.

Potential Impact Number: 24797 **SRP Section Number:** 15.2.8

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Document: NRC GENERIC LETTER 83-10E
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

We have completed our evaluation of your analyses of LOFT Test L3-6 and conclude that the evaluations acceptably predict the test results. Therefore, we find the currently approved B&W evaluation model for small break LOCAs in continued conformance with Appendix K to 10 CFR 50 for the case of limited RCP operation after reactor trip and for the range of licensed B&W reactor designs.

We have reviewed industry analyses and performed our own analyses to determine whether RCP trip is necessary during LOCAs, and evaluated the desirability of continued RCP operation during non-LOCA transients and accidents, including steam generator tube ruptures. We have 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, some of the calculations show that for certain small break LOCAs, primarily those with only one of the two High Pressure Safety Injection (HPSI) Pumps assumed available, continued operation of the RCPs or continued operation of the RCPs followed by delayed RCP trip could lead to core damage.

In summary, we have concluded that the need for RCP trip following a transient or accident should be determined by each licensee on a case-by-case basis, considering the Owners Group input. However, the staff must 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 licensees to comply with the Commission's rules and regulations.

We recognize that possible differences exist between the requirements of 10 CFR 50.46, which assure ample core cooling capacity, and the approaches described in the enclosure which are based upon assuring proper operator/ system response under conditions that may be faced during accidents and transients. Accordingly, in such cases, we will consider a request for exemption from specific requirements of 10 CFR 50.46 pursuant to 10 CFR 50.12.

For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual licensees, we strongly urge that licensees work collectively with owners of similar plants (i.e., owners Group) and propose setpoints and methods consistent with other licensees.

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However, once the technical bases for the decision are established, we intend to conduct inspections of individual licensees led by Regional personnel. During these inspections, we will examine the translation of the 10 CFR 50, Appendix K, and RCP operation mode evaluations into plant procedures.

Potential Impact Number: 24801 **SRP Section Number:** 15.1.5

Document: NRC GENERIC LETTER 83-10F
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

This letter provides guidance for a Babcock and Wilcox NSSS for the development of either satisfactory setpoints for RCP trip or the technical bases for allowing continued RCP operation in the event of a SBLOCA and other events.

Potential Impact Number: 24802 **SRP Section Number:** 15.2.8

Document: NRC GENERIC LETTER 83-10F
Title: RESOLUTION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

This letter provides guidance for a Babcock and Wilcox NSSS for the development of either satisfactory setpoints for RCP trip or the technical bases for allowing continued RCP operation in the event of a SBLOCA and other events.

Potential Impact Number: 24806 **SRP Section Number:** 15.1.5

Document: NRC GENERIC LETTER 85-12
Title: IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

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Summary:

The purpose of this letter is to inform you of (1) the staff's conclusions regarding the Westinghouse Owners Group (WOG) submittals on reactor coolant pump trip in response to Generic Letters 83-10c and d. and (2) provide guidance concerning implementation of the reactor coolant pump trip criteria. Our Safety Evaluation (SE) on this subject is enclosed for your use.

With regard to the WOG submittals referenced in Section V of the enclosed SE, we conclude that the methods employed by the WOG to justify manual reactor coolant pump (RCP) trip are consistent with the guidelines and criteria provided in Generic Letters 83-10c and d. The approved Westinghouse Small Break LOCA Evaluation Model was used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

We have determined that the Information provided by the WOG in Support of the alternative RCP trip criteria is acceptable on a generic basis. A suitable reactor coolant pump trip criterion can be selected by each licensee to minimize reactor coolant pump trip during steam generator tube ruptures and non-LOCA events, while still providing for RCP trip for small break LOCAs.

In reviewing the WOG RCP trip Criteria we note that the process of criterion selection involves a number of considerations which were assigned plant-specific status by the WOG during the process of the trip criteria review. Accordingly, we request that operating reactor licensees select and implement an appropriate RCP trip criterion based upon the WOG methodology.

Those applicants and licensees who choose not to endorse the WOG methodology should submit a schedule for submittal of plant specific RCP trip criteria or justification for non-trip of RCPs within 45 days of receipt of this letter.

Potential Impact Number: 24807 **SRP Section Number:** 15.2.8

Document: NRC GENERIC LETTER 85-12
Title: IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

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The purpose of this letter is to inform you of (1) the staff's conclusions regarding the Westinghouse Owners Group (WOG) submittals on reactor coolant pump trip in response to Generic Letters 83-10c and d. and (2) provide guidance concerning implementation of the reactor coolant pump trip criteria. Our Safety Evaluation (SE) on this subject is enclosed for your use.

With regard to the WOG submittals referenced in Section V of the enclosed SE, we conclude that the methods employed by the WOG to justify manual reactor coolant pump (RCP) trip are consistent with the guidelines and criteria provided in Generic Letters 83-10c and d. The approved Westinghouse Small Break LOCA Evaluation Model was used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

We have determined that the Information provided by the WOG in Support of the alternative RCP trip criteria is acceptable on a generic basis. A suitable reactor coolant pump trip criterion can be selected by each licensee to minimize reactor coolant pump trip during steam generator tube ruptures and non-LOCA events, while still providing for RCP trip for small break LOCAs.

In reviewing the WOG RCP trip Criteria we note that the process of criterion selection involves a number of considerations which were assigned plant-specific status by the WOG during the process of the trip criteria review. Accordingly, we request that operating reactor licensees select and implement an appropriate RCP trip criterion based upon the WOG methodology.

Those applicants and licensees who choose not to endorse the WOG methodology should submit a schedule for submittal of plant specific RCP trip criteria or justification for non-trip of RCPs within 45 days of receipt of this letter.

Potential Impact Number: 24811 **SRP Section Number:** 15.1.5

Document: NRC GENERIC LETTER 86-05
Title: IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

The purpose of this letter is to inform you of (1) the staff's conclusions regarding the B&W Owners Group (BWOG) submittals on reactor coolant pump trip in response to

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Generic Letters 83-10e and f, and (2) provide guidance concerning implementation of the reactor coolant pump trip criterion. Our Safety Evaluation (SE) on this subject is enclosed for your use.

With regard to the BWOG submittals referenced in Section V of the enclosed SE, we conclude that the methods employed by the BWOG to justify manual reactor coolant pump (RCP) trip are consistent with the guidelines and criteria provided in Generic Letters 83-10e and f. The approved B&W Small Break LOCA Evaluation Model was used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

We have determined that the information provided by the BWOG in support of the loss-of-subcooling RCP trip criterion is acceptable. The generic information presented by the BWOG, however, does not address plant specific concerns about instrumentation uncertainties, potential reactor coolant pump problems and operator training and procedures as requested in Generic Letter 83-10. This information, contained in Section IV of the SE, is now being requested to assess implementation of the RCP trip criterion. Accordingly, for those applicants and licensees who choose to endorse the BWOG methodology, we request that operating reactor licensees implement the RCP trip criterion based upon the BWOG methodology.

Those applicants and licensees who choose not to endorse the BWOG methodology should submit a schedule for submittal of plant specific RCP trip criteria or justification for non-trip of RCPs within 45 days of receipt of this letter.

Potential Impact Number: 24812 **SRP Section Number:** 15.2.8

Document: NRC GENERIC LETTER 86-05
Title: IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

The purpose of this letter is to inform you of (1) the staff's conclusions regarding the B&W Owners Group (BWOG) submittals on reactor coolant pump trip in response to Generic Letters 83-10e and f, and (2) provide guidance concerning implementation of the reactor coolant pump trip criterion. Our Safety Evaluation (SE) on this subject is enclosed for your use.

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With regard to the BWOG submittals referenced in Section V of the enclosed SE, we conclude that the methods employed by the BWOG to justify manual reactor coolant pump (RCP) trip are consistent with the guidelines and criteria provided in Generic Letters 83-10e and f. The approved B&W Small Break LOCA Evaluation Model was used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

We have determined that the information provided by the BWOG in support of the loss-of-subcooling RCP trip criterion is acceptable. The generic information presented by the BWOG, however, does not address plant specific concerns about instrumentation uncertainties, potential reactor coolant pump problems and operator training and procedures as requested in Generic Letter 83-10. This information, contained in Section IV of the SE, is now being requested to assess implementation of the RCP trip criterion. Accordingly, for those applicants and licensees who choose to endorse the BWOG methodology, we request that operating reactor licensees implement the RCP trip criterion based upon the BWOG methodology.

Those applicants and licensees who choose not to endorse the BWOG methodology should submit a schedule for submittal of plant specific RCP trip criteria or justification for non-trip of RCPs within 45 days of receipt of this letter.

Potential Impact Number: 24816 **SRP Section Number:** 15.1.5

Document: NRC GENERIC LETTER 86-06
Title: IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

The purpose of this letter is to inform you of (l) the staff's conclusions regarding the CE Owners Group (CEOG) submittals on reactor coolant pump trip in response to Generic Letters 83-10a and b, and (2) provide guidance concerning implementation of the reactor coolant pump trip criterion. Our Safety Evaluation (SE) on this subject is enclosed for your use

With regard to the CEOG submittals referenced in Section V of the enclosed SE, we conclude that the methods employed by the CEOG to justify manual reactor coolant pump (RCP) trip are consistent with the guidelines and criteria provided in Generic

Letters 83-10a and b. The approved CE Small Break LOCA Evaluation Model was used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

We have determined that the information provided by the CEOG in support of the trip-two/leave-two staggered reactor coolant pump trip criterion is acceptable. The generic information presented by the CEOG, however, does not address plant specific concerns about instrumentation uncertainties, potential reactor coolant pump problems and operator training and procedures as requested in Generic Letter 83-10. This information, contained in Section IV of the SE, is now being requested to assess implementation of the RCP trip criterion. Accordingly, for those applicants and licensees who choose to endorse the CEOG methodology, we request that operating reactor licensees implement the RCP trip criterion based upon the CEOG methodology.

Those applicants and licensees who choose not to endorse the CEOG methodology should submit a schedule for submittal of plant specific RCP trip criteria or justification for non-trip of RCPs within 45 days of receipt of this letter.

Potential Impact Number: 24817 **SRP Section Number:** 15.2.8

Document: NRC GENERIC LETTER 86-06
Title: IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5, 'AUTOMATIC TRIP OF REACTOR COOLANT PUMPS'

Summary:

The purpose of this letter is to inform you of (l) the staff's conclusions regarding the CE Owners Group (CEOG) submittals on reactor coolant pump trip in response to Generic Letters 83-10a and b, and (2) provide guidance concerning implementation of the reactor coolant pump trip criterion. Our Safety Evaluation (SE) on this subject is enclosed for your use

With regard to the CEOG submittals referenced in Section V of the enclosed SE, we conclude that the methods employed by the CEOG to justify manual reactor coolant pump (RCP) trip are consistent with the guidelines and criteria provided in Generic Letters 83-10a and b. The approved CE Small Break LOCA Evaluation Model was used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

We have determined that the information provided by the CEOG in support of the trip-two/leave-two staggered reactor coolant pump trip criterion is acceptable. The

generic information presented by the CEOG, however, does not address plant specific concerns about instrumentation uncertainties, potential reactor coolant pump problems and operator training and procedures as requested in Generic Letter 83-10. This information, contained in Section IV of the SE, is now being requested to assess implementation of the RCP trip criterion. Accordingly, for those applicants and licensees who choose to endorse the CEOG methodology, we request that operating reactor licensees implement the RCP trip criterion based upon the CEOG methodology.

Those applicants and licensees who choose not to endorse the CEOG methodology should submit a schedule for submittal of plant specific RCP trip criteria or justification for non-trip of RCPs within 45 days of receipt of this letter.

Potential Impact Number: 24821 **SRP Section Number:** 15.0

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Most overpressure transients should not result in the opening of the power-operated relief valve (PORV). Therefore, licensees should document that the PORV will open in less than 5% of all anticipated overpressure transients using the revised setpoints and anticipatory trips for the range of plant conditions which might occur during a fuel cycle.

Applicability

This requirement applies to all B&W operating reactors.

Potential Impact Number: 24822 **SRP Section Number:** 6.3

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

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Position

The anticipatory trip modification proposed by some licensees to confine the range of use to high-power levels 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.

Applicability

This requirement applies to selected Westinghouse operating reactors and operating license applicants.

Potential Impact Number: 24823 **SRP Section Number:** 5.4.6

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Currently, the reactor core isolation cooling (RCIC) system and the high-pressure coolant injection (HPCI) system both initiate on the same low-water-level signal and both isolate on the same high-water-level signal. The HPCI system will restart on low water level but the RCIC system will not. The RCIC system is a low-flow system when compared to the HPCI system. The 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. These changes have the potential to reduce the number of challenges to the HPCI system and could result in less stress on the vessel from cold water injection. Analyses should be performed to evaluate these changes. The analyses should be submitted to the NRC staff and changes should be implemented if justified by the analyses.

Applicability

This requirement applies to all operating BWRs and operating license applicants with RCIC and HPCI systems.

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Potential Impact Number: 24825 **SRP Section Number:** 6.3

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

The automatic depressurization system (ADS) actuation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment study is required to determine the optimum approach. One possible scheme that should be considered is ADS actuation on low reactor-vessel water level provided no high-pressure coolant injection (HPCI) or high-pressure coolant system (HPCS) flow exists and a low-pressure emergency core cooling (ECC) system is running. This logic would complement, not replace, the existing ADS actuation logic. Changes to Previous Requirements and Guidance The schedule has been changed to accommodate the vendor-projected completion date and staff review of a very complex change. Clarification

Applicability

This requirements applies to all operating boiling-water reactors (BWRs) and BWR applicants for operating license.

Potential Impact Number: 24826 **SRP Section Number:** 6.3

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

The core-spray and low-pressure, coolant-injection (LPCI) system flow may be stopped by the operator. These systems will not restart automatically on loss of water level if an initiation signal is still present. The core spray and LPCI system logic should be modified so that these systems will restart, if required, to assure adequate core cooling. Because this design modification affects several core-cooling modes under accident

conditions, a preliminary design should be submitted for staff review and approval prior to making the actual modification.

Clarification

Modification of system design should be made in accordance with those requirements set forth in Sections 4.12, 4.13, and 4.16 of IEEE Standard 279-1971 with regard to protective function bypasses and completion of protective action once initiated.

Applicability

This requirement applies to all BWR operating reactors and applicants for BWR operating license.

Potential Impact Number: 24827 **SRP Section Number:** 6.3

Document: NRC BULLETIN 80-01
Title: OPERABILITY OF ADS VALVE PNEUMATIC SUPPLY

Summary:

The bulletin describes a concern that the Automatic Depressurization System (ADS) pneumatic supply (either nitrogen or air) may not be operable for all possible events because of a combination of misapplication of check valves, a lack of leak testing of the accumulator system backing up each ADS valve operator and questions about the continued operability of the pneumatic supply in a seismic event.

The bulletin describes actions to be taken by licensees of GE BWR facilities with an operating license which use a pneumatic operator for the ADS function.

Potential Impact Number: 24828 **SRP Section Number:** 15.0

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

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Position

For anticipated transients combined with the worst single failure and assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which result from a stuck-open relief valve should be included in this category. Changes to Previous Requirements and Guidance There are no changes to the previous requirements.

Applicability

This requirement applies to all operating boiling-water reactors (BWRs) and BWR license applicants.

Potential Impact Number: 24829 **SRP Section Number:** 6.3

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction (a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment. (b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Potential Impact Number: 24830 **SRP Section Number:** 5.4.6

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Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction (a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment. (b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Potential Impact Number: 24831 **SRP Section Number:** 5.4.7

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction (a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment. (b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Potential Impact Number: 24833 **SRP Section Number:** 9.3.4

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction (a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment. (b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Potential Impact Number: 24836 **SRP Section Number:** 6.4

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

In accordance with Task Action Plan item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against

the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, 'Control Room,' of Appendix A, 'General Design Criteria for Nuclear Power Plants,' to 10 CFR Part 50).

Potential Impact Number: 24839 **SRP Section Number:** 17.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(3)(iii)

Summary:

Establish a quality assurance (QA) program based on consideration of:

(A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions;
(B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent;
(C) including QA personnel in the documented review of and concurrence in quality related
procedures associated with design, construction and installation;
(D) establishing criteria for determining QA programmatic requirements;
(E) establishing qualification requirements for QA and QC personnel;
(F) sizing the QA staff commensurate with its duties and responsibilities;
(G) establishing procedures for maintenance of "as-built" documentation; and
(H) providing a QA role in design and analysis activities.

Potential Impact Number: 24841 **SRP Section Number:** 17.1

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

The ABWR FSER has the following finding regarding the requirements of 10 CFR 50.34(f)(3)(iii):

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NUREG-0660, Item I.F.2 lists 11 considerations that the NRC should use to develop additional guidance to clarify requirements for the QA function. These considerations resulted in Revision 2 of SRP Section 17.1, 'Quality Assurance During the Design and Construction Phases,' in July 1981. SSAR Chapter 17 references NEDO-11209, "Nuclear Energy Business Operation Quality Assurance Program Description," Revision 7 dated May 1987. The staff reviewed this report against the acceptance criteria of Revision 2 of SRP Section 17.1 and found it to be acceptable. The staff concludes that GE's approach adequately addresses the requirements of this TMI item for the ABWR design.

Potential Impact Number: 24843 **SRP Section Number:** 17.1

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

The following is a summary of the finding in the CE 80+ FSER regarding compliance with I.F.2 and 10 CFR 52.47(a)(1)(iv):

The subissues for Issue I.F.2 that must be addressed for 10 CFR 52.47(a)(1)(iv) are the following: Item 2, include QA personnel in review and approval of plant procedures; Item 3, include QA personnel in all design, construction, installation, testing, and operation activities; Item 6, increase the size of the QA staff; and Item 9, clarify organizational reporting levels for the QA organization. The new requirements were incorporated into the SRP on QA.

Potential Impact Number: 24845 **SRP Section Number:** 17.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This portion of NUREG 0933 describes the current status of I.F.2. The overall objective of this issue is 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.

The discussion divides the issue into 11 elements, based on the NUREG 0660 identification of this issue. The discussion indicates that from the standpoint of establishing criteria for QA programs, the issue is resolved with the July 1981 revision to Chapter 17. The discussion indicates that there is an issue remaining regard management support of QA, however this portion of the issue was designated a low priority. Items I.F.2(2), I.F.2(3), I.F.2(6) and I.F.2(9) were subsequently designated as resolved. The completion of the management aspect of the remaining open issues currently has a low priority.

Potential Impact Number: 24850 **SRP Section Number:** 17.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This TMI Action Plan item 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 and coordinated with those made under TMI Action Plan Item I.E.6. The reports received as a result of these rule changes will provide increased information on component failures that affect safety so that prompt and effective corrective action can be taken. The information will also be used as input to an augmented role of the NRC's vendor and construction inspection program.

The issue was resolved with new requirements when amendments to 10 CFR 21 and 10 CFR 50.55(e) were issued. The staff's changes were presented to the Commission in SECY-91-150.

Potential Impact Number: 24851 **SRP Section Number:** 17.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

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Summary:

This TMI Action Plan item 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 and coordinated with those made under TMI Action Plan Item I.E.6. The reports received as a result of these rule changes will provide increased information on component failures that affect safety so that prompt and effective corrective action can be taken. The information will also be used as input to an augmented role of the NRC's vendor and construction inspection program.

The issue was resolved with new requirements when amendments to 10 CFR 21 and 10 CFR 50.55(e) were issued. The staff's changes were presented to the Commission in SECY-91-150.

Potential Impact Number: 24852 **SRP Section Number:** 17.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

The original 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. This was reflected in 10 CFR 50.34 (f)(3)(ii) which states: '...ensure that the Quality Assurance (QA) List required by Criteria II App. B, 10 CFR Part 50 includes all structures, systems and components important to safety (I.F.1)...' However, the staff's IREP Procedures Guide failed to identify either the need for a QA list for structures, systems, and components important to safety (ITS) or the basis for a generic list even if one should be needed In January 1984, Generic Letter 84-01 was issued to clarify NRC use of the terms, 'Important to Safety' and 'Safety Related.' This letter summarized NRC's intention to pursue QA requirements for important to safety equipment on a case-by-case basis. Further clarification was provided in the Commission's The first proposed rule on 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 and the issue was not addressed in the second staff submittal on the ITS rule in SECY-86-164.

Therefore, this issue was resolved and no new requirements were established.

Potential Impact Number: 24853 **SRP Section Number:** 17.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

The original 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. This was reflected in 10 CFR 50.34 (f)(3)(ii) which states: '...ensure that the Quality Assurance (QA) List required by Criteria II App. B, 10 CFR Part 50 includes all structures, systems and components important to safety (I.F.1)...' However, the staff's IREP Procedures Guide failed to identify either the need for a QA list for structures, systems, and components important to safety (ITS) or the basis for a generic list even if one should be needed In January 1984, Generic Letter 84-01 was issued to clarify NRC use of the terms, 'Important to Safety' and 'Safety Related.' This letter summarized NRC's intention to pursue QA requirements for important to safety equipment on a case-by-case basis. Further clarification was provided in the Commission's The first proposed rule on 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 and the issue was not addressed in the second staff submittal on the ITS rule in SECY-86-164.

Therefore, this issue was resolved and no new requirements were established.

Potential Impact Number: 24855 **SRP Section Number:** 17.1

Document: NRC GENERIC LETTER 84-01
Title: NRC Use of the Terms, 'Important to Safety' and 'Safety Related'

Summary:

While previous staff licensing reviews were not specifically directed towards determining whether, in fact permittees, or licensees have developed quality assurance

programs which adequately address all structures, systems and components important to safety, this was not because of any concern over the lack of regulatory requirements for this class of equipment. Rather, our practice was based upon the staff view that normal industry practice is generally acceptable for most equipment not covered by Appendix B within this class. Nevertheless, in specific situations in the past where we have found that quality assurance requirements beyond normal industry practice were needed for equipment 'important to safety,' we have not hesitated in imposing additional requirements commensurate with the importance to safety of the equipment involved. We intend to continue that practice.

Potential Impact Number: 24856 **SRP Section Number:** 17.3

Document: NRC GENERIC LETTER 84-01
Title: NRC Use of the Terms, 'Important to Safety' and 'Safety Related'

Summary:

While previous staff licensing reviews were not specifically directed towards determining whether, in fact permittees, or licensees have developed quality assurance programs which adequately address all structures, systems and components important to safety, this was not because of any concern over the lack of regulatory requirements for this class of equipment. Rather, our practice was based upon the staff view that normal industry practice is generally acceptable for most equipment not covered by Appendix B within this class. Nevertheless, in specific situations in the past where we have found that quality assurance requirements beyond normal industry practice were needed for equipment 'important to safety,' we have not hesitated in imposing additional requirements commensurate with the importance to safety of the equipment involved. We intend to continue that practice.

Potential Impact Number: 24858 **SRP Section Number:** 17.1

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

APPENDIX II
Potential Impacts

NUREG-0660 states that several systems important to safety of the TMI plant were not designed, fabricated, and maintained at a level equivalent to their safety importance. In accordance with the requirements of Criterion 1 of Appendix A to 10 CFR Part 50 and the guidance provided in SRP Section 17.3, 'Quality Assurance Program Description,' applicants are to include these types of non-safety-related items within the scope of their QA programs. This requirement was incorporated into Paragraph (3)(ii) of 10 CFR 50.34(f).

SSAR Table 3.2-1, 'Classification Summary,' identifies safety-related and non-safety-related items of the ABWR. Note 'e' in the table and SSAR Section 3.1.2.1.1.2 state

(1) the total QA program described in SSAR Chapter 17 is applied to the safety-related items

(2) the non-safety-related items are controlled by the QA program described in SSAR Chapter 17 in accordance with the functional importance of the item

The staff concludes that GE's approach adequately addresses the requirements of this TMI item for the ABWR design.

Potential Impact Number: 24859 **SRP Section Number:** 17.3

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

NUREG-0660 states that several systems important to safety of the TMI plant were not designed, fabricated, and maintained at a level equivalent to their safety importance. In accordance with the requirements of Criterion 1 of Appendix A to 10 CFR Part 50 and the guidance provided in SRP Section 17.3, 'Quality Assurance Program Description,' applicants are to include these types of non-safety-related items within the scope of their QA programs. This requirement was incorporated into Paragraph (3)(ii) of 10 CFR 50.34(f).

SSAR Table 3.2-1, 'Classification Summary,' identifies safety-related and non-safety-related items of the ABWR. Note 'e' in the table and SSAR Section 3.1.2.1.1.2 state

(1) the total QA program described in SSAR Chapter 17 is applied to the safety-related items

(2) the non-safety-related items are controlled by the QA program described in SSAR Chapter 17 in accordance with the functional importance of the item

The staff concludes that GE's approach adequately addresses the requirements of this TMI item for the ABWR design.

Potential Impact Number: 24861 **SRP Section Number:** 17.1

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

Issue I.F.1, in NUREG-0933, addressed improving the QA program for the design, construction, and operation of nuclear power plants. The licensees were to identify those SSCs that were not labeled safety-related but were important to safety, to prioritize their importance to safety, and to prepare a generic QA list. In GDC 1, the NRC requires that SSCs important to safety should be designed, built, and tested commensurate to their importance to safety. In January 1984, the staff issued GL 84-01, "NRC Use of the Terms, 'Important to Safety' and 'Safety-Related'," dated January 5, 1984, to clarify the use of the terms "important to safety" and "safety-related."

ABB-CE states in CESSAR-DC Section 20.2.98 that the classification of SSCs for the System 80+ design is provided in CESSAR-DC Table 3.2-1. The staff evaluates this table in Section 3.2 of this report and concludes that it conforms to GDC 1.

Therefore, Issue I.F.1 is resolved for the System 80+ design.

Potential Impact Number: 24862 **SRP Section Number:** 17.3

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

APPENDIX II
Potential Impacts

Summary:

Issue I.F.1, in NUREG-0933, addressed improving the QA program for the design, construction, and operation of nuclear power plants. The licensees were to identify those SSCs that were not labeled safety-related but were important to safety, to prioritize their importance to safety, and to prepare a generic QA list. In GDC 1, the NRC requires that SSCs important to safety should be designed, built, and tested commensurate to their importance to safety. In January 1984, the staff issued GL 84-01, "NRC Use of the Terms, 'Important to Safety' and 'Safety-Related'," dated January 5, 1984, to clarify the use of the terms "important to safety" and "safety-related."

ABB-CE states in CESSAR-DC Section 20.2.98 that the classification of SSCs for the System 80+ design is provided in CESSAR-DC Table 3.2-1. The staff evaluates this table in Section 3.2 of this report and concludes that it conforms to GDC 1.

Therefore, Issue I.F.1 is resolved for the System 80+ design.

Potential Impact Number: 24864 **SRP Section Number:** 17.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(3)(ii)

Summary:

Ensure that the quality assurance (QA) list required by Criterion II, App. B, 10 CFR Part 50 includes all structures, systems, and components important to safety.

Potential Impact Number: 24865 **SRP Section Number:** 17.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(3)(ii)

Summary:

APPENDIX II
Potential Impacts

Ensure that the quality assurance (QA) list required by Criterion II, App. B, 10 CFR Part 50 includes all structures, systems, and components important to safety.

Potential Impact Number: 24867 **SRP Section Number:** 5.4.7

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY 93-087 the staff recommended that the Commission approve positions related to probabilistic risk assessments and analysis of external events. In the SRM for SECY 93-087 the Commission approved modified versions of the positions proposed by the staff. The approved positions are as follows:

- The analyses submitted in accordance with 10 CFR 52.47 should include an assessment of internal and external events. PRA insights will be used to support a margins-type assessment of seismic events. A PRA-based seismic margins analysis will consider sequence-level High Confidence, Low Probability of Failures (HCLPFs) and fragilities for all sequences leading to core damage or containment failures up to approximately one and two- thirds the ground motion acceleration of the Design Basis SSE. Simplified probabilistic methods, such as but not limited to EPRI's FIVE methodology, will be used to evaluate fires. Traditional probabilistic techniques should be used to evaluate internal floods.

- ALWR vendors should perform bounding analysis of site-specific external events likely to be a challenge to the plant (such as river flooding, storm surge, tsunami, volcanism, high winds, and hurricanes). When a site is chosen, its characteristics should be compared to those assumed in the bounding analysis to ensure that the site is enveloped. If the site is enveloped, the COL applicant need not perform further PRA evaluations for these external events. The COL applicant should perform site-specific PRA evaluations to address any site-specific hazards for which a bounding analysis was not performed or which are not enveloped by the bounding analyses to ensure that no vulnerabilities due to siting exist.

APPENDIX II
Potential Impacts

The issues addressed by this potential impact are related to PRA evaluations not currently reviewed in the SRP. These issues should be considered as candidates for future work as part of an IPD 7.0.

Potential Impact Number: 24869 **SRP Section Number:** 6.3

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY 93-087 the staff recommended that the Commission approve positions related to probabilistic risk assessments and analysis of external events. In the SRM for SECY 93-087 the Commission approved modified versions of the positions proposed by the staff. The approved positions are as follows:

- The analyses submitted in accordance with 10 CFR 52.47 should include an assessment of internal and external events. PRA insights will be used to support a margins-type assessment of seismic events. A PRA-based seismic margins analysis will consider sequence-level High Confidence, Low Probability of Failures (HCLPFs) and fragilities for all sequences leading to core damage or containment failures up to approximately one and two- thirds the ground motion acceleration of the Design Basis SSE. Simplified probabilistic methods, such as but not limited to EPRI's FIVE methodology, will be used to evaluate fires. Traditional probabilistic techniques should be used to evaluate internal floods.

- ALWR vendors should perform bounding analysis of site-specific external events likely to be a challenge to the plant (such as river flooding, storm surge, tsunami, volcanism, high winds, and hurricanes). When a site is chosen, its characteristics should be compared to those assumed in the bounding analysis to ensure that the site is enveloped. If the site is enveloped, the COL applicant need not perform further PRA evaluations for these external events. The COL applicant should perform site-specific PRA evaluations to address any site-specific hazards for which a bounding analysis was not performed or which are not enveloped by the bounding analyses to ensure that no vulnerabilities due to siting exist.

APPENDIX II
Potential Impacts

The issues addressed by this potential impact are related to PRA evaluations not currently reviewed in the SRP. These issues should be considered as candidates for future work as part of an IPD 7.0.

Potential Impact Number: 24870 **SRP Section Number:** 5.4.7

Document: NRC GENERIC LETTER 88-20
Title: INDIVIDUAL PLANT EXAMINATION FOR SEVERE ACCIDENT VULNERABILITIES

Summary:

In the Commission policy statement on severe accidents in nuclear power plants issued on August 8, 1985 (50 FR 32138), the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on generic rulemaking or other regulatory requirements for these plants. However, the Commission recognizes, based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low cost improvements. Therefore, each existing plant should perform a systematic examination to identify any plant-Specific vulnerabilities to severe accidents and report the results to the Commission.

Therefore, consistent with the stated position of the Commission and pursuant to 10 CFR 50.54(f), you are requested to perform an Individual Plant Examination of your plant(s) for severe accident vulnerabilities and submit the results to the NRC.

Potential Impact Number: 24872 **SRP Section Number:** 6.3

Document: NRC GENERIC LETTER 88-20
Title: INDIVIDUAL PLANT EXAMINATION FOR SEVERE ACCIDENT VULNERABILITIES

Summary:

APPENDIX II
Potential Impacts

In the Commission policy statement on severe accidents in nuclear power plants issued on August 8, 1985 (50 FR 32138), the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on generic rulemaking or other regulatory requirements for these plants. However, the Commission recognizes, based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low cost improvements. Therefore, each existing plant should perform a systematic examination to identify any plant-Specific vulnerabilities to severe accidents and report the results to the Commission.

Therefore, consistent with the stated position of the Commission and pursuant to 10 CFR 50.54(f), you are requested to perform an Individual Plant Examination of your plant(s) for severe accident vulnerabilities and submit the results to the NRC.

Potential Impact Number: 24873 **SRP Section Number:** 5.4.7

Document: NRC POLICY STATEMENT 50 FR 32138
Title: POLICY STATEMENT ON SEVERE REACTOR ACCIDENTS REGARDING FUTURE DESIGNS AND EXISTING PLANTS

Summary:

This statement describes the policy the Commission intends to use to resolve safety issues related to reactor accidents more severe than design basis accidents. Its main focus is on the criteria and procedures the Commission intends to use to certify new designs for nuclear power plants.

One of the main purposes of this Policy Statement was to re-examine the need for the generic rulemaking proceeding contemplated in the TMI Action Plan commitment (NUREG-0660, Task II,B.8) on degraded core accidents, currently referred to as severe nuclear reactor accidents.

Potential Impact Number: 24875 **SRP Section Number:** 6.3

Document: NRC POLICY STATEMENT 50 FR 32138

APPENDIX II
Potential Impacts

Title: POLICY STATEMENT ON SEVERE REACTOR ACCIDENTS REGARDING FUTURE DESIGNS AND EXISTING PLANTS

Summary:

This statement describes the policy the Commission intends to use to resolve safety issues related to reactor accidents more severe than design basis accidents. Its main focus is on the criteria and procedures the Commission intends to use to certify new designs for nuclear power plants.

One of the main purposes of this Policy Statement was to re-examine the need for the generic rulemaking proceeding contemplated in the TMI Action Plan commitment (NUREG-0660, Task II,B.8) on degraded core accidents, currently referred to as severe nuclear reactor accidents.

Potential Impact Number: 24876 **SRP Section Number:** 5.4.7

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This section of NUREG 0933 describes the background related to this issue including the degraded core rulemaking referred to in the SRP. The discussion indicates that the specific proposed rule, and it's derivatives published in the Federal Register are no longer being pursued.

The following summarizes the conclusion of this NUREG 0933 item:

Based on the above evaluation, this item was given a high priority ranking. Work performed by RES on the hydrogen control aspect of the issue resulted in Revision 3 a Hydrogen Control Rule that was approved by the Commission and published in the Federal Register on January 25, 1985. The severe accident portion of the issue was addressed in April 1983 by a Policy Statement that set forth the Commission's intentions for rulemakings and other regulatory actions for resolving safety issues related to reactor accidents more severe than design basis accidents (48 FR 16014). Certain severe accident technical issues identified under the discussion of long-term rulemaking were to be dealt with for future and existing plants through procedures and ongoing severe accident programs identified in the Policy Statement and described

more fully in Chapter IV of NUREG-1070.Thus, with the issuance of the rule on hydrogen control, this item was resolved and new requirements were established.

Potential Impact Number: 24878 **SRP Section Number:** 6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This section of NUREG 0933 describes the background related to this issue including the degraded core rulemaking referred to in the SRP. The discussion indicates that the specific proposed rule, and it's derivatives published in the Federal Register are no longer being pursued.

The following summarizes the conclusion of this NUREG 0933 item:

Based on the above evaluation, this item was given a high priority ranking. Work performed by RES on the hydrogen control aspect of the issue resulted in Revision 3 a Hydrogen Control Rule that was approved by the Commission and published in the Federal Register on January 25, 1985. The severe accident portion of the issue was addressed in April 1983 by a Policy Statement that set forth the Commission's intentions for rulemakings and other regulatory actions for resolving safety issues related to reactor accidents more severe than design basis accidents (48 FR 16014). Certain severe accident technical issues identified under the discussion of long-term rulemaking were to be dealt with for future and existing plants through procedures and ongoing severe accident programs identified in the Policy Statement and described more fully in Chapter IV of NUREG-1070.Thus, with the issuance of the rule on hydrogen control, this item was resolved and new requirements were established.

Potential Impact Number: 24880 **SRP Section Number:** 6.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(vii)

Summary:

APPENDIX II
Potential Impacts

Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID 14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment.

Potential Impact Number: 24881 **SRP Section Number:** 5.4.7

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(vii)

Summary:

Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID 14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment.

Potential Impact Number: 24886 **SRP Section Number:** 13.2.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This item required the staff to develop criteria and procedures to be used in auditing training programs, including those provided by reactor vendors, and to increase the amount of auditing. Specifically, NRR was to: (1) audit training programs to assure training is formalized and eventually in conformance with accreditation; (2) conduct cold operator licensing certification at simulators; and (3) pending accreditation, require certain instructors to be SRO-certified. Elements (2) and (3) were implemented and have been incorporated into the Examiner Standards and Inspection Procedures. The issue of training audits was addressed by the Commission's Policy Statement on Training and Qualification of Nuclear Power Plant Personnel which endorsed the INPO-managed accreditation program.

This item was clarified in NUREG-0737 and new requirements were established.

APPENDIX II
Potential Impacts

Potential Impact Number: 24888 **SRP Section Number:** 13.2.1

Document: NRC GENERIC LETTER 81-29
Title: Simulator Examinations

Summary:

The letter discusses the implementation of Item I.A.3.1 of NUREG-0737. The letter also provides some additional clarifications related to I.A.3.1 as follows:

1. Simulator exams will generally consist of two applicants and two examiners. One applicant will assume a supervisory role. The exam will last 2-3 hours and involve at-power manipulations and concurrent malfunctions.

2. Plants without simulators may be granted waivers of the two-week allowance in scheduling simulator and plant examinations on a case-by- case basis, providing sufficient justification exists.

3. Applicants for a senior license, who hold a valid operator's license, will be required to take a simulator exam.

4. Simulators used for an examination must be NSSS specific.

Potential Impact Number: 24889 **SRP Section Number:** 13.2.1

Document: NRC ADMIN LETTER 94-05
Title: NOTIFICATION CONCERNING CHANGES TO 10 CFR PART 55

Summary:

This administrative letter was issued to inform licensed operators and holders of operating licenses or construction permits of the implementation of the amendments to the regulations in Title 10, Code of Federal Regulations, Part 55 (10 CFR Part 55) concerning renewal of operator licenses.

Section 306 of the Nuclear Waste Policy Act of 1982 directed the NRC to promulgate regulations or other appropriate guidance to establish "simulator training

requirements...and...requirements governing NRC administration of requalification examinations." On May 26, 1987, the NRC amended 10 CFR Part 55 to require each licensed operator to pass a comprehensive written requalification examination and an operating test conducted by the NRC during the term of the operator's 6-year license as a prerequisite for license renewal.

Potential Impact Number: 24890 **SRP Section Number:** 13.2.1

Document: NRC GENERIC LETTER 90-08
Title: SIMULATION FACILITY EXEMPTIONS

Summary:

As stated in the "Supplementary Information" published with the final version of 10 CFR Part 55 in the Federal Register (52 FR 9453, March 25, 1987), the NRC expected each facility licensee to have a simulation facility available within a reasonable period of time after the effective date of the new rule but not later than 4 years after the effective date. It was further stated that any deviations from the time requirements of 10 CFR 55.45(b) would be addressed on a case-specific basis. Therefore, facility licensees who believe they will not be able to meet the time requirements should request exemptions as soon as possible so that the request may be reviewed and, where appropriate, granted so that the facility licensees will not be in violation of the regulations discussed above.

The NRC will consider requests for exemptions in accordance with 10 CFR 50.12. Items that will be considered when reviewing justifications for exemption requests include the licensees' reasons for not meeting the schedule requirements including their actual order dates and estimated completion dates for new or upgraded simulation facilities and the means for conducting operating tests until a certified or approved simulation facility is available. Special circumstances, as described in 10 CFR 50.12(a)(2), must be present for any exemption to be granted.

Potential Impact Number: 24891 **SRP Section Number:** 13.2.1

Document: NRC NOTICE 93-24
Title: Distribution of Revision 7 of NUREG-1021, "Operator Licensing Examiner Standards"

APPENDIX II
Potential Impacts

Summary:

This information notice to inform all holders of operator and senior operator licenses at nuclear power reactors that the NRC issued Revision 7 of NUREG-1021, "Operator Licensing Examiner Standards." Revision 7 of NUREG-1021 contains changes to the processes for licensing operators and conducting NRC initial licensing and requalification examinations.

NUREG-1021 provides detailed discussions of the use of simulators in operator training, qualification, and licensing activities.

Potential Impact Number: 24892 **SRP Section Number:** 13.2.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Summary:

Establishes requirements for operator qualifications including requirements related to simulators.

Potential Impact Number: 24893 **SRP Section Number:** 13.2.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(i)

Summary:

Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCA's.

Potential Impact Number: 24905 **SRP Section Number:** 5.2.2

APPENDIX II
Potential Impacts

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(x)

Summary:

Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed.

Potential Impact Number: 24906 **SRP Section Number:** 5.2.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xi)

Summary:

Provide direct indication of relief and safety valve position (open or closed) in the control room.

Potential Impact Number: 24908 **SRP Section Number:** 15.1.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xii)

Summary:

Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room.

APPENDIX II
Potential Impacts

Potential Impact Number: 24909 **SRP Section Number:** 15.2.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xii)

Summary:

Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room.

Potential Impact Number: 24910 **SRP Section Number:** 15.2.7

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xii)

Summary:

Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room.

Potential Impact Number: 24911 **SRP Section Number:** 15.2.8

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xii)

Summary:

Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room.

Potential Impact Number: 24912 **SRP Section Number:** 8.3.1

APPENDIX II
Potential Impacts

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(xiii)

Summary:

Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available.

Potential Impact Number: 24916 **SRP Section Number:** 6.2.1.1.A

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xvii)

Summary:

Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples.

Potential Impact Number: 24917 **SRP Section Number:** 6.2.1.1.B

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xvii)

Summary:

Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas

effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples.

Potential Impact Number: 24918 **SRP Section Number:** 6.2.1.1.C

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xvii)

Summary:

Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples.

Potential Impact Number: 24926 **SRP Section Number:** 8.3.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xx)

Summary:

Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources.

Potential Impact Number: 24930 **SRP Section Number:** 6.2.1.1.A

APPENDIX II
Potential Impacts

Document: RG 1.97 **Revision:** 3
Title: INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-watercooled nuclear power plant.

Potential Impact Number: 24931 **SRP Section Number:** 6.2.1.1.B

Document: RG 1.97 **Revision:** 3
Title: INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-watercooled nuclear power plant.

Potential Impact Number: 24932 **SRP Section Number:** 6.2.1.1.C

Document: RG 1.97 **Revision:** 3
Title: INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-watercooled nuclear power plant.

APPENDIX II
Potential Impacts

Potential Impact Number: 24946 **SRP Section Number:** 6.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xxvi)

Summary:

Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal 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.

Potential Impact Number: 24947 **SRP Section Number:** 5.4.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xxvi)

Summary:

Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal 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.

Potential Impact Number: 24948 **SRP Section Number:** 5.4.7

APPENDIX II
Potential Impacts

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xxvi)

Summary:

Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal 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.

Potential Impact Number: 24950 **SRP Section Number:** 9.3.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xxvi)

Summary:

Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal 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.

Potential Impact Number: 24959 **SRP Section Number:** 6.4

Document: NRC GENERIC LETTER 83-36
Title: NUREG-0737 TECHNICAL SPECIFICATIONS

Summary:

APPENDIX II
Potential Impacts

Control Room Habitability Requirements (II.D.3.4)

Licensees should assure that control room operators will be adequately protected against the effects of the accidental release of toxic and/or radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions. If the results of the analyses of postulated accidental release of toxic gases (at or near the plant) indicated a need for installing the toxic gas detection system, it should be included in the Technical Specifications. Typical acceptable LCO and surveillance requirements for such a detection system (e,.g. chlorine detection system) are provided in Enclosure 3. All detection systems should be included in the Technical Specifications.

In addition to the above requirements, other aspects of the control room habitability requirements should be included in the Technical Specifications for control room emergency air filtration system. Two independent control room emergency air filtration system should be operable continuously during all modes of plant operation and capable of meeting design requirements. Sample Technical Specifications are provided in Enclosure 3.

Potential Impact Number: 24961 **SRP Section Number:** 6.4

Document: NUREG CR-5659
Title: Control Room Habitability System Review Models

Summary:

This report provides a method of calculating control room operator doses from postulated reactor accidents and toxic chemical spills as part of the resolution of TMI Action Plan III.D.3.4. The computer codes contained in this report use source concentrations calculated by either TACT5, FPFP, or EXTRAN, and transport them via user-defined flow rates to the control room envelope. The codes compute doses to six organs from up to 150 radionuclides (or 1 toxic chemical) for time steps as short as one second. Supporting codes written in Clipper assist in data entry and manipulation, and graphically display the results of the FORTRAN calculations.

Potential Impact Number: 24963 **SRP Section Number:** 13.1.1

APPENDIX II
Potential Impacts

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(3)(vii)

Summary:

Provide a description of the management plan for design and construction activities, to include: (A) The organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort.

Potential Impact Number: 24964 **SRP Section Number:** 13.1.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue deals with implementation of long-term organization and management improvements. The overall objective of this item is to 'improve licensee safety performance and ability to respond to accidents by upgrading licensee groups responsible for radiation protection and plant operation. The areas to be upgraded include: (1) staff size; (2) education and experience of staff members; (3) plant operating and emergency procedures; (4) management awareness of, and attention to, safety matters; and (5) numbers and types of personnel available to respond to accidents.' The evaluation of this issue includes the consideration of Item II.J.3.1.

The NRC stated its movement toward performance-based rather than prescriptive regulation in its 1986 Policy and Planning Guidance (NUREG-0885, Issue 5). The NRC is approaching Management and Organization by improving its responsiveness to licensee performance, e.g., SALP improvement. The Nuclear Utility Management and Resources Committee (NUMARC) was established by the industry in March 1984 with an objective of reviewing management and people-related issues in nuclear operations

and developing industry-wide resolutions. The Commission has responded to NUMARC initiatives in a positive manner. In a letter dated January 23, 1985, to J. H. Miller, Chairman of NUMARC, Chairman Palladino supported initiatives which include improvement in management and organization and stated, in part, '...we strongly encourage industry efforts to enhance the performance of utility corporate management.' Based upon industry initiatives and the Commission's policy guidance, work on Items I.B.1.1(1,2,3,4) was terminated.

OIE routinely develops and issues inspection procedures which address new or revised regulations and requirements. Thus, Item I.B.1.1(5) was determined to be resolved.

In July 1984, OIE noted /652/ that the proposed revision to Regulatory Guide 1.33 was being addressed under Issue 75. This effort negated the need to pursue a separate resolution to those parts of Items I.B.1.1(6,7) that required a revision to Regulatory Guide 1.33. The parts of Items I.B.1.1(6,7) that require a revision to Regulatory Guide 1.8 are addressed under Item I.A.2.6(1).

Potential Impact Number: 24965 **SRP Section Number:** 13.1.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

The objective of this task is to improve the qualification of licensees for operating nuclear power plants by requiring greater oversight of design, construction, and modification activities.

The criteria and regulatory guidelines for this issue are being addressed and developed by DHFS as a part of Item I.B.1.1. Therefore, this issue is included in, and prioritized with, Item I.B.1.1.

Potential Impact Number: 24966 **SRP Section Number:** 8.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

APPENDIX II
Potential Impacts

Location: 50.34(f)(2)(v)

Summary:

Provide for automatic indication of the bypassed and operable status of safety systems.

Potential Impact Number: 24967 **SRP Section Number:** 7.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(v)

Summary:

Provide for automatic indication of the bypassed and operable status of safety systems.

Potential Impact Number: 24968 **SRP Section Number:** 8.1

Document: RG 1.47 **Revision:** 0
Title: BYPASSED AND INOPERABLE STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS

Summary:

This guide 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 (as defined in IEEE Std 279-1971), 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.

Potential Impact Number: 24969 **SRP Section Number:** 7.1

Document: RG 1.47 **Revision:** 0

APPENDIX II
Potential Impacts

Title: BYPASSED AND INOPERABLE STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS

Summary:

This guide 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 (as defined in IEEE Std 279-1971), 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.

Potential Impact Number: 24970 **SRP Section Number:** 8.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This TMI Action Plan item 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.

Based on the estimated public risk reduction and the value/impact score, this issue was given a MEDIUM priority ranking.

Potential Impact Number: 24971 **SRP Section Number:** 7.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This TMI Action Plan item 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.

APPENDIX II
Potential Impacts

Based on the estimated public risk reduction and the value/impact score, this issue was given a MEDIUM priority ranking.

Potential Impact Number: 24981 **SRP Section Number:** 5.4.7

Document: NRC BULLETIN 79-05A **Revision:** 0 , 4/5/79
Title: NUCLEAR INCIDENT AT THREE MILE ISLAND-SUPPLEMENT

Summary:

Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service. b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing. c. A means of notifying involved reactor operating personnel whenever a safety-related system is removed from and returned to service.

Potential Impact Number: 24982 **SRP Section Number:** 6.3

Document: NRC BULLETIN 79-05A **Revision:** 0 , 4/5/79
Title: NUCLEAR INCIDENT AT THREE MILE ISLAND-SUPPLEMENT

Summary:

Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service. b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing. c. A means of notifying involved reactor operating personnel whenever a safety-related system is removed from and returned to service.

APPENDIX II
Potential Impacts

Potential Impact Number: 24983 **SRP Section Number:** 5.4.7

Document: NRC BULLETIN 79-06B
Title: REVIEW OF OPERATIONAL ERRORS AND SYSTEM MISALIGNMENTS IDENTIFIED DURING THE THREE MILE ISLAND INCIDENT

Summary:

Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.
b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing. c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

Potential Impact Number: 24984 **SRP Section Number:** 6.3

Document: NRC BULLETIN 79-06B
Title: REVIEW OF OPERATIONAL ERRORS AND SYSTEM MISALIGNMENTS IDENTIFIED DURING THE THREE MILE ISLAND INCIDENT

Summary:

Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.
b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing. c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

Potential Impact Number: 24985 **SRP Section Number:** 5.4.7

APPENDIX II
Potential Impacts

Document: NRC BULLETIN 79-08
Title: EVENTS RELEVANT TO BOILING WATER POWER REACTORS IDENTIFIED DURING THREE MILE ISLAND INCIDENT

Summary:

Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service. b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing. c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

Potential Impact Number: 24986 **SRP Section Number:** 6.3

Document: NRC BULLETIN 79-08
Title: EVENTS RELEVANT TO BOILING WATER POWER REACTORS IDENTIFIED DURING THREE MILE ISLAND INCIDENT

Summary:

Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service. b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing. c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

Potential Impact Number: 24987 **SRP Section Number:** 1.8

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

APPENDIX II
Potential Impacts

Summary:

II.K.1.5, 'Review and Modify Procedures for Removing Safety-Related Systems from Service,' was designated in Enclosures 1 and 2 as an item applicable to all operating license holders and applicants. No clarification was provided for this item in Enclosure 3.

Potential Impact Number: 24988 **SRP Section Number:** 6.3

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

II.K.5.1, 'Review and Modify Procedures for Removing Safety-Related Systems from Service,' was designated in Enclosures 1 and 2 as an item applicable to all operating license holders and applicants. No clarification was provided for this item in Enclosure 3.

Potential Impact Number: 24989 **SRP Section Number:** 5.4.6

Document: NRC BULLETIN 79-08
Title: EVENTS RELEVANT TO BOILING WATER POWER REACTORS IDENTIFIED DURING THREE MILE ISLAND INCIDENT

Summary:

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, describe in summary form the procedure, by which this action is taken in a timely sense.

Potential Impact Number: 24990 **SRP Section Number:** 5.4.6

Document: NUREG 0737

APPENDIX II
Potential Impacts

Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Item II.K.1.22, 'Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When Feedwater System Not Operable,' is identified as a TMI action plan item applicable to all operating license holders and applicants in Enclosures 1 and 2. No clarification is provided for this item in Enclosure 3.

Potential Impact Number: 24991 **SRP Section Number:** 5.4.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xxi)

Summary:

Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable.

Potential Impact Number: 24995 **SRP Section Number:** 15.1.5

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

10 CFR 50.34(f)(1)(iii): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Impact of RCP Seal Damage Following SBLOCA With Loss of Offsite Power (TMI Item II.K.2(16)) and

Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Effect of Loss of ac Power on Pump Seals (TMI Item II.K.3(25))

APPENDIX II
Potential Impacts

TMI Item II.K.2(16) is applicable to PWRs only and does not need to be addressed for the ABWR, but II.K.3(25)applies to BWRs. Paragraph (1)(iii) of 10 CFR 50.34(f) requires analyses or experiments to determine the consequences at each plant of a loss of cooling water to the reactor recirculation pump seal coolers. Pump seals should be designed to withstand a complete loss of ac power for at least two hours and the adequacy of the seal design should also be demonstrated. The design should prevent an excessive loss of reactor coolant inventory after an anticipated operational occurrence. It is assumed that the loss of ac power constitutes a loss of offsite power.

SSAR Sections 5.4.1 and 1A.2.30 state that the ABWR design features reactor internal pumps (RIPs) that do not require pump shaft seals. During a loss of ac power, the RIPs are shut down automatically, but there are no shaft seals that require restoration of cooling. During its review of this issue, the staff required GE to confirm that the failure of the following systems would not cause a LOCA:

- recirculation motor cooling system

- recirculation motor seal purge system

- recirculation motor inflatable shaft seal subsystem.

GE confirmed in SSAR Section 1A.2.30 that an ac failure would temporarily disrupt the operation of these systems, but their failure would not generate a LOCA. Therefore, the staff concludes that GE adequately addressed the requirements of this TMI item for the ABWR design.

Potential Impact Number: 25002 **SRP Section Number:** 6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

All 7 B&W plants with OLs and all OL applicants with B&W reactors were required to comply with this NUREG-0660 item which called for the affected plants to assess the loading on steam generator tube sheets induced from slug flow during natural circulation cooldown. A clarification affecting both groups of plants was issued in NUREG-0737.

APPENDIX II
Potential Impacts

CONCLUSION

This item was RESOLVED and requirements were issued.

Potential Impact Number: 25003 **SRP Section Number:** 15.0

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

All 7 B&W plants with OLs were required to comply with this NUREG-0660 item which called for the plants to determine the consequence of voiding in the reactor vessel and the hot legs during normal anticipated transients. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Item I.C.1. Clarifications were issued in NUREG-0737 to include all PWRs (OLs and OL applicants).

CONCLUSION

This item was RESOLVED, requirements were issued.

Potential Impact Number: 25004 **SRP Section Number:** 15.2.7

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

All 7 B&W plants with OLs were required to comply with this NUREG-0660 item which called for the evaluation of the steam generator model in the small-break licensing code (CRAFT-2) by predicting the Crystal River asymmetric cooldown start-up test. For OL applicants with B&W reactors, it was

determined that the issue was being addressed by Item I.C.1. Clarifications were issued in NUREG-0737 to include all PWRs (OLs and OL applicants).

CONCLUSION

APPENDIX II
Potential Impacts

This item was RESOLVED, requirements were issued.

Potential Impact Number: 25006 **SRP Section Number:** 15.1.5

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

All operating Babcock and Wilcox (B&W) plants were ordered to be shut down shortly after the TMI-2 accident. The orders included both short-term and long-term actions. The NRR Bulletins and Orders Task Force reviewed the licensees' compliance with the short-term actions of the orders and issued safety evaluation reports which served as the basis for plant restart. Additional items were identified in the review of the long-term actions which require further work by the licensees.

Clarification

The licensees were required to comply with the Commission Orders regarding certain short-term and long-term auxiliary feedwater system (AFWS) modifications. The staff evaluated the short-term actions, and safety evaluations were prepared before the plants were allowed to return to operation. The staff evaluation of the additional (long-term) items will be performed in conjunction with item II.E.1.1 in NUREG-0660, Auxiliary Feedwater System Evaluation and item II.E.1.2, AFWS Automatic Initiation and Flow Indicator.

Applicability

This requirement applies to all B&W operating reactors.

Potential Impact Number: 25007 **SRP Section Number:** 15.2.8

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

APPENDIX II
Potential Impacts

Summary:

Position

All operating Babcock and Wilcox (B&W) plants were ordered to be shut down shortly after the TMI-2 accident. The orders included both short-term and long-term actions. The NRR Bulletins and Orders Task Force reviewed the licensees' compliance with the short-term actions of the orders and issued safety evaluation reports which served as the basis for plant restart. Additional items were identified in the review of the long-term actions which require further work by the licensees.

Clarification

The licensees were required to comply with the Commission Orders regarding certain short-term and long-term auxiliary feedwater system (AFWS) modifications. The staff evaluated the short-term actions, and safety evaluations were prepared before the plants were allowed to return to operation. The staff evaluation of the additional (long-term) items will be performed in conjunction with item II.E.1.1 in NUREG-0660, Auxiliary Feedwater System Evaluation and item II.E.1.2, AFWS Automatic Initiation and Flow Indicator.

Applicability

This requirement applies to all B&W operating reactors.

Potential Impact Number: 25011 **SRP Section Number:** 15.1.5

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

All 7 B&W plants with OLs were initially required to complete this long-term NUREG-0660 /48/ action. However, a clarification was issued in NUREG-0737 which superseded this item with Items II.E.1.1 and II.E.1.2. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Items II.E.1.1 and II.E.1.2.

CONCLUSION

APPENDIX II
Potential Impacts

This item is covered in Items II.E.1.1 and II.E.1.2.

Potential Impact Number: 25012 **SRP Section Number:** 15.2.8

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

All 7 B&W plants with OLs were initially required to complete this long-term NUREG-0660 /48/ action. However, a clarification was issued in NUREG-0737 which superseded this item with Items II.E.1.1 and II.E.1.2. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Items II.E.1.1 and II.E.1.2.

CONCLUSION

This item is covered in Items II.E.1.1 and II.E.1.2.

Potential Impact Number: 25019 **SRP Section Number:** 15.1.5

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Evaluate 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.

Applicability

This requirement applies to all B&W operating reactors and operating license applicants.

APPENDIX II
Potential Impacts

Potential Impact Number: 25020 **SRP Section Number:** 15.2.8

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

Position

Evaluate 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.

Applicability

This requirement applies to all B&W operating reactors and operating license applicants.

Potential Impact Number: 25022 **SRP Section Number:** 15.1.5

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

All 7 B&W plants with OLs and all OL applicants with B&W reactors were required to comply with this NUREG-0660 item which called for the investigation of the consequences of losing coolant to the seals of the reactor coolant pumps during loss of offsite power. Clarifications affecting both groups of plants were issued in NUREG-0737.

CONCLUSION

This item was RESOLVED, requirements were issued.

APPENDIX II
Potential Impacts

Potential Impact Number: 25023 **SRP Section Number:** 15.2.8

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

All 7 B&W plants with OLs and all OL applicants with B&W reactors were required to comply with this NUREG-0660 item which called for the investigation of the consequences of losing coolant to the seals of the reactor coolant pumps during loss of offsite power. Clarifications affecting both groups of plants were issued in NUREG-0737.

CONCLUSION

This item was RESOLVED, requirements were issued.

Potential Impact Number: 25024 **SRP Section Number:** 15.1.5

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

DESCRIPTION

This NUREG-0660 item was originated to evaluate RCP seal damage and leakage during a SBLOCA in B&W plants (OLs and OL applicants). However, prior to the publication of NUREG-0660, it was determined that the issue was being addressed by Item II.K.2(16).

CONCLUSION

This item is covered in Item II.K.2(16).

Potential Impact Number: 25025 **SRP Section Number:** 15.2.8

APPENDIX II
Potential Impacts

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

DESCRIPTION

This NUREG-0660 item was originated to evaluate RCP seal damage and leakage during a SBLOCA in B&W plants (OLs and OL applicants). However, prior to the publication of NUREG-0660, it was determined that the issue was being addressed by Item II.K.2(16).

CONCLUSION

This item is covered in Item II.K.2(16).

Potential Impact Number: 25032 **SRP Section Number:** 1.8

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This section of NUREG 0933 describes the background related to this issue including the degraded core rulemaking referred to in the SRP. The discussion indicates that the specific proposed rule, and it's derivatives published in the Federal Register are no longer being pursued.

The following summarizes the conclusion of this NUREG 0933 item:

Based on the above evaluation, this item was given a high priority ranking. Work performed by RES on the hydrogen control aspect of the issue resulted in Revision 3 a Hydrogen Control Rule that was approved by the Commission and published in the Federal Register on January 25, 1985. The severe accident portion of the issue was addressed in April 1983 by a Policy Statement that set forth the Commission's intentions for rulemakings and other regulatory actions for resolving safety issues related to reactor accidents more severe than design basis accidents (48 FR 16014). Certain severe accident technical issues identified under the discussion of long-term rulemaking were to be dealt with for future and existing plants through procedures and

ongoing severe accident programs identified in the Policy Statement and described more fully in Chapter IV of NUREG-1070. Thus, with the issuance of the rule on hydrogen control, this item was resolved and new requirements were established.

Potential Impact Number: 25033 **SRP Section Number:** 5.4.7

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

The TMI-2 accident resulted in a metal-water reaction which involved hydrogen generation in excess of the amounts specified in 10 CFR 50.44. As a result, it became apparent to the NRC that additional hydrogen control and mitigation measures would have to be considered for all nuclear power plants.

The purpose of this TMI Action Plan item II.B.7 was to establish the technical basis for the interim hydrogen control measures on small containment structures and to establish the basis for continued operation and licensing of plants, pending long-term resolution of the hydrogen control issue. The long-term resolution of this issue was accomplished by rulemaking as part of Item II.B.8. A final rule was published on December 2, 1981 requiring inerting of the small BWR MARK I and II containments. In addition, based on Commission guidance, interim hydrogen control systems were required as a licensing condition for the intermediate volume ice condenser and MARK III containments. A proposed rule was published on December 23, 1981 (Federal Register 46 FR 62281) which required these systems for the intermediate volume containments. Except for pending CP and ML applications, no additional requirements for hydrogen control or hydrogen analyses were imposed at that time for large, dry containments. However, the proposed rule required that dry containments be analyzed to determine their ability to accommodate the release of large quantities of hydrogen (75% metalwater reaction). Also, hydrogen control requirements were established as part of the final Near Term CP and ML Rule published on January 15, 1982.

Based on the accomplishments above, the basis for continued operation and licensing of plants with respect to the hydrogen control issue was established. Future work related to finalizing the proposed rule dealing with intermediate volume containments (Ice Condenser and MARK III) and large, dry containments continued as part of Item II.B.8.

APPENDIX II
Potential Impacts

Potential Impact Number: 25034 **SRP Section Number:** 13.2.1

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

NUREG-0660 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. The regulations will be applied to all current and future operating power plants.

On October 24, 1984, the Commission notified the staff that it would not promulgate a rule on fitness for duty for a minimum of
two years, but would issue a policy statement on the subject. A proposed policy statement was submitted to the Commission on January 17, 1985, as SECY-85-21. In a separate action, a notice withdrawing the final fitness for duty rule was submitted to the Commission on April 12, 1985, in SECY-85-21A. The proposed policy statement (SECY-85-21) was reaffirmed by the staff in SECY-85-12B submitted to the Commission on August 26, 1985. In recognition of industry's efforts in establishing fitness for duty programs, the Commission approved a Policy Statement in July 1986. Thus, this issue was RESOLVED and no new requirements were established.

Potential Impact Number: 25035 **SRP Section Number:** 13.2.1

Document: NRC GENERIC LETTER 89-23
Title: NRC STAFF RESPONSES TO QUESTIONS PERTAINING TO IMPLEMENTATION OF 10 CFR PART 26

Summary:

Enclosed is a copy of NUREG-1385, 'Fitness-for-Duty in the Nuclear Power Industry: Responses to Implementation Questions.' The NUREG responds to questions concerning implementation of the Fitness-for-Duty Rule (10 CFR Part 26), and may be useful in developing programs which implement the rule.

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The attached NUREG does not constitute any new requirements beyond 10 CFR Part 26. Furthermore, transmittal of the NUREG does not impose any new regulatory requirements.

Potential Impact Number: 25037 **SRP Section Number:** 13.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

Under the TMI Action Plan I.A.2.2, the NRC may require reactor licensees to review their training and qualification programs for all operations personnel. This is interpreted to include licensed and auxiliary operators, technicians, maintenance personnel and supervisors. The review is to examine current practices in light of the safety significance of the duties of the operations staff. If the review determines that the current practices adequately assure proper safety-related staff conduct, then documentation of the justification for this determination is required. The documentation need not be submitted to the NRC but must be maintained on site. If the review uncovers inadequacies, the licensee is required to upgrade the training and qualification practices to ensure adequate performance of operations personnel.

This issue was determined to be high priority. However, in June 1985, the Commission recognized that the industry had made progress in developing programs to improve nuclear utility training and personnel qualification. As a result, the Commission adopted a Policy Statement on Training and Qualifications which made the training accreditation program managed by INPO the focus of training improvement in the industry. Thus, this item was RESOLVED and no new requirements were established.

Potential Impact Number: 25038 **SRP Section Number:** 13.2.2

Document: NRC POLICY STATEMENT 53 FR 46603
Title: Commission Policy Statement on Training and Qualification of Nuclear Power Plant Personnel

Summary:

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This statement presents the policy of the Nuclear Regulatory Commission (NRC) with respect to training and personnel qualification and describes the activities that the NRC will use to execute its responsibilities to ensure the health and safety of the public. This Policy Statement provides regulatory guidance called for by the Nuclear Waste Policy Act (Pub. L 97-425) which directed the NRC to promulgate regulations or other regulatory guidance for the training and qualifications of civilian nuclear power plant operator, supervisors, technicians and other appropriate operating personnel. In recognition of industry initiatives underway to upgrade training programs, the NRC endorses the Institute of Nuclear Power Operations (INPO)-managed Training Accreditation Program in that it encompasses the element of performance-based training and will provide the basis to ensure that personnel have qualifications commensurate with the performance requirements of their jobs.

Potential Impact Number: 25039 **SRP Section Number:** 1.8

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.44

Summary:

10 CFR 50.44 establishes the regulatory requirements associated with containment combustible gas control.

Potential Impact Number: 25040 **SRP Section Number:** 13.2.1

Document: CODE OF FED. REGS 10CFR26
Title: FITNESS-FOR-DUTY PROGRAM

Summary:

This Part 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, and by licensees authorized to possess, use, or transport formula quantities of strategic special nuclear material (SSNM).

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Potential Impact Number: 25044 **SRP Section Number:** 5.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This NUREG-0660 item required all operating plants to report safety and relief valve failures promptly and challenges annually. All OL applicants were also required to complete this item. Clarifications affecting both groups of plants were issued in NUREG-0737. This item was RESOLVED, requirements were issued.

Potential Impact Number: 25045 **SRP Section Number:** 15.1.5

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This NUREG-0660 item required all PWR operating plants to study the need for automatic trip of RCPs and to modify procedures or designs, as appropriate. OL applicants with PWRs were also required to complete this item. Clarifications affecting both groups of plants were issued in NUREG-0737. This item was RESOLVED, requirements were issued.

Potential Impact Number: 25046 **SRP Section Number:** 15.2.8

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This NUREG-0660 item required all PWR operating plants to study the need for automatic trip of RCPs and to modify procedures or designs, as appropriate. OL applicants with PWRs were also required to complete this item. Clarifications affecting

both groups of plants were issued in NUREG-0737. This item was RESOLVED, requirements were issued.

Potential Impact Number: 25050 **SRP Section Number:** 15.0

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This NUREG-0660 item required all B&W operating plants (OLs and OL applicants) to document that their PORVs would open in less than 5% of all anticipated overpressure transients. Clarifications were issued in NUREG-0737 to include all PWRs. This item was RESOLVED and requirements were issued.

Potential Impact Number: 25051 **SRP Section Number:** 6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This NUREG-0660 item required that the anticipatory trip modification proposed by some licensees to confine the range of use of high-power levels not be made until it could be shown that the probability of a small-break LOCA resulting from a stuck-open PORV was substantially unaffected by the modification. The applicability of the item to W operating plants and OL applicants with W reactors was to be determined on a plant-by-plant basis. Clarifications affecting both groups of plants were issued in NUREG-0737.

This item was RESOLVED, requirements were issued.

Potential Impact Number: 25052 **SRP Section Number:** 5.4.6

Document: NUREG 0933

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Title: A Prioritization of Generic Safety Issues

Summary:

This NUREG-0660 item required all GE plants (OLs and OL applicants) to analyze the benefits to be gained from separating HPCI and RCIC initiation levels and providing auto-start of RCIC on low-low level. Clarifications were issued in NUREG-0737 to include all operating BWRs and OL applicants with RCIC and HPCI systems.

This item was RESOLVED, requirements were issued.

Potential Impact Number: 25054 **SRP Section Number:** 6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This NUREG-0660 item required all GE plants (OLs and OL applicants) to modify their ADS actuation logic to eliminate the need for manual actuation to assure adequate core cooling. A feasibility study and risk assessment study were required to determine the optimum approach. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.

This item was RESOLVED, requirements were issued.

Potential Impact Number: 25055 **SRP Section Number:** 6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This NUREG-0660 item required all GE plants (OLs and OL applicants) to modify their core spray and LPCI system logic so that these systems would restart, if required, to assure adequate core cooling. It was believed that the core spray and LPCI system flow may be stopped by the operator. These systems could not start automatically on loss of

water level if an initiation signal were still present. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.

This item was RESOLVED, requirements were issued.

Potential Impact Number: 25056 **SRP Section Number:** 5.4.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This NUREG-0660 item required all operating GE plants (OLs and OL applicants) to verify that HPCI and RCIC are designed to withstand loss of offsite power for at least 2 hours. Clarifications affecting all BWRs (OLs and OL applicants) with HPCI and RCIC systems were issued in NUREG-0737.

This item was RESOLVED, requirements were issued.

Potential Impact Number: 25058 **SRP Section Number:** 15.1.5

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This NUREG-0660 item required all GE plants (OLs and OL applicants) to verify the adequacy of pump seals to withstand loss of cooling water due to loss of AC power for at least 2 hours. Clarifications were issued in NUREG-0737 to include all BWRs, W and CE operating reactors, and all OL applicants.

This item was RESOLVED, requirements were issued.

Potential Impact Number: 25061 **SRP Section Number:** 6.3

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Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This NUREG-0660 item affected all plants with GE reactors (OLs and OL applicants). These plants were required to assure that air or nitrogen accumulators for ADS valves had sufficient capacity to cycle the valves open five times at design pressure. However, clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.

This item was RESOLVED, requirements were issued.

Potential Impact Number: 25062 **SRP Section Number:** 15.0

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This NUREG-0660 item required all GE plants (OLs and OL applicants) to show that transients combined with the worst single failure would not result in significant fuel damage. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.

This item was RESOLVED, requirements were issued.

Potential Impact Number: 25063 **SRP Section Number:** 6.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This NUREG-0660 item required all GE plants (OLs and OL applicants) to analyze depressurization modes other than full ADS for possible inclusion in emergency procedures. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.

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This item was RESOLVED, requirements were issued.

Potential Impact Number: 25073 **SRP Section Number:** 5.2.2

Document: NRC GENERIC LETTER 83-02
Title: NUREG-0737 Technical Specifications

Summary:

This letter describes appropriate technical specifications (TS) to address TMI action plan item II.K.3.3.

NUREG-0660 stated that safety and relief valve failures be reported promptly and challenges be reported annually. The sections of your TSs that discuss reporting requirements should be accordingly changed; model TSs are given in Enclosure 2. Note that an acceptable alternative would be to report challenges monthly.

Potential Impact Number: 25075 **SRP Section Number:** 5.3.2

Document: RG 1.154 **Revision:** 0
Title: FORMAT AND CONTENT OF PLANT-SPECIFIC PRESSURIZED THERMAL SHOCK SAFETY ANALYSIS REPORTS FOR PRESSURIZED WATER REACTORS

Summary:

This regulatory guide describes a format and content acceptable to the NRC staff for these plant-specific PTS safety analyses and describes acceptance criteria that the NRC staff will use in evaluating licensee analyses and proposed corrective measures.

Potential Impact Number: 25076 **SRP Section Number:** 5.2.2

Document: NRC POLICY STATEMENT 58 FR 39132 Revision Date: 7/22/93
Title: FINAL POLICY STATEMENT ON TECHNICAL SPECIFICATIONS IMPROVEMENTS FOR NUCLEAR POWER PLANTS

APPENDIX II
Potential Impacts

Summary:

This statement presents the policy of the Nuclear Regulatory Commission (NRC) with respect to the scope and purpose of Technical Specifications for nuclear power plants as required by 10 CFR 50.36. It establishes a specific set of objective criteria as guidance for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. It encourages licensees to implement a voluntary program to update their Technical Specifications to be consistent with improved vendor-specific Standard Technical Specifications (STS).

Potential Impact Number: 25078 **SRP Section Number:** 5.4.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(1)(v)

Summary:

Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, 'high pressure core spray' for 'high pressure coolant injection' and 'HPCS' for 'HPCI').

Potential Impact Number: 25080 **SRP Section Number:** 6.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(1)(viii)

Summary:

Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the

systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present.

Potential Impact Number: 25081 **SRP Section Number:** 5.4.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(1)(ix)

Summary:

Perform 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. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI").

Potential Impact Number: 25083 **SRP Section Number:** 6.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(1)(xi)

Summary:

Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown.

Potential Impact Number: 25084 **SRP Section Number:** 15.0

Document: NRC GENERIC LETTER 81-32

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Title: NUREG-0737, ITEM II.K.3.44-EVALUATION OF ANTICIPATED TRANSIENTS COMBINED WITH SINGLE FAILURE

Summary:

The letter indicates that the staff completed a review of the General Electric BWR Owner Group report dated December 29, 1980 pertaining to II.K.3.44 of NUREG-0737. The staff found the response contained in the report to be acceptable on a generic basis. However, requested licensees confirm the applicability of this report by verification that the assumptions and initial conditions used in the analyses were representative to the licensees specific plant.

Potential Impact Number: 25085 **SRP Section Number:** 15.0

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

This issue requires licensees and applicants of GE plants to demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery for anticipated transients combined with the worst single failure and proper operator actions. This category includes transients that result from a stuck-open relief valve.

GE has endorsed the results of the BWROG study (NEDO-24708, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," which had been accepted by the staff) as applicable to ABWR in this area. In a letter dated December 29, 1980, from D.B. Walters (BWROG) to NRC, the BWROG enclosed an evaluation (NEDO-24708) in which it stated that the worst-case transient with single failure combination for BWR/2-6 plants is the loss-of- feedwater event with a failure of the high-pressure ECCS. However, since the ABWR design includes three high-pressure core injection systems, the probability of a loss of all high-pressure ECCS is low. GE also considered an event with a stuck-open relief valve, and a high-pressure ECCS failure, and concluded that the core remained covered throughout the transient either because the RCIC system operated or because the RCS was depressurized by automatic or manual means, permitting low-pressure inventory makeup. GE also assumed the operator would manually depressurize the vessel to permit low-pressure injection.

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The staff has reviewed the results of the BWROG's study and its applicability to ABWR and finds that GE has shown that the ABWR design can keep the core covered and have no fuel damage from core uncovery for transients combined with the worst single failure. Therefore, GE adequately addressed the requirements of this TMI item for the ABWR design.

Potential Impact Number: 25086 **SRP Section Number:** 6.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(xxviii)

Summary:

Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID 14844 source term release, and make necessary design provisions to preclude such problems.

Potential Impact Number: 25088 **SRP Section Number:** 6.4

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue was raised in response to a 1982 ACRS letter which outlined deficiencies in the design, installation, maintenance and testing of control room systems for maintaining operator habitability. The solution to this issue 'has been identified.'

Potential Impact Number: 25091 **SRP Section Number:** 15.1.5

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

APPENDIX II
Potential Impacts

Summary:

Issue II.K.3(25) in NUREG-0933, addressed requiring that BWR licensees determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the RCP seal coolers. Adequacy of the seal design to withstand a LOOP should be demonstrated. This position should prevent excessive loss of RCS inventory following an anticipated operational occurrence.

ABB-CE states CESSAR-DC Section 20.2.121 that the RCP seals are normally cooled by redundant systems: seal injection from the CVCS and CCWS. In the event of LOOP, seal injection can be restored by manually aligning Class 1E power to the normal charging pump and CCWS pump, or by using the positive displacement dedicated seal injection pump. Two of the four CCWS pumps can be powered from the EDGS to cool the RCP seals.

During a complete loss of ac power (i.e., loss of both offsite power and the diesel generators), power can be supplied to the dedicated seal injection pump, one charging pump, and one CCWS pump from the onsite ac power source described in CESSAR-DC Section 8.3.1.1.5. ABB-CE states that the use of redundant, diverse seal cooling systems with multiple electrical power sources significantly reduces the probability of losing seal cooling for the RCPs.

The requirements for this issue in NUREG-0737 are that the consequences of a loss of cooling water to the pump seal coolers is determined and the pump seals should be designed to withstand a complete LOOP for at least 2 hours. If seal failure is the consequence of loss of cooling water for 2 hours, an acceptable solution would be providing emergency power to the CCWS pump.

Resolution of this issue also includes the resolution of Issue 23, "Reactor Coolant Pump Seal Failures." The staff reviewed the resolution of Issue 23 in Section 20.2 of this chapter and considers this issue resolved for the System 80+ design.

Potential Impact Number: 25095 **SRP Section Number:** 15.1.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(1)(iii)

Summary:

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Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage.

Potential Impact Number: 25099 **SRP Section Number:** 3.7.3

Document: FINAL SER CE80 CH 1
Title: INTRODUCTION AND GENERAL DISCUSSION

Summary:

In accordance with 10 CFR 52.48, the staff used the current regulations in 10 CFR Parts 20, 50, 73, and 100 in reviewing ABB-CE's application for design certification. During this review, the staff recognized that if certain regulations were applied to the System 80+ design, the use of those regulations would not serve the underlying purpose of the rule and/or would not be needed for achieving compliance with the fundamental intent of the rule. These exemptions are discussed in the sections of this report listed below.

In the SRM pertaining to SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor Designs," the Commission approved the staff's recommendation to proceed with design-specific rulemakings as part of individual design certifications to resolve selected technical and severe accident issues as "applicable regulations" for the advanced boiling water reactor and System 80+ standard designs. Among these issues are staff positions that deviate from and/or are not encoded in current regulations applicable to the System 80+ design.

These issues were addressed in various Commission papers, such as SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced LWR designs." The "applicable regulations" that are specific to the System 80+ design are identified and evaluated in this report. The proposed design certification rule will then include these "applicable regulations" for the System 80+ design for the purposes of 10 CFR 52.48, 52.54, 52.59, and 52.63. These "applicable regulations" are discussed in the sections of this report, including Section 3.1.1, "Exemption from operating-basis earthquake design requirement."

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Potential Impact Number: 25100 **SRP Section Number:** 3.7.3

Document: FINAL SER ABWR CH 1
Title: INTRODUCTION AND GENERAL DISCUSSION

Summary:

In the SRM pertaining to SECY 91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor Designs," the Commission approved the staff's recommendation to proceed with design specific rulemakings as part of design certification rulemakings to establish selected technical and severe accident issues as "applicable regulations" for the ABWR and System 80+ standard designs. These issues included staff positions that deviated from and/or were not encoded in current regulations applicable to the standard designs.

These issues were addressed in various Commission papers, such as SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolution and Advanced LWR Designs." The "applicable regulations" that are specific to the ABWR design are identified and evaluated in this report, including Section 3.1.1, Exemption from operating basis earthquake design requirement.

Potential Impact Number: 25102 **SRP Section Number:** 19.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10CFR50.34(f)(3)(iv)

Summary:

Plants should provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system.

Potential Impact Number: 25103 **SRP Section Number:** 15.2.8

Document: NUREG 0933

APPENDIX II
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Title: A Prioritization of Generic Safety Issues

Summary:

DESCRIPTION

All 7 B&W plants with OLs completed this short-term NUREG-0660 action before they were permitted to restart. These accomplishments were made in July 1979, prior to the publication of NUREG-0660. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Items II.E.1.1 and II.E.1.2.

CONCLUSION

This item was RESOLVED and requirements were issued.

Potential Impact Number: 25104 **SRP Section Number:** 15.2.7

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

DESCRIPTION

All 7 B&W plants with OLs completed this short-term NUREG-0660 action before they were permitted to restart. These accomplishments were made in July 1979, prior to the publication of NUREG-0660. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Items II.E.1.1 and II.E.1.2.

CONCLUSION

This item was RESOLVED and requirements were issued.

Potential Impact Number: 25105 **SRP Section Number:** 15.1.5

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

APPENDIX II
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Summary:

DESCRIPTION

All 7 B&W plants with OLs completed this short-term NUREG-0660 action before they were permitted to restart. These accomplishments were made in July 1979, prior to the publication of NUREG-0660. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Items II.E.1.1 and II.E.1.2.

CONCLUSION

This item was RESOLVED and requirements were issued.

Potential Impact Number: 25107 **SRP Section Number:** 5.4.7

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

TASK II.E.3: Decay Heat Removal

The objective of this task was to improve the reliability and capability of nuclear power plant systems for removing decay heat and achieving safe shutdown conditions following transients and under post-accident conditions.

ITEM II.E.3.2: Systems Reliability

This item was intended to focus on shutdown heat removal system reliability. Shutdown heat removal systems generally consist of two independent trains, each of which is quite reliable. Moreover, other systems can be used to prevent a core-melt under many circumstances. Nevertheless, given the importance of decay heat removal, the reliability of the shutdown heat removal systems remained in question. The issue called for NRR to conduct a generic study to assess the capability and reliability of shutdown heat removal systems under various transients and degraded plant conditions, including complete loss of all feedwater. Deterministic and probabilistic methods were to be used to identify design weaknesses and possible system modifications that could be made to improve the capability and reliability of these systems under all shutdown conditions (i.e., startup, hot standby, shutdown, etc.). This

item was incorporated into the resolution of Issue A-45, 'Shutdown Decay Heat Removal Requirements.'

ITEM II.E.3.3: Coordinated Study of Shutdown Heat Removal Requirements

A shutdown heat removal system is necessary in a nuclear reactor to establish and maintain a safe shutdown condition during normal and accident conditions. If the normal shutdown heat removal system does not perform its intended safety function, then an alternate method must be used. Therefore, this TMI Action Plan item called for a coordinated effort to evaluate shutdown heat removal requirements in a comprehensive manner, thereby permitting a judgment of adequacy in terms of overall system requirements. As part of this effort, a study was to be conducted to assess the desirability of, and possible requirements for, a diverse heat removal path, such as feed-and-bleed, particularly if all secondary side cooling were unavailable. The need for at removal systems for PWRs and BWRs was to be evaluated based on value/impact or cost/benefit analyses. If such systems appeared to provide a significant safety benefit, alternative concepts were to be studied and recommendations made. This item was reviewed and considered in the resolution of Issue A-45, 'Shutdown Decay Heat Removal Requirements.'

Potential Impact Number: 25108 **SRP Section Number:** 15.2.6

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

DESCRIPTION

All 7 B&W plants with OLs completed this short-term NUREG-0660 action before they were permitted to restart. These accomplishments were made in July 1979, prior to the publication of NUREG-0660. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Items II.E.1.1 and II.E.1.2.

CONCLUSION

This item was RESOLVED and requirements were issued.

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Potential Impact Number: 25116 **SRP Section Number:** 5.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix H

Summary:

Paragraph II.B of 10 CFR 50, Appendix H states that reactor vessels that do not meet the conditions of paragraph II.A. of the Appendix must have their beltline materials monitored in accordance with the requirements of the appendix. Paragraph II.B.1 of Appendix H states, "That part of the surveillance program conducted prior to the first capsule withdrawal must meet the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. Later editions of ASTM E 185 may be used, but including only those editions through 1982. For each capsule withdrawal after July 26, 1983, the test procedures and reporting requirements must meet the requirements of ASTM E 185-82 to the extent practical for the configuration of the specimens in the capsule. For each capsule withdrawal prior to July 26, 1983 either the 1973, the 1979, or the 1982 edition of ASTM E 185 may be used."

Potential Impact Number: 25117 **SRP Section Number:** 5.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix G, paragraph IV.A

Summary:

Paragraph IV.A of Appendix G states that the pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the ASME Code and the appendix for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences.

Potential Impact Number: 25118 **SRP Section Number:** 10.3

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Potential Impacts

Document: NUREG 0800
Title: STANDARD REVIEW PLAN

Summary:

In SRP Section 6.2.4 the Containment Systems and Severe Accidents Branch (SCSB) reviews the Containment Isolation portion of the Main Steam System to verify that it is provided with leak detection, isolation, and redundancy/ reliability performance capabilities.

Potential Impact Number: 25119 **SRP Section Number:** 10.3

Document: NUREG 0800
Title: STANDARD REVIEW PLAN

Summary:

In SRP Section 6.2.6 the Containment Systems and Severe Accidents Branch (SCSB) reviews the portion of the Main Steam System that penetrates containment to verify that it is provided with the capability to test valve leakage and determine if it is within acceptable limits.

Potential Impact Number: 25120 **SRP Section Number:** 10.3

Document: NUREG 0800
Title: STANDARD REVIEW PLAN

Summary:

In SRP Section 6.7 the Plant Systems Branch (SPLB) reviews the Main Steam Isolation Valve Leakage Control System (MSIVLCS) design to verify that it meets applicable codes and standards.

Potential Impact Number: 25121 **SRP Section Number:** 9.2.2

APPENDIX II
Potential Impacts

Document: DRAFT RG DG-1008
Title: REACTOR COOLANT PUMP SEALS

Summary:

This draft regulatory guide describes means acceptable to the NRC staff for enhancing safety by including the RCP seals in the QA program to better ensure that the reactor coolant pressure boundary has an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. This guide also describes methods acceptable to the NRC staff for enhancing the capability of nuclear power plants to withstand loss-of-seal-cooling events, given the potential for failure of RCP seals.

Potential Impact Number: 25122 **SRP Section Number:** 9.3.4

Document: DRAFT RG DG-1008
Title: REACTOR COOLANT PUMP SEALS

Summary:

This draft regulatory guide describes means acceptable to the NRC staff for enhancing safety by including the RCP seals in the QA program to better ensure that the reactor coolant pressure boundary has an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. This guide also describes methods acceptable to the NRC staff for enhancing the capability of nuclear power plants to withstand loss-of-seal-cooling events, given the potential for failure of RCP seals.

Potential Impact Number: 25134 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The ABWR design eliminates the main steam isolation valve leakage control system for the ABWR plant design. 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 core damage. 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 position, which was discussed in Section II.E of SECY-93-087 and was approved by the Commission in its SRM dated July 21, 1993, is that the main steam piping beyond the outermost isolation valve up to the seismic interface restraint and connecting branch lines up to the first normally closed valve be classified as QG B (SC 2) and seismic Category I. The MSL from the seismic interface restraint up to but not including the turbine stop valve (including branch lines to the first normally closed valve) will be classified as QG B and inspected in accordance with the applicable portions of ASME Section XI. This portion of the steamline may be classified as non-seismic Category I if it has been analyzed using a dynamic seismic analysis method to demonstrate its structural integrity under SSE loading conditions. However, all pertinent QA requirements of Appendix B to 10 CFR Part 50 are applicable to ensure that the quality of the piping material is commensurate with its importance to safety during normal operational, transient, and accident conditions. To ensure the integrity of the remainder of GE's proposed alternative leakage path, the staff position is that (1) the main steam piping between the turbine stop valve and the turbine inlet, the turbine bypass line from the bypass valve to the condenser, and the main steam drain line from the first valve to the condenser are not required to be classified as safety-related or as seismic Category I, but should be analyzed using a dynamic seismic analysis to demonstrate their structural integrity under SSE loading conditions, and (2) the condenser anchorage shall be seismically analyzed to demonstrate that it is capable of sustaining the SSE loading conditions without failure.

The turbine stop valve, the MSL from the turbine stop valve to the turbine, the turbine bypass valve, the turbine bypass line from the bypass valve to the condenser and the main steam drain line from the first valve to the condenser, are all non-safety class, non-seismic Category I, but dynamically analyzed for the SSE.

Although not explicitly stated in SSAR Table 3.2-1, the staff stated that the commitment to QG B requires inservice inspections in accordance with applicable portions of ASME Section XI.

Potential Impact Number: 25136 **SRP Section Number:** 5.2.5

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

APPENDIX II
Potential Impacts

Summary:

The ABWR design provides for monitoring intersystem leakage. Detection of leakage into systems connected to the RCS is necessary because intersystem leakage could result in damage to those connected systems. The systems of concern are:

- HPCF
- RCIC
- RHR Shutdown Cooling (SDC)
- RHR Low Pressure Core Flooder (LPCF)
- Secondary Side of RHR Heat Exchangers
- Secondary Side of RIP Heat Exchangers
- Secondary Side of RWCS Heat Exchangers
- Secondary Side of Fuel Pool Cooling (FPC) Heat Exchangers

Potential Impact Number: 25158 **SRP Section Number:** 13.2.2

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION
Location: 59 FR 5132

Summary:

Federal Register Notice 59 FR 5132 identifies proposed rulemaking with regard to 10 CFR 20 and associated changes to 10 CFR 19.12.

Summary: The Nuclear Regulatory Commission (NRC) proposes to amend its regulations concerning radiation protection requirements. The proposed rule would: Delete the definition of "Controlled area" to make it clear that any area to which access is restricted for the purpose of radiological protection is a restricted area as defined in the regulation, revise the definition of "Occupational dose" to delete reference to the "Restricted area," revise the definition of unrestricted area to be consistent with the deletion of controlled area, revise the provision entitled "Instruction to Workers," so that radiation protection training will be provided to all persons with the potential to be occupationally exposed and restore a provision to provide that whenever licensees are required to report exposures of individual members of the public to the NRC, then those individuals are to receive copies of the report.

APPENDIX II
Potential Impacts

Potential Impact Number: 25167 **SRP Section Number:** 5.4.2.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The controls imposed during welding of austenitic stainless steels in the RCPB conform to the recommendations of RG 1.31, "Control of Stainless Steel Welding," and RG 1.71, "Welder Qualification for Areas of Limited Accessibility," or provide acceptable alternative approaches. These controls offer reasonable assurance that welded components of austenitic stainless steel will not develop microfissures during welding and will have high structural integrity. These controls fulfill the quality standards requirements of GDCs 1 and 30, and 10 CFR 50.55a, and conform to the requirements of GDC 14 relative to prevention of leakage and failure of the RCPB.

Potential Impact Number: 25183 **SRP Section Number:** 16.0

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 59 FR 48180

Summary:

Federal Register Notice 59 FR 48180 involves proposed changes to 10 CFR 50.36 with regard to criteria for establishing technical specification limiting conditions for operation.

The NRC is proposing to amend its regulations pertaining to technical specifications for nuclear power reactors. The proposed rule would codify criteria for determining the content of technical specifications. These criteria were developed in recognition of the overly broad use of technical specifications to impose requirements, diverting both NRC and licensee attention from the more important requirements in these documents to the extent that it has resulted in an adverse impact on safety. Licensees covered by these regulations may voluntarily use the criteria as a basis to propose the relocation of existing technical specifications that do not meet any of the criteria from the facility license to licensee-controlled documents.

APPENDIX II
Potential Impacts

Potential Impact Number: 25226 **SRP Section Number:** 2.5.4

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: 59 FR 52255

Summary:

Federal Register Notice 59 FR 52255 involves proposed changes to amend the NRC's regulations regarding plant siting criteria, seismic design criteria, and siting and design dose calculation criteria.

The NRC is proposing to amend its regulations to update the criteria used in decisions regarding power reactor siting, including geologic, seismic, and earthquake engineering considerations for future nuclear power plants. The proposed rule would allow NRC to benefit from experience gained in the application of the procedures and methods set forth in the current regulation and to incorporate the rapid advancements in the earth sciences and earthquake engineering. This proposed rule primarily consists of two separate changes, namely, the source term and dose considerations, and the seismic and earthquake engineering considerations of reactor siting.

Potential Impact Number: 25234 **SRP Section Number:** 3.7.1

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: 59 FR 52255

Summary:

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separate changes, namely, the source term and dose considerations, and the seismic and earthquake engineering considerations of reactor siting.

Potential Impact Number: 25235 **SRP Section Number:** 3.7.2

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: 59 FR 52255

Summary:

Federal Register Notice 59 FR 52255 involves proposed changes to amend the NRC's regulations regarding plant siting criteria, seismic design criteria, and siting and design dose calculation criteria.

The NRC is proposing to amend its regulations to update the criteria used in decisions regarding power reactor siting, including geologic, seismic, and earthquake engineering considerations for future nuclear power plants. The proposed rule would allow NRC to benefit from experience gained in the application of the procedures and methods set forth in the current regulation and to incorporate the rapid advancements in the earth sciences and earthquake engineering. This proposed rule primarily consists of two separate changes, namely, the source term and dose considerations, and the seismic and earthquake engineering considerations of reactor siting.

Potential Impact Number: 25236 **SRP Section Number:** 3.7.3

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: 59 FR 52255

Summary:

Federal Register Notice 59 FR 52255 involves proposed changes to amend the NRC's regulations regarding plant siting criteria, seismic design criteria, and siting and design dose calculation criteria.

APPENDIX II
Potential Impacts

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Potential Impact Number: 25274 **SRP Section Number:** 9.5.1

Document: FINAL SER CE80 CH 19
Title: SEVERE ACCIDENTS

Summary:

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 review of System 80+ fire protection during shutdown. The System 80+ fire-protection program stipulates that the fire- protection systems shall be operable for all modes of operation, including low-power and shutdown operations.

Potential Impact Number: 25275 **SRP Section Number:** 9.5.1

Document: NUREG 1449
Title: SHUT DOWN AND LOW-POWER OPERATION AT NUCLEAR POWER PLANTS IN THE UNITED STATES. FINAL REPORT

Summary:

The report contains the results of the NRC staff's evaluation of shutdown and low-power operations at U.S. commercial nuclear power plants. The report describes studies conducted by the staff in the following areas: operating experience related to shutdown and low-power operations, probabilistic risk assessment of shutdown and

low-power conditions and utility programs for planning and conducting activities during periods the plant is shut down. The report also documents evaluations of a number of technical issues regarding shutdown and low-power operations performed by the staff, including the principal findings and conclusions. Potential new regulatory requirements are

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Potential Impact Number: 25278 **SRP Section Number:** 9.5.1

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The automatic fire detection systems will be designed and installed in accordance with NFPA Standard 72; they will be provided for all significant hazards and safe-shutdown components.

Potential Impact Number: 25279 **SRP Section Number:** 9.5.1

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

A previous amendment to the SSAR stated that the ABWR automatic fire detection systems are designed and installed in accordance with National Fire Protection Association (NFPA) Standards 72D and 72E, and will protect all safe-shutdown components from all significant hazards. In the DFSER, the staff stated that NFPA 72A, 72B, 72D, and 72E have been incorporated into NFPA 72, and are no longer separate standards. Therefore, all references to NFPA 72D and 72E should be changed to NFPA 72. The staff designated this as DFSER Open Item 9.5.1.3.1-1. Amendment 33 to the SSAR states that fire detection systems are designed according to NFPA 72 Class A and NFPA 70 Class 1 requirements. Therefore, DFSER Open Item 9.5.1.3.1-1 is resolved.

APPENDIX II
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Potential Impact Number: 25280 **SRP Section Number:** 9.5.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix R, paragraph III.G.3

Summary:

Alternative or dedicated shutdown capability and its associated circuits, independent of cables, systems or components in the area, room or zone under consideration, shall be provided:

a. Where the protection of systems whose function is required for hot shutdown does not satisfy the requirement of paragraph G.2 of this section; or

b. Where redundant trains of systems required for hot shutdown located in the same fire area may be subject to damage from fire suppression activities or from the rupture or inadvertent operation of fire suppression systems.

Potential Impact Number: 25307 **SRP Section Number:** 19.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: 10 CFR 50.34(f)(2), 10 CFR 50.34(f)(2), 10 CFR 50.34(f)(3)(V)

Summary:

Paragraphs (2)(ix), (2)(ix), 2(xix), and (3)(v) of 10 CFR 50.34(f), pertaining to hydrogen control measures and instrumentation, must be met by all applicants covered by this rule.

(f)(2)(ix) Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (f)(1)(xii) of this section is sufficient at the construction permit stage.

(f)(2)(xvii) Provide instrumentation to measure containment pressure, containment water level, containment hydrogen concentration, containment radiation intensity, and noble gas effluents at all potential accident release points.

(f)(2)(xix) Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

(f)(3)(v) Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that: Containment integrity will 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, Subsubarticle 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.

Potential Impact Number: 25308 **SRP Section Number:** 19.2

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 10 CFR 52.47(a)(1)(ii)

Summary:

10 CFR 52.47(a)(1)(ii) outlines the following requirement:

(a) The requirements of this paragraph apply to all applications for design certification.

(1) An application for design certification must contain:

(ii) Demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f).

Potential Impact Number: 25309 **SRP Section Number:** 19.2

APPENDIX II
Potential Impacts

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-16 the staff addressed hydrogen generation and control provisions for future designs. Because of the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, the staff recommended in SECY 90-16 that evolutionary ALWRs should be designed, as a minimum, to satisfy the following criteria: - accommodate hydrogen generation equivalent to a 100-percent metal-water reaction of the fuel cladding; - limit containment hydrogen concentration to no greater than 10 percent; and, - provide containment-wide hydrogen control (such as igniters or inerting)

The staff recommended that the Commission approve its position that the requirements of 10 CFR 50.34(f)(2)(ix) remain unchanged for the evolutionary ALWRs. In the SRM for SECY 90-016 the Commission approved the staff's position that the requirements of 10 CFR 50.34(f) be followed. This issue was also addressed in SECY 93-087 with the only change being extending the above criteria to the passive plants.

Potential Impact Number: 25310 **SRP Section Number:** 19.2

Document: FINAL SER CE80 CH 19
Title: SEVERE ACCIDENTS

Summary:

In paragraph 19.2.3.3.1.1, Features to Prevent and Mitigate (Hydrogen Generation), the following review is provided: In 10 CFR 52.47(a)(1)(ii), the NRC requires applicants for a standard design certification to demonstrate compliance with any technically relevant portions of the TMI requirements in 10 CFR 50.34(f). In 10 CFR 50.34(f)(2)(ix), the NRC requires a system for hydrogen control that can show with reasonable assurance that uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent during and following an accident that releases an amount of hydrogen equivalent to the amount that would be generated from a 100-percent fuel-clad metal-water reaction, or that the postaccident atmosphere will not support hydrogen combustion.

APPENDIX II
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In SECY-90-016, the staff recommended that the Commission approve the staff's position that the requirements of 10 CFR 50.34(f)(2)(ix) remain unchanged for evolutionary LWRs. In its June 26, 1990, SRM, the Commission approved the staff's position.

The System 80+ core contains approximately 25,240 kg (55,655 lb) of zirconium in the active fuel-clad region. Oxidation of this amount of zirconium with steam would produce about 1,100 kg (2,440 lb) of hydrogen. This amount of hydrogen uniformly distributed throughout the containment would result in a hydrogen concentration of approximately 13 percent. Therefore, to comply with 10 CFR 50.34(f)(2)(ix), ABB-CE has equipped the System 80+ with a HMS composed of 80 shielded GMAC model 7G thermal igniter glow plugs. The intent of the HMS is to ignite the hydrogen as soon as sufficient hydrogen has accumulated to achieve a combustible mixture. This early combustion will limit the hydrogen concentration well below the 10-percent limit referenced in the rule.

Potential Impact Number: 25311 **SRP Section Number:** 19.2

Document: FINAL SER ABWR CH 19
Title: SEVERE ACCIDENTS

Summary:

In paragraph 19.2.3.3.1, Hydrogen Generation and Control, the following review is provided:

Generation and combustion of large quantities of hydrogen is a severe accident phenomenon that can threaten containment integrity. The major source of hydrogen generated is from the oxidation of zirconium with steam when the zirconium reaches temperatures well above normal operating levels. This reaction is commonly referred to as the metal-water reaction. Research indicates that in-vessel hydrogen generation associated with core-damage can vary over a wide range. The specific amount of oxidation is dependent on a variety of parameters related to sequence progression. These include the RCS pressure, the timing and flow rate of reflooding if it occurs, and the temperature profile of the reactor core during the course of the accident sequence. In addition, ex-vessel hydrogen generation must be considered. Hydrogen is produced as a result of ex-vessel core debris reacting with steam or concrete.

19.2.3.3.1.1 Preventive and/or Mitigative Features: 10 CFR 52.47(a)(1)(ii) requires applicants for a standard design certification to provide demonstration of compliance

with any technically relevant portions of the Three Mile Island Requirements set forth in 10 CFR 50.34(f). 10 CFR 50.34(f)(2)(ix) requires a system for hydrogen control that can provide with reasonable assurance that uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel-clad metal-water reaction, or that the postaccident atmosphere will not support hydrogen combustion.

In SECY-90-016, the staff recommended that the Commission approve the staff's position that the requirements of 10 CFR 50.34(f)(2)(ix) remain unchanged for evolutionary LWRs. In its June 26, 1990 SRM, the Commission approved the staff's position.

To comply with 10 CFR 50.34(f)(2)(ix), the ABWR will have an inert atmosphere during normal operation. The inert containment prevents hydrogen combustion and/or detonations from occurring.

Potential Impact Number: 25312 **SRP Section Number:** 19.2

Document: FINAL SER ABWR CH 19
Title: SEVERE ACCIDENTS

Summary:

19.2.4 Containment Performance: The NRC approach for ensuring containment survivability from severe accident challenges consists of requiring inclusion of accident prevention and consequence mitigation features and the containment performance goal (CPG). The CPG ensures that the containment would perform its function in the face of most severe accident challenges and that the design (including its mitigation features) would be adequate if called upon to mitigate a severe accident. Two alternative CPGs were identified in SECY-90-016: a conditional containment failure probability (CCFP) of 0.1 or a deterministic containment performance goal that offers comparable protection. In its June 26, 1990, SRM, the Commission approved the use of a 0.1 CCFP as a basis for establishing regulatory guidance for evolutionary LWRs. Two definitions of containment failure were discussed in SECY-91-309, "Draft Safety Evaluation Report on the General Electric Boiling Water Reactor Design Covering Chapter 19 of the Standard Safety Analysis Report, Response to Severe Accident Policy Statement." These include a CCFP based on a structural integrity definition and on a dose definition. For internal events, Section 19.1.3.5.1 of this report provides the results of

the CCFP analyses and concludes that the ABWR design limits the CCFP to approximately 0.1. The treatment of external events for the ABWR is discussed in Section 19.1.2 of this report. The Commission directed that the use of a 0.1 CCFP should not be imposed as a requirement, and that the use of the CCFP should not discourage accident prevention. Therefore, the staff's proposed applicable regulation for containment performance is as follows: "The standard design must include design features to limit the conditional containment failure probability for the more likely severe accident challenges."

Section 19.1.3.5 of this report provides the staff's analysis of the design features that contribute to limiting the CCFP. The severe accident phenomena that are mitigated by these design features are evaluated in Sections 19.2.3.3 and 19.2.6 of this report. Based on the evaluations in these sections, the staff concludes that the acceptance criteria in SECY-90-016, SECY-93-087, and the staff's proposed applicable regulation for containment performance have been met.

19.1.3.5.1 Containment Performance and CCFP: The results of the Level 2 and 3 portions of the ABWR PRA indicate that the ABWR containment is quite robust and able to accommodate severe accidents with a low attendant probability of containment failure. In assessing the probability of containment failure, two alternative definitions of containment failure were considered: (1) loss of containment structural integrity, and (2) releases that result in doses of 25 rem or greater at a distance of 0.8 km (0.5 mile) from the reactor.

Both the GE and the staff estimates of the ABWR CCFP are within the Commission's containment performance goal (0.10) using either containment failure definition. Combined with the core damage frequency estimate from the Level 1 portion of the PRA (1.6E-7), this results in an extremely low likelihood of containment failure in absolute terms (i.e., on the order of 1E-8 to 1E-9 per year). GE's estimate of CCFP using the structural integrity definition is 0.005. The staff's estimate is 0.026, and is higher due to the treatment of LOCAs outside containment as containment failures, and the higher loads associated with DCH in the staff's assessment. GE's estimate of CCFP based on the dose definition is 0.002. In contrast, the staff's estimate of CCFP using the dose definition of containment failure is about 0.1. The staff's estimate is higher because of primarily (1) a higher source term used by the staff to represent scenarios with vented containment and suppression pool scrubbing, and (2) the use of the Melcor Accident Consequence Code System (MACCS) code for the consequence in calculations. However, the staff concludes that CCFP does not exceed 0.1, even under conservative staff assumptions, and therefore, meets the Commission's safety goal.

APPENDIX II
Potential Impacts

Potential Impact Number: 25313 **SRP Section Number:** 19.2

Document: FINAL SER CE80 CH 19
Title: SEVERE ACCIDENTS

Summary:

19.2.4 Containment Performance The NRC approach for ensuring containment survivability from severe accident challenges consists of requiring inclusion of accident prevention and consequence mitigation features and the CPG. The CPG ensures that the containment would perform its function in the face of most severe-accident challenges and that the design (including its mitigation features) would be adequate if called upon to mitigate a severe accident. Two alternative CPGs were identified in SECY-90-016: a CCFP of 0.1 or a deterministic CPG that offers comparable protection. In its June 26, 1990, SRM, the Commission approved the use of the 0.1 CCFP as a basis for establishing regulatory guidance for evolutionary ALWRs. In assessing the probability of containment failure, two definitions of containment failure were considered. These include a CCFP based on structural integrity and on a dose definition. The Commission directed that the use of a 0.1 CCFP should not be imposed as a requirement, and that the use of the CCFP should not discourage accident prevention. Therefore, the staff's proposed applicable regulation for containment performance is as follows: "The standard design must include design features to limit the CCFP for the more likely severe accident challenges." Section 19.1.3.2 of this chapter provides the staff's analysis of the design features that contribute to limiting the CCFP. The severe accident phenomena that are mitigated by these design features are evaluated in Sections 19.2.3.3 and 19.2.6 of this chapter. Based on the evaluations in these sections, the staff concludes that the acceptance criteria in SECY-90-016, SECY-93-087, and the staff's proposed applicable regulation for containment performance have been met.

19.1.3.2.1 Conditional Containment Failure Probability: In assessing the probability of containment failure, two alternative definitions of containment failure were considered: (1) loss of containment structural integrity (i.e., the structural integrity definition) and (2) releases that result in significant offsite doses (i.e., the dose definition). Rather than attempt to define a "large release," the staff used the EPRI criterion of 0.25 Sv (25 rem) at 0.8 km (0.5 miles) from the reactor as the definition of containment failure in the latter case. This is judged to bound any reasonable definition of large radiological release.

The staff concludes that the estimated CCFP for the System 80+ design satisfies the Commission's CPG (0.10). Specifically, within the 24-hour period after core damage, which is the focus of the CPG, the probability of containment failure (using either the

structural integrity or the dose definition of containment failure) is below the goal. The probability of containment failure is somewhat higher when failures beyond 24 hours are included; however, CCFP remains less than the goal using the dose definition of failure (3 percent), and is only slightly higher than the goal (11 percent) using the structural definition of failure. In SECY-90-016, the staff stated that in view of the low probability of accidents that would challenge the integrity of the containment, the CCFP for evolutionary designs should not exceed "approximately" 0.1. Furthermore, in the related SRM, the Commission directed that the CCFP objective of 0.1 should not be imposed as a requirement in and of itself. In view of the approximate nature of the CPG, the recognition that PRA results, particularly bottom-line numbers, contain considerable uncertainties, and the fact that the majority of containment failures reflected in the 11-percent CCFP estimate are late, containment basemat melt-throughs rather than releases to the atmosphere, the staff concludes that the System 80+ design satisfies the Commission's CPG.

Potential Impact Number: 25314 **SRP Section Number:** 19.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY-93-087 the staff endorsed the use of a containment performance goal for evolutionary plants of either a conditional containment failure probability (CCFP) of 0.1 or an equivalent deterministic goal as had previously been recommended in SECY-90-016. The staff also noted that this position was approved by the Commission in its June 26, 1995 SRM for SECY-90-016.

This SECY also provided insight into the staff's containment performance goals recommendations. The staff stated they recognized the limitations of a CCFP approach. Specifically, as accident prevention is increased by decreasing core damage frequency, the ability of the containment to withstand events of even lower probability becomes less clear. The limitations of a CCFP approach are also evident in the uncertainties prevalent within a probabilistic risk assessment (PRA). In spite of its limitations, the staff believes the CCFP approach ensures that evolutionary ALWRs maintain a balance between accident prevention and consequence mitigation.

APPENDIX II
Potential Impacts

Potential Impact Number: 25316 **SRP Section Number:** 19.2

Document: FINAL SER ABWR CH 19
Title: SEVERE ACCIDENTS

Summary:

The Containment Overpressure Protection System (COPS) passively relieves containment pressurization before containment pressure reaches ASME Service Level C limits. This system provides for a controlled release through a containment vent pathway with fission product scrubbing provided by the suppression pool. With respect to core-concrete interaction (CCI), the COPS prevents catastrophic overpressurization failure of the containment for severe accident sequences involving prolonged periods of CCI. The COPS ensures that containment pressurization resulting from CCI does not exceed the ASME Service Level C limit of 666.9 kPa gage (97 psig), as the actuation setpoint is 617.8 kPa gage (90 psig)

In SECY-93-087, the staff concluded that the evolutionary light water reactors should 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. In addition, designers should ensure that the containment capability has a margin to accommodate uncertainties in the environmental conditions from core-concrete interactions.

The staff concluded that twenty-four hours was an appropriate time period based on sufficient time to allow for decay of fission products, operator intervention, utilization of accident management strategies, fission product deposition in the containment through natural mechanisms, and offsite protective measures. It was developed as a guideline and not a strict criterion in recognition of the uncertainties in severe accident progression and phenomenology.

The staff did not rely on any one specific sequence or scenario performed by GE using the MAAP-ABWR code nor by the staff's contractor (Sandia National Laboratories) in determining whether the ABWR met the criterion in SECY-93-087 for ensuring that containment conditions do not exceed Service Level C for approximately 24 hours from CCI. Rather, the staff evaluated the range of results provided by these codes, with due consideration of the uncertainties inherent within them, and the capability of the design to extend the time period to COPS actuation through intervention. The a-c independent

water addition system (ACIWA) is fundamental to prolonging the period to COPS actuation. Once COPS is actuated, containment pressurization is relieved through a controlled pathway that takes advantage of scrubbing by the suppression pool. The staff recognizes that there are sequences in which COPS actuation in under 24 hours is required to maintain containment stresses below ASME Service Level C limits. The staff concludes that the ABWR design meets the criterion when use of the mitigation systems incorporated into the design is factored in, such as the lower drywell flooding (LDF) and ACIWA system.

Potential Impact Number: 25317 **SRP Section Number:** 19.2

Document: FINAL SER CE80 CH 19
Title: SEVERE ACCIDENTS

Summary:

Paragraph (3)(iv) of 10 CFR 50.34(f) requires one or more dedicated containment penetrations, equivalent in size to a single 0.91 m (3 ft) diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered containment vent system. This requirement is intended to ensure provision of a containment vent design feature with sufficient safety margin well ahead of a need that may be perceived in the future to mitigate the consequences of a severe accident situation.

ABB-CE shows that the containment is sufficiently robust to not require venting before 24 hours. However, to further improve containment performance, the System 80+ containment is equipped with two 7.6-cm (3.0-in.) diameter hydrogen purge vents which can be used to relieve containment pressure before containment pressure reaches ASME Code Service Level C. With respect to CCI, the vent could be used to prevent catastrophic overpressurization failure of the containment for severe-accident sequences involving prolonged periods of CCI. The hydrogen purge vents are capable of opening when exposed to an internal pressure corresponding to ASME Code Service Level C, of 972 kPa (141 psia) at a temperature of 177 C (350 F), and can be powered by the AAC source (i.e., CTG).

The staff did not rely on any one specific sequence or scenario performed by ABB-CE using the MAAP 3 code nor by the staff's contractor (BNL) in determining whether the System 80+ design conformed to the criterion in SECY-93-087 for ensuring that containment conditions do not exceed Service Level C for approximately 24 hours.

Rather, the staff evaluated the range of results provided by these codes, with due consideration of the uncertainties inherent within them, and the capability of the design to extend the time period to containment overpressurization. The cavity flooding system (CFS) and containment spray system (CSS) are fundamental to prolonging the period to containment overpressurization or melt-through of the containment liner.

The staff concludes that the System 80+ design conforms to the criterion when the mitigation systems incorporated into the design, such as the CFS and CSS, are considered.

Potential Impact Number: 25318 **SRP Section Number:** 19.2

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY-93-087 the staff restates the containment vent positions from SECY-90-016. Although no changes to the original positions were stated, SECY-93-087 provides a more general insight into the NRC's current approach to the issue of containment vents. Specifically, SECY-93-087 states that the staff considers the containment vent as one of many plant systems that can be used to mitigate the consequences of an accident. If acceptable analysis indicate that a vent would not be needed to meet the severe accident criteria, such as a containment performance goal, the staff would not propose to implement a vent requirement. SECY-93-087 also provides the following as an acceptable containment performance deterministic goal: "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."

Potential Impact Number: 25319 **SRP Section Number:** 6.1.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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Summary:

ABB-CE 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 (recommended in RG 1.44), "Control of the Use of Sensitized Stainless Steel," has been used by ABB-CE since the mid-1970s for verifying non-sensitization of austenitic stainless steel materials and weldments. ABB-CE 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 ABB-CE'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 tests for ASTM A-262, Practice A or E.

Potential Impact Number: 25320 **SRP Section Number:** 19.2

Document: FINAL SER ABWR CH 19
Title: SEVERE ACCIDENTS

Summary:

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," the staff recommended that the Commission approve the position that both the evolutionary and passive LWR designs meet the following criteria: (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 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. In addition, they must ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions from core-concrete interactions. In its July 21, 1993 SRM, the Commission approved the staff's position. 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."

GE incorporated numerous features in the ABWR to help mitigate the effects of core-concrete interaction. The following features were judged by the staff as being most important: a large lower drywell floor area with minimal obstructions to the spreading of core debris, an LDF system, an ac-independent water addition system, use of sacrificial basaltic concrete for the lower drywell floor, a thick reactor pedestal wall, and a COPS.

Potential Impact Number: 25321 **SRP Section Number:** 19.2

Document: FINAL SER CE80 CH 19
Title: SEVERE ACCIDENTS

Summary:

In SECY-93-087, the staff recommended that the Commission approve the position that both the evolutionary and passive LWR designs meet the following criteria: (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 that the best-estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed ASME Code Service Level C limits for steel containments or factored load category for concrete containments, for approximately 24 hours. In addition, ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions from CCIs. In its July 21, 1993, SRM, the Commission approved the staff's position.

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 CCIs do not exceed ASME Code Service Level C limits for steel containments or factored load category for concrete containments, for approximately 24 hours."

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ABB-CE has incorporated many features in the System 80+ design to help mitigate the effects of CCI. The following features were judged by the staff as being most important: a large reactor cavity floor area with minimal obstructions to the spreading of core debris, a CFS, an external means of adding water to the reactor cavity, use of sacrificial limestone-based concrete for the reactor cavity floor and robust reactor vessel support structures (e.g., corbels and lower cavity walls).

Potential Impact Number: 25322 **SRP Section Number:** 6.1.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

ABB-CE took exception to the recommendations in Position C.2 of RG 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels," for controls imposed on preheat temperatures for welding ferritic steels. These controls would 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. ABB-CE, in a letter of November 24, 1992, stated 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 ABB-CE'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. This approach should provide adequate assurance that low-alloy steel weldments will not be developing cracking due to hydrogen. On this basis, DSER Open Item 5.2.3-1 is resolved.

Potential Impact Number: 25323 **SRP Section Number:** 19.2

Document: FINAL SER ABWR CH 19
Title: SEVERE ACCIDENTS

Summary:

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In SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," the staff concluded that evolutionary LWR designs should include a depressurization system and cavity design features to contain ejected core debris. In its June 26, 1990, SRM, the Commission approved the staff's position that evolutionary LWR designs include a depressurization system and cavity design to contain core debris. In addition, the Commission stated that the cavity design, as a mitigating feature, should not unduly interfere with operations including refueling, maintenance, or surveillance activities. In SECY-93-087, Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, the staff recommended that the Commission approve the general criteria that the evolutionary LWR designs provide a reliable depressurization system and cavity design features to decrease the amount of ejected core debris that reaches the upper containment. In its July 21, 1993, SRM, the Commission approved the staff's position.

Based on engineering judgment, the staff believes that examples of cavity design features that will decrease the amount of ejected core debris that reaches the upper containment include ledges or walls that would deflect core debris and an indirect path from the lower drywell to the upper containment. The staff position within SECY-93-087 evolved from the staff position in SECY-90-016 and forms the basis for the staff's review and evaluation. Therefore, the staff's proposed applicable regulation for high-pressure core melt ejection is as follows: "The standard design must provide a reliable means to depressurize the reactor coolant system 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."

The ADS of the ABWR is provided with a reliable nitrogen supply and dc power to ensure its operability. The containment design of the ABWR is expected to decrease the amount of ejected core debris that reaches the upper drywell. This decrease is anticipated through the following: (1) capture and trapping of debris in the lower drywell, (2) impaction and removal of core debris as it is transported between the lower and upper drywell, and (3) division of exiting core debris and gas from the lower drywell into both the upper drywell and wetwell. Based on the above, the staff concludes that the criteria of SECY-93-087 and the staff's proposed applicable regulation for high-pressure core melt ejection have been met.

Potential Impact Number: 25324 **SRP Section Number:** 19.2

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Document: FINAL SER CE80 CH 19
Title: SEVERE ACCIDENTS

Summary:

In SECY-90-016, the staff concluded that evolutionary LWR designs should have a depressurization system and cavity design features to contain ejected core debris. In its June 26, 1990, SRM, the Commission approved the staff's position that evolutionary LWR designs should have a depressurization system and cavity design to contain core debris. In addition, the Commission stated that the cavity design, as a mitigating feature, should not unduly interfere with such operations as refueling, maintenance, or surveillance. In SECY-93-087, the staff recommended that the Commission approve the general criteria that the evolutionary LWR designs should have a reliable depressurization system and cavity design features to decrease the amount of ejected core debris that reaches the upper containment. In its July 21, 1993, SRM, the Commission approved the staff's position. On the basis of engineering judgment, the staff believes that examples of cavity design features that will decrease the amount of ejected core debris reaching the upper containment include ledges or walls that would deflect core debris and an indirect path from the reactor cavity to the upper containment. The staff position in SECY-93-087 evolved from its position in SECY-90-016 and therefore, is the basis for the staff's review and evaluation.

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."

The RDS has a reliable dc power supply to ensure its operability. The containment design is expected to decrease the amount of ejected core debris that leaves the reactor cavity. This decrease is anticipated through (1) capture and trapping of debris in the reactor cavity and (2) impaction and removal of core debris as it is transported between the reactor cavity and upper containment. On this basis, the staff concludes that the criteria of SECY-93-087 and the staff's proposed applicable regulation for HPME have been met.

Potential Impact Number: 25325 **SRP Section Number:** 19.2

Document: SECY 90-016

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Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016 the staff recommended that ALWR designs include a depressurization system and cavity design features to contain ejected core debris. In an SRM dated June 26, 1990, the Commission approved the staff's recommendation and also noted that the cavity design, as a mitigating feature, should not unduly interfere with operations including refueling, maintenance, or surveillance activities.

Potential Impact Number: 25326 **SRP Section Number:** 19.2

Document: FINAL SER CE80 CH 19
Title: SEVERE ACCIDENTS

Summary:

In SECY-93-087, the NRC requires that, during the review of the credible severe-accident scenarios for ALWRs, the staff evaluate the ALWR design certification applicant's identification of the equipment needed to perform mitigative functions and the conditions under which the mitigative systems must operate.

Beyond design basis events can generally be categorized into in-vessel and ex-vessel severe accidents. The environmental conditions resulting from these events are generally more limiting than those from design bases events. The NRC established a criterion to provide a reasonable level of confidence that the necessary equipment will function in the severe accident environment for the time span for which it is needed. This criterion is commonly referred to as "equipment survivability" and is fundamentally different from equipment qualification.

The applicable criteria for equipment, both mechanical and electrical, required for recovery from in-vessel severe accidents are provided in 10 CFR 50.34(f). 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.

The applicable criteria for equipment, both electrical and mechanical, required to mitigate the consequences of ex-vessel severe accidents is discussed in the Equipment Survivability section of SECY-90-016. In its SRM of June 26, 1990, relating to SECY-90-016, the Commission approved the staff's position that features provided only (not required for DBAS) for severe-accident protection (prevention and mitigation) need not be subject to the 10 CFR 50.49 environmental qualification requirements; 10 CFR Part 50, Appendix B quality assurance requirements; and 10 CFR Part 50, Appendix A redundancy/diversity requirements. The reason for this judgement is that the staff does not believe that severe core damage accidents should be DBAs in the traditional sense that DBAs have been treated in the past. However, 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 cases where safety-related equipment (equipment provided for DBAs) is relied upon to cope with severe accident situations, there should be reasonable assurance that this equipment will survive accident conditions for the period that is needed to perform its intended function. 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."

Potential Impact Number: 25327 **SRP Section Number:** 19.2

Document: FINAL SER ABWR CH 19
Title: SEVERE ACCIDENTS

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Summary:

In its SRM of June 16, 1990, relating to SECY-90-016, the Commission approved the staff position that features provided only (not required for design basis accidents) for severe-accident protection (prevention and mitigation) need not be subject to the 10 CFR 50.49 environmental qualification requirements; 10 CFR Part 50, Appendix B quality assurance requirements; and 10 CFR Part 50, Appendix A redundancy/diversity requirements. The reason for this judgement is that the staff does not believe that severe core damage accidents should be design basis accidents in the traditional sense that DBAs have been treated in the past. 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."

The applicable criterion for equipment, both mechanical and electrical, required for recovery from in-vessel severe accidents is provided in 10 CFR 50.34(f).

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. These regulations collectively indicate the need to perform a systematic evaluation of all equipment, both electrical and

mechanical, and instrumentation to ensure its survivability for intervention into an in-vessel severe accident.

The applicable criteria for equipment, both electrical and mechanical, required to mitigate the consequences of ex-vessel severe accidents is discussed in the Equipment Survivability section of SECY-90-016. This section indicates that features provided only (not required for design basis accidents) for severe-accident protection (prevention and mitigation) need not be subject to the 10 CFR 50.49 environmental qualification requirements; 10 CFR Part 50, Appendix B quality assurance requirements; and 10 CFR Part 50, Appendix A redundancy/diversity requirements. The reason for this judgement is that the staff does not believe that severe core damage accidents should be design basis accidents in the traditional sense that DBAs have been treated in the past.

However, 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 cases where safety-related equipment (equipment provided for DBAs) is relied upon to cope with severe accident situations, there should be reasonable assurance that this equipment will survive accident conditions for the period that is needed to perform its intended function.

Potential Impact Number: 25328 **SRP Section Number:** 9.3.4

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

Section 5.2.3 of the SER provides a discussion related to reactor coolant chemistry and material selection for compatibility. ABB-CE stated that the Chemical and Volume Control System (CVCS) maintains the reactor coolant chemistry within the limits defined in Electric Power Research Institute (EPRI) guidelines (EPRI, "PWR Primary Water Chemistry Guidelines: Revision 2," Report NP-7077, November 1990). In addition, ABB-CE indicated that the reactor coolant chemistry is consistent with EPRI Utility Requirements Document (URD) (EPRI, "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, ALWR Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991). However, the water chemistry stated in CESSAR-DC Section 5.2.3.4.1.2.1 was not consistent with EPRI NP-7077 and EPRI URD.

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ABB-CE subsequently identified the chemistry limits in CESSAR-DC Section 5.2.3.4.1.2.1 as applying to cleaning water used on NSSS components fabricated with austenitic stainless steel and, accordingly, would not be consistent with reactor coolant water chemistry limits. These cleaning water chemistry limits provide contamination control during component fabrication, shipment, and storage as required by RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants." Also, to prevent halide-induced intergranular corrosion, which would occur with the presence of significant quantities of oxygen, flushing water is inhibited by additions of hydrazine. The staff finds that the cleaning and flushing water does not have to conform to the chemistry limits of reactor coolant water.

In addition, EPRI URD supplements the guidelines in EPRI NP-7077 as follows for the makeup water storage tank:

Conductivity at 25C-0.2 S/cm
Oxygen-0.100 ppm (max)
Chloride-0.15 ppm (max)
Fluoride-0.15 ppm (max)
Suspended solids*-1.0 ppm (max)

*Solids concentration is determined by filtration through a filter having a 0.45-micron pore size.

ABB-CE revised the primary water chemistry in CESSAR-DC Table 9.2.3-1 to meet or exceed the guidelines of EPRI URD. On this basis, DSER Open Item 5.2.3-7 is resolved.

Potential Impact Number: 25329 **SRP Section Number:** 9.3.4

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The EPRI proposed water chemistry design basis requirements for PWR plant systems in accordance with EPRI NP-5960, PWR Primary Water Chemistry Guidelines, Revision 1, and its subsequent revisions; EPRI NP-6239, PWR Secondary Water Chemistry Guidelines, Revision 2, and its subsequent revisions; as supplemented by the

guidelines in Table 1.3 of the FSER are acceptable. The applicant for FDA/DC or a combined license should reference EPRI NP-7077, PWR Primary Water Chemistry Guidelines: Revision 2, November 1990 instead of EPRI NP-5960, Revision 1. The staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license.

Potential Impact Number: 25342 **SRP Section Number:** 4.5.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 4

Summary:

Criterion 4 -- Environmental and dynamic effects design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Potential Impact Number: 25343 **SRP Section Number:** 4.5.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 31

Summary:

Criterion 31 -- Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when

stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Potential Impact Number: 25344 **SRP Section Number:** 4.5.1

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

In CESSAR-DC Section 4.5.1.1, ABB-CE states that Inconel X-750 (AMS 5698B and AMS 5699B) will be used. ABB-CE uses these materials for springs that will be subjected to elevated temperatures and must resist relaxation. ABB-CE affirms that prototype testing of the CEDM and extensive previous experience in existing reactors have demonstrated the acceptability of these materials for their intended use. The springs are not part of the primary pressure boundary; therefore, Section III of the ASME Code and RG 1.85 do not require them to be listed as accepted.

AMS 5698 and 5699 (Inconel X-750 materials) are drawn from hot-finished wire rod that has been previously ground or has had surface preparation (other than by pickling) to remove seams or other injurious surface imperfections. The wire is heat-treated at 2100 degrees F before being reduced to size. The staff finds that these materials should perform their function when supported by the maintenance and inspections prescribed by ABB-CE.

Potential Impact Number: 25347 **SRP Section Number:** 4.5.1

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

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ABB-CE has demonstrated that the properties of materials selected for the CEDM components exposed to reactor coolant satisfy Appendix I to Section III, and Parts A, B, and C of Section II, of the ASME Boiler and Pressure Vessel Code. ABB-CE selected all materials in this system for their compatibility with the reactor coolant. Materials were extensively tested in a CEDM assembly that exceeded lifetime requirements. Moreover, all materials have served satisfactorily in many other NRC-licensed operating reactors for more than 20 years.

The mechanical properties of structural materials selected by ABB-CE for control rod system components exposed to the reactor coolant conform with the staff position that the yield strength of cold worked austenitic stainless steel should not exceed 90 Ksi (620.5 MPa). ABB-CE has met the guidelines of RG 1.85, "Code Case Applicability i ASME Section III Materials," by using construction materials that are approved by ASME Code cases.

Analyst Note: While most of the above staff findings appear consistent with current SRP Section 4.5.1, it should be noted that SRP Section 4.5.1 Acceptance Criteria (specific criterion II.1) does not include Part C of Section II of the Code although Review Procedures and Evaluation Findings do reference Part C.

Potential Impact Number: 25349 **SRP Section Number:** 4.5.1

Document: FINAL SER ABWR CH 4
Title: REACTOR

Summary:

The SSAR states that the properties of the materials selected for the ABWR CRD mechanism will be equivalent to those given in Appendix I to Section III of the American Society of Mechanical Engineers (ASME) Code; Parts A, B, and C of Section II of the ASME Code; and RG 1.85. No cold-worked austenitic stainless steels except those with controlled hardness or strain are employed in the CRD system. All materials used in this system will be selected for their compatibility with the reactor coolant as described in Articles NB-2160 and NB-3120 of the ASME Code. The materials selected as identified in SSAR Section 4.5.1 will be resistant to stress corrosion in a BWR environment. The controls imposed on the austenitic stainless steel of the CRD mechanism conform to the recommendations of RGs 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," (Revision 3) and 1.44, "Control of the Use of Sensitized Stainless Steel," (Revision 0).

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Analyst Note: While most of the above staff findings appear consistent with current SRP Section 4.5.1, it should be noted that SRP Section 4.5.1 Acceptance Criteria (specific criterion II.1) does not include Part C of Section II of the Code although Review Procedures and Evaluation Findings do reference Part C.

Potential Impact Number: 25350 **SRP Section Number:** 4.5.2

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

As discussed in CESSAR-DC Section 4.5.1, ABB-CE proposes to allow the continued reference to ASTM A-708 for the purpose of maintaining the qualifications of older weld procedures. ASTM A-262, Practice A or E (recommended in RG 1.44), has been used by ABB-CE since the mid-1970s for verifying non-sensitization of austenitic stainless steel materials and weldments. ABB-CE qualified weld procedures developed prior to the mid-1970s and used A-708 in their qualification. These weld procedures are still in use. There have been no IGSCC failures involving austenitic stainless steel weldments in ABB-CE's NSSS units. In addition, the staff allowed in SRP Section 4.5.1, Paragraph III.2, ASTM A-708 as an acceptable test for ASTM A-262, Practice A or E. On this basis, DSER Open Item 4.5.2-2 is resolved.

Potential Impact Number: 25354 **SRP Section Number:** 4.5.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 4

Summary:

Criterion 4 -- Environmental and dynamic effects design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and

discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Potential Impact Number: 25355 **SRP Section Number:** 4.5.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 10

Summary:

Criterion 10 -- Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Potential Impact Number: 25369 **SRP Section Number:** 6.8

Document: NUREG 0737
Title: CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Summary:

This NUREG contains several TMI action items that are potentially applicable to PWR reactor coolant system depressurization systems. The subject items are as follows:

II.D.1 - Conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents.

II.D.3 - Reactor coolant system relief and safety valves shall 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.

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II.G.1 - The power supply for PORVs and PORV Block Valves shall meet the following requirements: (1) Motive and control components of the PORVs and PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when offsite power is not available, (2) Motive and power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

II.K.2.20 - Provide an analysis which shows the plant response to a small-break LOCA during which the reactor coolant system is repressurized to the PORV setpoint with subsequent failure of the PORV to close (B&W plants only).

II.K.3.1 - All PWR licensees should provide a system that uses the PORV block valve to protect against a small-break LOCA. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened.

II.K.3.2 - (1) The licensee should submit a report for staff review documenting the various actions taken to decrease the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV) and show how those actions constitute sufficient improvements in reactor safety. (2) Safety-valve failure rates based on past history of the operating plants designed by the specific nuclear steam supply system (NSSS) vendor should be included in the report submitted in response to (1) above.

II.K.3.3 - All licensees should report safety valve and relief valve failures and challenges. (Note: no amplification of this item is given, it is merely listed as applicable to all plants.)

II.K.3.7 - Most overpressure transients should not result in opening of the PORV. Therefore, licensees should document that the PORV will open in less than 5% of all anticipated overpressure transients using the revised setpoints and anticipatory trips for the range of plant conditions which might occur during a fuel cycle (B&W plants only).

Potential Impact Number: 25370 **SRP Section Number:** 6.8

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

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Summary:

This NUREG-0660 item required all operating PWRs to document the action to be taken to decrease the probability of a small-break LOCA caused by a stuck-open PORV. OL applicants with PORVs were also required to complete this item. Clarifications affecting both groups of plants were issued in NUREG-0737. This item was RESOLVED, requirements were issued.

Potential Impact Number: 25371 **SRP Section Number:** 6.8

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.34(f) -(1)(iv), (2)(x), (2)(xi), (2)(xx)

Summary:

Several TMI action items applicable to RCS depressurization systems are contained in the following paragraphs of 10 CFR 50.34(f):

(1)(iv) Perform an analysis of the probability of a small-break loss-of- coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCA's from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWR's only). (TMI Action Plan item II.K.3.2)

(2)(x) Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. (TMI Action Plan item II.D.1)

(2)(xi) Provide direct indication of relief and safety valve position (open or closed) in the control room. (TMI Action Plan item II.D.3)

(2)(xx) Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources. (Applicable to PWR's only). (TMI Action Plan item II.G.1)

Potential Impact Number: 25372 **SRP Section Number:** 6.8

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

Issue 70: PORV and Block Valve Reliability. Power-operated relief valves (PORVs) and block valves were originally designed as non-safety components in the reactor pressure control system for use only when plants are in operation; the block valves were installed because of expected leakage from the PORVs. Neither valve type was needed to safely shut down a plant or mitigate the consequences of accidents. In 1983, the staff determined that PORVs were relied on to mitigate design-basis SGTR accidents and questioned the acceptability of relying on non-safety-grade components to mitigate DBAs. Issue 70, in NUREG-0933, addressed the assessment of the need for improving the reliability of PORVs and block valves. In CESSAR-DC Section 20.2.18, ABB-CE states that the design purpose of PORVs is to prevent challenges to spring-operated safety valves and to provide rapid depressurization capability. Older ABB-CE plants had PORVs, but the newer ABB-CE System 80+ design does not include PORVs and block valves. Instead, the System 80+ includes a SDS, which is a safety-grade system, providing venting and rapid depressurization capability for mitigation of beyond-design- basis accidents. In Section 6.7 of this report, the staff approved the design of the SDS.

Potential Impact Number: 25373 **SRP Section Number:** 6.8

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

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PORVs and block valves were originally designed as non-safety components in the reactor pressure control system for use only when plants are in operation. The block valves were installed because of expected leakage from the PORVs. Neither the PORVs nor the block valves were required to safely shut down a plant or mitigate the consequences of accidents. RSB determined in 1983 that PORVs are relied upon to mitigate a design basis SGTR. However, the acceptability of relying on non-safety grade PORVs to mitigate a design basis accident (e.g., an SGTR) was questioned by RSB. This issue addressed the assessment of the need for improving the reliability of PORVs and block valves.

Potential Impact Number: 25374 **SRP Section Number:** 6.8

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

Issue II.K.3(2): Report on Overall Safety Effect of PORV Isolation System

Issue II.K.3(2), in NUREG-0933, addressed requiring applicants to document the action to be taken to decrease the probability of a SBLOCA caused by a stuck-open PORV. The design purpose of PORVs is to prevent RCS overpressure and to reduce challenges to the safety valves for DBE. The requirements were issued in NUREG-0737.

ABB-CE states in CESSAR-DC Section 20.2.121 that the System 80+ design does not include PORVs and, therefore, this issue is not applicable to the System 80+ design. However, to satisfy the staff requirements in SECY-90-016 regarding rapid depressurization for mitigation of beyond-design-basis events, ABB-CE includes the SDS for the System 80+ design. The SDS is a safety grade system and is manually operated to mitigate beyond- design-basis events. The SDS has two trains, each with two MOVs in series. These valves are manually operated from the control room and the opening of one train would result in a medium LOCA with the plant response being the same as for any other medium LOCA. In Section 6.7 of this report, the staff approved the SDS.

The general ways in which an SDS LOCA might occur is the mechanical failure of the series valves or the inadvertent opening of the valves. As discussed under Issue II.D.3 in this section and Section 7.7 of this report, there is position indication for these valves,

so the opening of an SDS train would be indicated to the control room operators. The probability that an SDS valve train will fail open and a medium LOCA will occur is discussed in Section 19.3.3.1 of CESSAR-DC and the staff concluded that the ABB-CE evaluation was acceptable.

Therefore, Issue II.K.3(2) is resolved for the System 80+ design.

Potential Impact Number: 25375 **SRP Section Number:** 6.8

Document: NRC GENERIC LETTER 90-06
Title: 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,'

Summary:

The staff, in its evaluation, determined that over a period of time the role of PORVs has changed such that PORVs are now relied upon by many Westinghouse, B&W, and CE designed plants with PORVs to perform one, or more, of the following safety-related functions:

1. Mitigation of a design-basis steam generator tube rupture accident, 2. Low-temperature overpressure protection of the reactor vessel during startup and shutdown, or 3. Plant cooldown in compliance with Branch Technical Position RSB 5-1 to SRP 5.4.7, "Residual Heat Removal (RHR) System."

Where PORVs are used or could be used to perform one, or more, of the safety-related functions identified above or to perform any other safety-related function that may be identified in the future, it is appropriate to reconsider the safety classification of PORVs and the associated block valves. For certain PWR plants receiving an operating license in recent years, the staff has required these valves to be classified as safety-related components if they perform one, or more, safety-related functions.

Section three of Enclosure A identifies staff positions/requirements resulting from the resolution of Generic Issue 70, PORV and Block Valve Reliability.

3. IMPROVEMENTS TO ALL PORVs AND BLOCK VALVES

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3.1 Operating PWR Plants and Construction Permit Holders

Based on the analysis and findings for GI-70, the staff concludes that the following actions should be taken to improve the reliability of PORVs and block valves:

1. Include PORVs and block valves within the scope of an operational quality assurance program that is in compliance with 10 CFR Part 50, Appendix B. This program should include the following elements: a. The addition of PORVs and block valves to the plant operational Quality Assurance List. b. Implementation of a maintenance/refurbishment program for PORVs and block valves that is based on the manufacturer's recommendations or guidelines and is implemented by trained plant maintenance personnel. c. When replacement parts and spares, as well as complete components, are required for existing non-safety-grade PORVs and block valves (and associated control systems), it is the intent of this generic letter that these items may be procured in accordance with the original construction codes and standards.

2. Include PORVs, valves in PORV control air systems, and block valves within the scope of a program covered by Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants," of Section XI of the ASME Boiler and Pressure Vessel Code. Stroke testing of PORVs should only be performed during Mode 3 (HOT STANDBY) or Mode 4 (HOT SHUTDOWN) and in all cases prior to establishing conditions where the PORVs are used for low-temperature overpressure protection. Stroke testing of the PORVs should not be performed during power operation. Additionally, the PORV block valves should be included in the licensees' expanded MOV test program discussed in NRC Generic Letter 89-10, "Safety-Related Motor Operated Valve Testing and Surveillance," dated June 28, 1989.

3. For operating PWR plants, modify the limiting conditions of operation of PORVs and block valves in the technical specifications for Modes 1, 2, and 3 to incorporate the position adopted by the staff in recent licensing actions. Attachments A-1 through A-3 are provided for guidance. The staff recognizes that some recently licensed PWR plants already have technical specifications in accordance with the staff position. Such plants are already in compliance with this position and need merely state that in their response. These recent technical specifications require that plants that run with the block valves closed (e.g., due to leaking PORVs) maintain electrical power to the block valves so they can be readily opened from the control room upon demand. Additionally, plant operation in Modes 1, 2, and 3 with PORVs and block valves inoperable for reasons other than seat leakage is not permitted for periods of more than 72 hours.

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Potential Impact Number: 25376 **SRP Section Number:** 6.8

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

In SECY 90-016 the staff concluded that the ALWR designs should include a depressurization system to mitigate the potential for high pressure core melt ejection. The staff additionally stated 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."

In the SRM for SECY 90-016 dated June 26, 1990, the Commission approved the staff's position that ALWR designs should include a depressurization system and that severe accident equipment must be designed for survivability.

Potential Impact Number: 25377 **SRP Section Number:** 6.8

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY-93-087 the staff requested that the Commission approve certain positions related to severe accidents in which molten debris could be ejected from the reactor vessel under high-pressure. One of the positions addressed design measures that could prevent a high-pressure melt ejection during a severe accident. The position indicates that evolutionary and passive ALWR designs should provide a reliable depressurization system. Additionally, the staff discussed the importance of equipment survivability and reconfirmed the SECY 90-016 position that severe accident mitigative equipment "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."

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The positions proposed by the staff were approved by the Commission in an SRM dated July 21, 1993.

Potential Impact Number: 25378 **SRP Section Number:** 6.8

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

The EPRI FSER reviewed the safety depressurization and venting system in regards to direct containment heating under section 5.4 safety injection system. The EPRI FSER states, "In revision 3 of chapter 5, EPRI requires that the safety depressurization and venting system be capable of reducing reactor coolant system pressure to 250 psig or less prior to reactor vessel melt-through, as a means to preclude containment challenges through direct containment heating (DCH). The staff concludes that this is an acceptable design objective. However, the designer should justify that the automatic depressurization system will depressurize low enough to preclude DCH. The designer should also demonstrate that the depressurization system is adequate to provide sufficient capacity to handle primary feed-and-bleed operations during a total loss-of-feedwater event and to prevent creep rupture of steam generator tubes from a postulated high-pressure core-damage event."

Potential Impact Number: 25379 **SRP Section Number:** 6.8

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

The staff concluded that the capacity of the depressurization system should be defined by DBA requirements as well as by requirements that exceed the design basis (including primary feed and bleed during a total loss of feedwater and severe-accident scenarios) and should be taken into consideration during the development of procedures for managing accidents.

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Potential Impact Number: 25380 **SRP Section Number:** 6.8

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

The RDS is designed as a manually operated safety-grade system that removes steam or water from the pressurizer through two isolation valves in each of two parallel depressurization lines to the IRWST. The RDS is designed to mitigate the consequences of a beyond-design-basis event such as a total loss of normal and emergency feedwater (TLOFW) event.

The design-basis event for determining the size of the RDS bleed valves is a TLOFW event. ABB-CE performed the analysis using a realistic version of the CEFLASH-4AS(REM) code with assumed best estimate decay heat values. The CEFLASH-4AS(REM) is documented in CEN-420P, "SBLOCA Realistic Evaluation Model". The staff determined that use of the realistic version of the CEFLASH-4AS code is acceptable because the RDS is designed to mitigate accidents beyond the design basis.

The RDS also performs an important function in mitigating a severe accident. During a core melt, the system would allow the RCS to be depressurized and reduce the possibility of a challenge to the containment, such as from direct containment heating.

The staff has reviewed the SDS (RDS) description in CESSAR-DC Section 6.7, the piping and instrumentation drawings, and the response to the staff's review questions. The SDS design is acceptable because it meets GDC 17, 19, 34, and 36 and RG 1.29.

Potential Impact Number: 25384 **SRP Section Number:** 6.8

Document: NRC GENERIC LETTER 88-20
Title: INDIVIDUAL PLANT EXAMINATION FOR SEVERE ACCIDENT VULNERABILITIES

Summary:

This GL discusses the requirement for plants to conduct systematic evaluations (PRAs) to identify any plant-specific vulnerabilities to severe accidents.

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Potential Impact Number: 25387 **SRP Section Number:** 4.5.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In Chapter 1, paragraph 5.3.1.1.5 of the URD, EPRI states that although the stated controls on grinding on stainless steel may be desirable for use on PWR components, they are not required for avoiding IGSCC. ABB-CE intends to avoid fabrication processes which would severely cold work the surface of austenitic stainless steel components. Grinding is done with resin-bonded or rubber bonded, aluminum oxide or silicon carbide wheels which were not previously used on materials other than austenitic alloys. This will offer reasonable assurance of stainless steel component integrity, and that stainless steel components in contact with reactor coolant are not likely to crack. On this basis, DSER Open Item 5.2.3-11 is resolved.

Potential Impact Number: 25388 **SRP Section Number:** 4.5.2

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In Chapter 1, paragraph 5.3.1.1.5 of the URD, EPRI states that although the stated controls on grinding on stainless steel may be desirable for use on PWR components, they are not required for avoiding IGSCC. ABB-CE intends to avoid fabrication processes which would severely cold work the surface of austenitic stainless steel components. Grinding is done with resin-bonded or rubber bonded, aluminum oxide or silicon carbide wheels which were not previously used on materials other than austenitic alloys. This will offer reasonable assurance of stainless steel component integrity, and that stainless steel components in contact with reactor coolant are not likely to crack. On this basis, DSER Open Item 5.2.3-11 is resolved.

In FSER Chapter 6, Section 6.1.1, the staff indicates that the COL applicant will perform grinding with resin-bonded or rubber-bonded aluminum oxide or silicon carbide wheels that have been used only on Type 300-series stainless alloys. Grinding wheels bonded

with rubber compositions that include halides or sulfur will not be used on austenitic stainless steels.

Potential Impact Number: 25394 **SRP Section Number:** 6.8

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

The qualification program which will be implemented for mechanical, instrumentation, and electrical equipment meets the requirements and recommendations of IEEE 344-1987 and the regulatory positions of RGs 1.61, 1.89, 1.92, 1.100, and SRP Section 3.9.3, and provides adequate assurance that such equipment will function properly under all imposed design and service loads, including the loadings imposed by the SSE, postulated accidents, and LOCAs. This program constitutes an acceptable basis for satisfying the applicable requirements of GDC 2, 4, 14, and 30 of Appendix A to 10 CFR Part 50 and Appendix A to 10 CFR Part 100 as they relate to qualification of equipment.

Potential Impact Number: 25395 **SRP Section Number:** 6.8

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

The RDS valves are closed during normal operation. These valves are motor- operated and fail in the "as is" position. The valves are designed to be operated during a station blackout (SBO). Thus, the Class 1E dc buses supply electrical power to the motor operators (through dc to ac inverters). The FMEA documented in CESSAR-DC Table 6.7-3 demonstrates that an RDS bleed path can be established in the event of a single failure in the mechanical equipment or battery banks with total SBO.

Analyst Note: The above described power supply for the RDS valves meets the requirements of GDC 17.

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Potential Impact Number: 25397 **SRP Section Number:** 6.8

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 2

Summary:

Criterion 2 -- Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Potential Impact Number: 25398 **SRP Section Number:** 6.8

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 4

Summary:

Criterion 4 -- Environmental and dynamic effects design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis

when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Potential Impact Number: 25399 **SRP Section Number:** 6.8

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 17

Summary:

Criterion 17 -- Electric power systems. An onsite electric power system 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 anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained. Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

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Potential Impact Number: 25403 **SRP Section Number:** 6.8

Document: NRC POLICY STATEMENT 50 FR 32138
Title: POLICY STATEMENT ON SEVERE REACTOR ACCIDENTS REGARDING FUTURE DESIGNS AND EXISTING PLANTS

Summary:

This statement describes the policy the Commission intends to use to resolve safety issues related to reactor accidents more severe than design basis accidents. Its main focus is on the criteria and procedures the Commission intends to use to certify new designs for nuclear power plants.

Custom designs that are variations of the present generation of LWRs will be reviewed in future construction permit applications under guidelines identified for approval or certification of standard plant designs.

Policy for New Plant Applications
1. Introduction - No new commercial nuclear reactors have been ordered in the United States since December 1978. However, the Commission has received several applications for reference design approvals that are currently under review. A reference design is one of the options in the Commission's standardization policy. When approved by the NRC staff, a reference design could be incorporated by reference in a new CP application and, ultimately, in an Operating License (OL) application. During the corresponding CP and OL reviews, the NRC staff would not duplicate that portion of its review encompassed by its reference design approval. Therefore, even in the absence of new CP applications, in order to provide guidelines for the current reference design reviews, the Commission has recognized the need to promptly establish the criteria by which new designs can be shown to be acceptable in meeting severe accident concerns. The Commission now believes that there exists an adequate basis from which to establish an appropriate set of criteria. This belief is supported by current operating reactor experience, ongoing severe accident research, and insights from a variety of risk analyses. The resultant criteria and procedural requirements are listed below.

2. Criteria and Procedural Requirements - The Commission believes that a new design for a nuclear power plant (as well as a proposed custom plant) can be shown to be acceptable for severe accident concerns if it meets the following criteria and procedural requirements:

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a. Demonstration of compliance with the procedural requirements and criteria of the current Commission regulations, including the Three Mile Island requirements for new plants as reflected in the CP Rule (10 CFR 50.34(f)):

b. Demonstration of technical resolution of all applicable Unresolved Safety Issues and the medium- and high-priority Generic Safety Issues, including a special focus on assuring the reliability of decay heat removal systems and the reliability of both AC and DC electrical supply systems;

c. Completion of a Probabilistic Risk Assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety; 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 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.

Potential Impact Number: 25422 **SRP Section Number:** 6.8

Document: NUREG 1316
Title: TECHNICAL FINDINGS AND REGULATORY ANALYSIS RELATED TO GENERIC ISSUE 70, 'EVALUATION OF POWER OPERATED RELIEF VALVE AND BLOCK VALVE RELIABILITY IN PWR NUCLEAR POWER PLANTS.'

Summary:

This NUREG describes the typical arrangement and use of PORVs and block valves. These valves are non-safety related but have been determined to fulfill some safety-related functions. The impact of a stuck open PORV on the TMI accident is reviewed.

Potential Impact Number: 25424 **SRP Section Number:** 3.9.4

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Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 4

Summary:

Criterion 4 -- Environmental and dynamic effects design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

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 control rod drives. 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, thus the applicability of GDC 4 seems clear. 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. 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).

Potential Impact Number: 25425 **SRP Section Number:** 5.4.2.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The controls to avoid stress corrosion cracking in RCPB components constructed of austenitic stainless steels include, (1) not using cold-worked austenitic stainless steel to

limit the yield strength, and (2) conforming to the recommendations of RG 1.44, "Control of the Use of Sensitized Stainless Steel," and RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Plants." The controls followed in accordance with these recommendations during material selection, fabrication, examination, and protection, prevent excessive yield strength, sensitization and contamination, and offer reasonable assurance that the austenitic stainless steel RCPB components will be in a metallurgical condition that minimizes susceptibility to stress corrosion cracking during service. These controls fulfill the requirements of GDC 4 relative to compatibility of components with environmental conditions and the requirements of GDC 14 relative to prevention of leakage and failure of the RCPB.

Potential Impact Number: 25428 **SRP Section Number:** 13.6

Document: NUREG 1267
Title: TECHNICAL RESOLUTION OF GENERIC SAFETY ISSUE A-29, NUCLEAR POWER PLANT DESIGN FOR REDUCTION OF VULNERABILITY TO INDUSTRIAL SABOTAGE.

Summary:

This report summarizes key technical findings related to Safety Issue A- 29, "Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage." The findings in this report deal with (1) an historical review of reported sabotage-related events at nuclear facilities, (2) NRC physical security requirements, (3) industry measures to prevent/mitigate sabotage, (4) design and procedural approaches that could be used to deter sabotage, (5) current NRC and industry initiatives aimed at personnel screening and selection, and (6) design considerations applicable to Advanced Light Water Reactors (ALWRs). The results reveal that insider sabotage at U.S. operating nuclear plants has not been a significant problem in the United States to date and that there are no singular design modifications or procedures that by themselves would completely eliminate or mitigate the threat of insider sabotage. Rather, it will take a combination of systematic and focused improvements in the three areas of reliable personnel, effective design features, and plant procedures developed to provide a strategy to deal with prevention of insider sabotage and to be able to mitigate adverse actions.

APPENDIX II
Potential Impacts

Potential Impact Number: 25429 **SRP Section Number:** 13.6

Document: FINAL SER ABWR CH 14
Title: INITIAL TEST PROGRAM

Summary:

The discussion of preoperational and startup tests were limited to those systems and components within the scope of, or directly related to, the ABWR standard plant. Other testing, with respect to utility specific aspects of the plant will be necessary to satisfy certain ABWR requirements. Testing of such systems and components should be adequate to demonstrate conformance to such requirements as defined throughout the specific chapters of the SSAR. Below are systems that may require such testing:

(1) electrical switchyard and equipment
(2) the site security plan
(3) personnel monitors and radiation survey instruments
(4) the automatic dispatcher control system (if applicable)

Potential Impact Number: 25430 **SRP Section Number:** 13.6

Document: FINAL SER CE80 CH 14
Title: VERIFICATION PROGRAMS

Summary:

The startup test program includes security lighting system and security radio system tests. Security "lock-down" of the protected area and startup testing of the rest of the security system (i.e., intrusion detection system, alarm assessment system, access control system, etc.) were not addressed. In its December 17, 1991, response to RAI Q500.31, the applicant stated Security `lockdown' of the protected area and startup testing of the rest of the security system (i.e., intrusion detection system, alarm assessment system, access control system, etc.) is considered sensitive information which may be withheld from the public by the directive of 10 CFR 2.790(d). Full disclosure and description of these sensitive systems and their prerequisite testing will, however, be a part of the site security plan to be submitted by the utility. The staff finds it acceptable that detailed description and NRC review of the rest of the security system and its test and acceptance criteria be deferred to the COL applicant's security plan submittal required by 10 CFR 50.34.

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Potential Impact Number: 25460 **SRP Section Number:** 5.4.7

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," specifies the staff's position on protection against the possibility of a LOCA occurring outside the containment for those systems linked to the RCS. The staff position is that future advanced light-water reactor (ALWR) designs should reduce the possibility of a LOCA outside the containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an URS at least equal to full RCS pressure. The "extent practicable" phrase is a realization that all systems must eventually interface with atmospheric pressure, and that, for certain large tanks and heat exchangers, it would be difficult or prohibitively expensive to design such systems to the URS equal to full RCS pressure. Note that the degree of isolation or number of barriers (e.g., three isolation valves) is not sufficient justification for using low- pressure components that can be practically designed to the URS criteria. For example, piping runs should always be designed to meet the URS criteria, as should all associated flanges, connectors, and packing, including valve stem seals, pump seals, heat exchanger tubes, valve bonnets, and RCS drains and vent lines. The designer should attempt to reduce the level of pressure challenge to all systems and subsystems connected to the RCS.

Potential Impact Number: 25461 **SRP Section Number:** 5.4.7

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

To protect against an intersystem LOCA (ISLOCA), designers of future ALWR plants should reduce the possibility of a LOCA outside 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 the normal RCS operating pressure.

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Potential Impact Number: 25462 **SRP Section Number:** 14.2

Document: FINAL SER ABWR CH 14
Title: INITIAL TEST PROGRAM

Summary:

The staff reviewed thirteen areas relating to initial plant test programs, described in Chapter 14 of the SSAR, and submitted by GE as part of its design certification application. These areas of review are listed below:

Summary of the Test Program and Objectives

Organization and Staffing

Test Procedures

Conduct of Test Program

Review, Evaluation, and Approval of Test Results

Test Records

Conformance of Test Program With Regulatory Guides

Utilization of Reactor Operating and Testing Experience in the Development of Test Program

Trial Use of Plant Operating and Emergency Procedures

Initial Fuel Loading and Initial Criticality

Test Program Schedule

Individual Test Descriptions

Combined License (COL) Information - ITP

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Potential Impact Number: 25463 **SRP Section Number:** 14.2

Document: RG 1.70 **Revision:** 3
Title: STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS

Summary:

Regulatory Guide 1.70, Section 14.2 provides guidance for the content of FSARs related to the initial test program. Regulatory Guide 1.70, Section 14.2 identifies four basic areas related to the initial test program which are not explicitly included in SRP Section 14.2 Areas of Review in addition to the eight basic Areas of Review described in SRP Section 14.2. The four areas not included in SRP Section 14.2 are Organization and Staffing; Conduct of the Test Program; Review, Evaluation and Approval of Test Results; and Test Records.

Potential Impact Number: 25464 **SRP Section Number:** 14.2

Document: FINAL SER CE80 CH 14
Title: VERIFICATION PROGRAMS

Summary:

The staff cited RG 1.139 in conjunction with its review of initial test program compliance with Regulatory Guides and initial test abstracts. The staff requested that the applicant revise the CESSAR-DC to address guidance applicable to the initial test program and guidance for residual heat removal system features whose testing should be described in test abstracts.

Potential Impact Number: 25465 **SRP Section Number:** 14.2

Document: FINAL SER ABWR CH 14
Title: INITIAL TEST PROGRAM

Summary:

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The staff cited RG 1.139 in conjunction with its review of initial test program compliance with Regulatory Guides. The staff requested that the applicant revise the SSAR to address guidance applicable to the initial test program and guidance for residual heat removal system features whose testing should be described in test abstracts.

Potential Impact Number: 25466 **SRP Section Number:** 14.2

Document: FINAL SER ABWR CH 14
Title: INITIAL TEST PROGRAM

Summary:

The staff's review of preoperational and startup tests for design certification were limited to those systems and components within the scope of, or directly related to, the ABWR standard plant. Other testing, with respect to utility specific aspects of the plant will be necessary to satisfy certain ABWR requirements. Testing of such systems and components should be adequate to demonstrate conformance to such requirements as defined throughout the specific chapters of the SSAR. Systems that may require such testing are (1) electrical switchyard and equipment, (2) the site security plan, (3) personnel monitors and radiation survey instruments, and (4) the automatic dispatcher control system (if applicable).

The staff concluded that GE provided a sufficient level of detail to adequately describe system-specific test prerequisites and acceptance criteria for design certification. The staff also reviewed cross-references to acceptance criteria information in other parts of the SSAR and found them acceptable. An additional review of COL action items and interfacing systems will be performed at the time a COL applicant referencing the ABWR design submits an application for a license under 10 CFR Part 52.

Potential Impact Number: 25470 **SRP Section Number:** 6.8

Document: NRC GENERIC LETTER 83-02
Title: NUREG-0737 Technical Specifications

Summary:

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A number of NUREG-0737 items which require Technical Specifications were scheduled for implementation by December 31, 1981. Each of those items is present in Enclosure 1. Included in the enclosure is guidance on the scope of a specification which the staff would find acceptable. Enclosure 2 are samples in Standard Technical Specification format with blanks or parentheses appearing where the information is plant specific. It includes appropriate pages as background information for facilities that do not have Standard Technical Specifications. These samples are for your information only.

(excerpt from Enclosure 2)

6.9 REPORTING REQUIREMENTS

ANNUAL REPORTS

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March l of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

c. Documentation of all challenges to (safety valves or) safety/relief valves.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP 6.9.1.8 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Regional Administrator of the Regional Office, or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

j. Failure or malfunction of the (safety valves or) safety/relief valves.

Potential Impact Number: 25471 **SRP Section Number:** 5.4.7

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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Summary:

Because an event that occurs during shutdown and at low power has safety significance, the staff has determined that concerns with the shutdown operation should be satisfactorily addressed before the NRC issues a final design approval for the System 80+ design. ABB-CE was required to demonstrate adequate treatment of shutdown risk by submitting information, including (1) documentation showing adequate assessment of shutdown and low-power risk, with identification of design-specific vulnerabilities and weaknesses and (2) documentation showing consideration and incorporation of design features that minimize shutdown and low-power risk vulnerabilities. The staff has reviewed the shutdown risk report submitted by ABB-CE and found it acceptable. The staff's evaluation is included Section 19.3 of this report. On this basis, DSER Open Item 5.4.3.5-1 is resolved.

In addition, the staff asked ABB-CE to describe any deviations and a justification of the deviations from GL 88-17 recommendations regarding instrumentation, operating procedures, operator training programs, and TS implementations, addressing the concerns for reduced inventory operation. The

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Potential Impact Number: 25472 **SRP Section Number:** 5.4.7

Document: FINAL SER ABWR CH 19
Title: SEVERE ACCIDENTS

Summary:

The staff found that the ABWR design includes enhanced features that reduce risk during shutdown operations when compared to operating BWRs. These features specifically address the more risk-significant operations during shutdown identified in NUREG-1449 including three independent residual heat removal (RHR) divisions.

Potential Impact Number: 25474 **SRP Section Number:** 10.3.6

Document: NUREG 1344
Title: EROSION/CORROSION-INDUCED PIPE WALL THINNING IN U.S. NUCLEAR POWER PLANTS

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Summary:

NUREG-1344 provides background information regarding erosion/corrosion. Erosion/corrosion in single-phase piping systems was not clearly recognized as a potential safety issue before the pipe rupture incident at the Surry Power Station in December 1986. This incident reminded the nuclear industry and the regulators that neither the U.S. Nuclear Regulatory Commission (NRC) nor the American Society of Mechanical Engineers (ASME) Section XI Boiler and Pressure Vessel Code require utilities to monitor erosion/corrosion in the secondary systems of nuclear power plants. This report provides a brief review of the erosion/corrosion phenomenon and its major occurrences in nuclear power plants. In addition, efforts by the NRC, the industry, and the ASME Section XI Committee to address this issue are described. Finally, results of the survey and plant audits conducted by the NRC to assess the extent of erosion/corrosion-induced piping degradation and the status of program implementation regarding erosion/corrosion monitoring are discussed. This report will support a staff recommendation for an additional regulatory requirement concerning erosion/corrosion monitoring.

Potential Impact Number: 25475 **SRP Section Number:** 10.3.6

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

General corrosion of all materials, except unclad carbon and low-alloy steels will be negligible. For unclad carbon and low-alloy steel, GE has provided conservative corrosion allowances for all exposed surfaces in accordance with ASME Code, Section III. This compatibility with the reactor coolant and compliance with the ASME Code satisfy the requirements of GDC 4 as related to the compatibility of components environmental conditions.

Potential Impact Number: 25476 **SRP Section Number:** 5.3.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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Summary:

In Chapter 1, paragraph 5.3.1.1.5 of the URD, EPRI states that although the stated controls on grinding on stainless steel may be desirable for use on PWR components, they are not required for avoiding IGSCC. ABB-CE intends to avoid fabrication processes which would severely cold work the surface of austenitic stainless steel components. Grinding is done with resin-bonded or rubber bonded, aluminum oxide or silicon carbide wheels which were not previously used on materials other than austenitic alloys. This will offer reasonable assurance of stainless steel component integrity, and that stainless steel components in contact with reactor coolant are not likely to crack. On this basis, DSER Open Item 5.2.3-11 is resolved.

Potential Impact Number: 25477 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

ABB-CE used American Nuclear Society (ANS) Safety Class 1, 2, 3, and non-nuclear safety (NNS) as defined in American National Standards Institute (ANSI)/ANS 51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," for the classification of system components as an alternative method of meeting RG 1.26. CESSAR-DC Section 3.2.2 and Table 3.2-3 correlate (1) CESSAR-DC Safety Class 1, 2, 3, and NNS of ANSI/ANS 51.1, (2) Quality Group A, B, C, and D of RG 1.26, and (3) ASME Code Section III Classes 1, 2, and 3. The FSER indicates that the relationship between the three methods of classification in the CESSAR-DC is only applicable to pressure-retaining components.

The FSER indicates that the applicant's correlation between the above classification methods is acceptable for defining the relationship between the three methods of classification for pressure-retaining components. However, the staff has not completely endorsed ANSI/ANS 51.1-1983 and cannot rely on the safety classifications in this document in determining the acceptability of non-pressure-retaining SSCs. Therefore, for non-pressure-retaining components, the staff's review of CESSAR-DC Table 3.2-1 concentrates on an evaluation of quality assurance (QA) in accordance with 10 CFR Part 50 (Appendix B) and seismic classifications.

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The staff also relies on its review of system piping and instrumentation diagrams (P&IDs) to determine the appropriate classification boundaries for pressure-retaining components.

Potential Impact Number: 25479 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

In response to RAI Q210.7 and DSER Open Item 3.2.2-3, which questioned ABB-CE's classification of the new- and spent-fuel racks as non-nuclear-safety and with a QA requirement of N (not subject to 10 CFR Part 50, Appendix B requirements), ABB-CE reclassified the new and spent fuel racks as Safety Class 3, seismic Category I, and QA Class 1. The revised classification meets the applicable QA requirements of Appendix B to 10 CFR Part 50; is commensurate with the importance of the safety function of the new and spent fuel racks; and is, therefore, acceptable. On this basis, DSER Open Item 3.2.2-3 is resolved.

Potential Impact Number: 25480 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

In response to RAI Q210.9, in which the staff questioned ABB-CE's commitment to specific portions of RG 1.151, "Instrument Sensing Lines," Revision 0, ABB-CE revised CESSAR-DC Sections 3.2.1 and 3.2.2 by stating that the COL applicant will construct instrument-sensing lines and their supports per RG 1.151, in particular, positions relating to applicable seismic and ASME Code criteria. Also, ABB-CE revised CESSAR-DC Table 1.8-1 to reference CESSAR-DC Sections 3.2.1 and 3.2.2. The staff concludes that RG 1.151 Positions C.2 and C.3, which provide guidance on the above issues, are satisfied. On this basis, DSER Confirmatory Item 3.2.2-1 is resolved.

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Potential Impact Number: 25482 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

The nuclear annex (NA) floor drainage systems are separated by division and Safety Class 3 valves will prevent backflow of water to areas containing safety-related equipment. Each subsphere quadrant is provided with redundant sump pumps (Safety Class 3) and associated instrumentation. Safety Class 3 floor drain sump pumps and associated instrumentation in the diesel generator building will prevent flooding of the diesel generators. The staff accepted the design for flood protection as described in Section 3.4.1 of the FSER.

Potential Impact Number: 25483 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the spent fuel storage capability in accordance with SRP Section 9.1.2. In Sections 9.1.2 and 9.1.3 of the FSER, the staff discusses the relationship between the design of the fuel building ventilation system (FBVS) and the design of the spent fuel pool cooling and purification system (PCPS). The staff accepted the design of the purification portion of the PCPS as a non-safety-related system based in part upon the design/classification of the FBVS.

The staff also reviewed the FBVS in accordance with SRP Section 9.4.2 as described in Section 9.4.2 of the FSER. The FBVS is located completely within a seismic Category I structure, and all safety- related components (exhaust system and associated ductwork and filter train and fans) are seismic Category I, Safety Class 3, and the quality assurance requirements of 10 CFR Part 50 (Appendix B) are applicable. The system is designed to accommodate a postulated fuel-handling accident. The equipment design

and testing conform to the requirements of ANSI/ASME Standards AG-1, N509, and N510.

ABB-CE stated that the fuel building (handling area) is part of the nuclear annex structure which is classified as seismic Category I (Safety Class 3).

The staff accepted the design of the spent-fuel storage facilities and their ventilation system as described in the FSER.

Potential Impact Number: 25484 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

With the exception of the containment penetrations and isolation devices, the compressed air systems (instrument air, breathing air, and station air) are all classified as non-safety-related, non-seismic Quality Group D. The containment penetrations and isolation devices are classified as Safety Class 2, seismic Category I, Quality Group B. The staff reviewed the compressed air systems in accordance with SRP Section 9.3.1 and accepted the design of the system as described in the FSER.

Potential Impact Number: 25485 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed sampling and postaccident sampling systems in accordance with SRP Section 9.3.2 and determined that the proposed systems comply with the quality standards requirements of GDC 1 and the seismic requirements of GDC 2 with respect to design of the sampling lines and components of the process and postaccident sampling systems to conform to the classification of the system to which each sample line and component is connected, in accordance with Positions C.1, C.2, and C.3 of RG 1.26, Positions C.1, C.2, C.3, and C.4 of RG 1.29, and the guidelines of RG 1.97.

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Potential Impact Number: 25486 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the equipment and floor drainage system (EFDS) according to the guidance of SRP Section 9.3.3. The EFDS includes equipment and floor drains, vertical drain risers, sloped horizontal drain pipes, sumps, sump pumps, piping, valves, controls, and instrumentation needed to carry radioactive and nonradioactive liquids to the appropriate collection or disposal point. Most of the systems are not safety related, that is Quality Group D and non-seismic. Piping whose failure could flood areas where safety-related equipment is located will be designed to seismic Category I requirements. The containment penetrations and the isolation valves for the containment building drain system are Safety Class 2, Quality Group B, and seismic Category I. Therefore, the design presented in the CESSAR-DC complies with Regulatory Positions C.1, and C.2 of RG 1.29 as related to the system being capable of withstanding the effects of earthquakes.

Portions of the reactor building subsphere floor drainage system protect some ESF equipment from floods. Backflow check valves in the EFDS also prevent backflow into the safety-related equipment area. ABB-CE notes that the sump pumps in the reactor building subsphere floor drainage system are safety grade, and in CESSAR-DC Table 3.2-1 their classification is given as seismic Category I Safety Class 3. ABB-CE committed to incorporate this information in the text and figures of CESSAR-DC Section 9.3.3. Also, the check valves which provide backflow protection for safety-related equipment should be identified as being seismic Category I, Safety Class 3. ABB-CE's commitment to incorporate this information in CESSAR-DC Section 9.3.3 was designated as DSER Confirmatory Item 9.3.3-1. ABB-CE revised Section 9.3.3.2.2 and Figure 9.3.3-2 to include the safety grade of the sump pumps and the backflow check valves. Upon further evaluation, the staff determined that the intent of the confirmatory item is met since the sump pumps are classified in CESSAR-DC Table 3.2-1. Therefore, Confirmatory Item 9.3.3-1 is resolved. Additionally, figures in CESSAR-DC Section 9.3.3 did not indicate where system component safety classifications change. ABB-CE stated that the P&IDs in CESSAR-DC Section 9.3.3 would be revised as necessary to indicate component safety classification and the check valve designation would be incorporated in a future revision to the CESSAR-DC. This was designated as DSER Confirmatory Item 9.3.3-2. ABB-CE revised Figures 9.3.3-1 through 9.3.3-5 to

indicate the safety classification of the piping and components, including the backflow check valves. The staff finds the revised figures acceptable. Therefore, DSER Confirmatory Item 9.3.3-2 is resolved.

Potential Impact Number: 25487 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

The staff reviewed the containment isolation system (CIS) in accordance with SRP 6.2.4 and BTP CSB 6-4.

In the CESSAR-DC, ABB-CE states that the CIS design conforms to the following regulations and standards:

- ANSI N18.2 and RG 1.29, "Seismic Design Classification:" Design and construction of the CIV and connecting piping (Safety Class 2 and seismic Category I).

- ASME Code: Qualification of the CIV and piping in accordance with Section III of the ASME Code as Class 2 components

In Amendment U to the CESSAR-DC, ABB-CE took a new position regarding the containment penetrations for the four Safety Injection System (SIS) direct vessel injection (DVI) lines and the two parallel shutdown cooling system (SCS) discharge lines and SIS warmup bypass lines. They pointed out that these penetrations are part of closed loops outside of containment and have only a single outboard containment isolation valve for each line. By taking this position, the inboard check valves next to the containment are no longer considered to be CIVs. These lines are in an ESF system and will have higher system reliability in its ECCS function by minimizing the number of valves in the line. There are two barriers on each line for containment isolation; one is the outboard containment isolation valve, and the other is the closed loop piping outside containment. The closed loop SIS outside containment is protected from missiles, is designed to seismic Category I and Safety Class 2, and has a design temperature and pressure exceeding the containment design temperature and pressure. The integrity of the closed system is maintained during normal plant operation and is periodically pressure tested in accordance with the pre-service and in-service inspection programs required by 10 CFR 50.55a and the ASME Code. The

staff accepted the design of the containment isolation system as described in Section 6.2.4 of the FSER.

Potential Impact Number: 25489 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In CESSAR-DC Table 3.2.1, ABB-CE states that the vital instrumentation and equipment rooms (including battery rooms) and the components of the air conditioning system for the main control room (including the air handling units and its filters, fans, ductwork, water-cooling coils, and heating coils) are Safety Class 3 and seismic Category I; will meet the quality assurance requirements of Appendix B to 10 CFR Part 50; and will remain functional after a safe-shutdown earthquake. The provision for seismic Category 1, Class 1E chemical instrumentation to detect toxic releases at air intakes if needed, is included in CESSAR-DC Section 6.4.2.2. The staff reviewed control room habitability systems in accordance with SRP Section 6.4 and found them to be acceptable as discussed in Section 6.4 of the FSER.

Potential Impact Number: 25490 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In CESSAR-DC Section 6.8, ABB-CE presents the design and description of the in-containment water storage system. This system will collect, deliver, and store water and is the heat sink inside the containment during normal operation and accident conditions. It consists of the IRWST, the HVT, the steam relief system (SRS), and the cavity flooding system (CFS). The in-containment water storage system is Safety Class 2. Therefore, it will be manufactured, and tested in accordance with the rules of ASME Boiler and Pressure Vessel Code, Section III. The staff found the design of the system to be acceptable as discussed in Section 6.8 of the FSER.

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Section 19.2.3.3.2.1.2 of the FSER provides further detail indicating that, as stated in CESSAR-DC Table 3.9-4, "Seismic I Active Valves," and 3.9-15, "Inservice Testing Safety-Related Pumps and Valves," the CFS (a severe accident management system, not used for design basis events) are ASME Section III, Code Class 2 and Safety Class 2. The CFS valves are stroke tested each refueling outage in accordance with the Inservice Inspection and Testing Program, and Section XI of the ASME Code.

Potential Impact Number: 25491 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the spent-fuel pool cooling and purification systems (PCPS) in accordance with SRP Section 9.1.3 and accepted the design as described in the FSER. The spent-fuel pool cooling portion of the system is designed to seismic Category I, Safety Class 3 requirements. The purification portion of the system is classified as non-safety-related with the exception of the penetrations to and from the refueling pool (located inside the containment). These penetrations consist of inboard and outboard manual containment isolation valves (CIVs) on both the suction and discharge lines. The penetrations are designed to seismic Category I, Safety Class 2 requirements.

Makeup water for the PCPS is provided by the chemical and volume control system (CVCS), a non-safety-related, seismic Category I, Safety Class 3 system which provides up to 379 L/minute (100 gpm) of borated makeup water to the PCPS. Non-borated makeup water is provided by the demineralized water makeup system (DWMS) to replace evaporative water loss.

Potential Impact Number: 25492 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

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The staff reviewed the design of the Component Cooling Water System (CCWS) in accordance with SRP Section 9.2.2. Each CCWS division consists of an essential and nonessential cooling loop. The essential cooling loop piping and components (e.g., heat exchangers, pumps, and surge tanks) are seismic Category I and Safety Class 3. CIVs and containment penetration piping are seismic Category I and Safety Class 2. Nonessential components are supplied CCW by means of non-nuclear safety class cooling loops with the exception of the charging pump motor coolers and miniflow heat exchangers, the instrument air compressors, and the DG engine starting air aftercoolers which are supplied CCW by means of Safety Class 3 cooling loops. The isolation valves separating seismic Category I portions from the nonseismic portions are Class 2 or 3 (i.e., Quality Group B or C, respectively). In addition, the CCWS is designed to preclude any adverse interaction with nonseismic systems in the vicinity. The staff found the system design to be acceptable as described in the FSER.

Potential Impact Number: 25494 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the design of the chilled water systems in accordance with SRP Section 9.2.2.

The essential chilled water system (ECWS) consists of two divisions of equal size. Each division can supply 100 percent of the cooling capacity required to meet system demands during normal and accident conditions. Each 100-percent-capacity ECWS division consists of a chiller, a heat exchanger, two chilled water pumps, an expansion tank, a chemical addition tank, piping, valves, controls and instrumentation.

ABB-CE indicates that the ECWS and the normal chilled water system (NCWS) are indirectly connected through a heat exchanger. As shown in CESSAR-DC Table 3.2-1, the ECWS heat exchanger is designed to seismic Category I, Safety Class 3. The ECWS heat exchanger is designed to allow the NCWS to serve all of the ECWS during periods of normal operation without directly connecting the water pathways. Therefore, the integrity of the safety-related ECWS is not degraded by postulated failures in the NCWS.

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The NCWS is a non-safety-related system. Each division of the NCWS enters the primary containment through two penetrations: one for the supply line and the other for the return line. Isolation valves and piping for the primary containment penetrations are safety related and are designed to seismic Category I Safety Class 2 and 10 CFR Part 50 (Appendix B) standards. The non-safety-related portions of the system whose failure during a seismic event could affect any structure, system, or component important to safety, are designed in accordance with seismic Category II criteria. The staff accepted the design of the chilled water systems as described in the FSER.

Potential Impact Number: 25495 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the radwaste building ventilation system (RWBVS) in accordance with SRP Section 9.4.3.

ABB-CE states (in CESSAR-DC Table 3.2-1) the following (1) RWBVS components are non-nuclear safety class, non-seismic, and the quality assurance requirements of 10 CFR Part 50 (Appendix B) are not applicable; and (2) the radwaste facility structure is seismic Category II, and Appendix B quality assurance requirements are applicable. The staff accepted the design of the facility structure and ventilation system as described in the FSER.

Potential Impact Number: 25496 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the diesel building ventilation system (DBVS) in accordance with SRP Section 9.4.5. The DBVS serves two redundant EDGs located in separate areas inside the nuclear annex on opposite sides of the reactor building. The DBVS is designed for once-through flow using inlet and exhaust fans, filters, and dampers.

Electric heaters, activated on low temperature, maintain temperature above freezing and fans are automatically activated to control elevated temperature. Each division of the non-safety-related supply portion of the system consists of one 100-percent-supply fan equipped with a damper and prefilter. Air is exhausted to the outdoors through each division of the safety-related exhaust portion of the system which consists of two 50-percent-supply fans. The system is located in a seismic Category I building and all safety-related components are rated seismic Category I. ABB-CE revised CESSAR-DC Table 3.2-1 to identify the system space heaters and supply fan and their associated ductwork and dampers as Safety Class 3, seismic Category I; therefore, the quality assurance requirements of 10 CFR Part 50 (Appendix B) are applicable. The remaining components are designed as non-seismic safety class. The staff accepted the design of the ventilation system as described in the FSER.

Potential Impact Number: 25498 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the containment cooling and ventilation system (CC&VS) in conformance with GDC 2, as related to the system being capable of withstanding the effects of earthquakes, referenced in SRP Section 9.4.5. This system maintains suitable environmental conditions inside the containment for normal operation, maintenance, and testing. The system is non-safety related except for high-and-low purge CIVs and penetration ductwork that isolate portions of the system inside the containment from portions of the system in the nuclear annex up to the HEPA filters. The low-purge and high-purge systems are designed to maintain the containment under slight negative pressure with respect to the atmosphere. The CC&VS comprises (1) the recirculation cooling system, (2) the low- and high-purge supply and exhaust subsystems, (3) the containment air cleanup system, (4) the pressurizer compartment cooling redundant fans, (5) the reactor cavity compartment cooling redundant fans, and (6) the CEDM cooling system. High and low purge exhaust ductwork from the containment penetration to the HEPA filters, penetration ductwork and CIVs are seismic Category I, Safety Class 2 and Quality Class 1 and the quality assurance requirements of 10 CFR Part 50, (Appendix B) are applicable. The safety-related equipment, fans, dampers, coils, and ductwork are designed and tested in accordance with RG 1.52 and ASME/ANSI N509, N510, and AG-1 codes and standards. The staff accepted the design of the ventilation system as described in the FSER.

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Potential Impact Number: 25499 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the nuclear annex ventilation system (NAVS) in accordance with SRP Section 9.4.5. The safety-related cooling systems for the mechanical equipment room, serving the CCWS pump rooms and the safety-related CWS system pump and chiller rooms, are designed to seismic Category I requirements and remain functional following a design-basis earthquake (DBE). These cooling systems consist of two completely redundant, independent, full-capacity systems. ABB-CE revised CESSAR-DC Table 3.2-1 in response to the staff's RAI Q210.1 to identify (1) the nuclear annex structure as seismic Category I; (2) safety-related recirculation units and room recirculating unit cooling coils as Safety Class 3, seismic Category I; and (3) the quality assurance requirements of 10 CFR Part 50 (Appendix B) as applicable. The remaining components are not safety-related. All safety-related components are designed as seismic Category I equipment and are located in a seismic Category I structure. The safety-related equipment, fans, dampers, coils and ductwork are designed and will be tested in accordance with RG 1.52 and ASME/ANSI N509, N510, and AG-1 codes and standards. The staff accepted the design of the ventilation system as described in the FSER.

Potential Impact Number: 25500 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the equipment and floor drainage system (EFDS) including the diesel generator building sump pump system (DGBSPS) in accordance with SRP Section 9.3.3. The DGBSPS is an ANSI Class 3 piping system and the pumps and system components are designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 3. Instrumentation associated with

sump pump operation is Safety Class 3 and Seismic Category I. The staff accepted the design of the system as described in the FSER.

Potential Impact Number: 25503 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

The staff evaluated the quality group and seismic design classification applied to equipment and structures housing radioactive waste management systems (liquid, gaseous, and solid waste systems). The staff used the acceptance criteria given in SRP Sections 11.2, 11.3, and 11.4 in its review of radioactive waste management systems. The staff cited Regulatory Guide 1.143 extensively as guidance applicable to the overall review of these systems. The staff accepted the design of the systems as described in the FSER.

Potential Impact Number: 25505 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

The staff evaluated the quality group and seismic design classification applied to equipment and structures housing radioactive waste management systems (liquid, gaseous, and solid waste systems). The staff cited Regulatory Guide 1.143 extensively as guidance applicable to the overall review of these systems. The staff accepted the design of the systems as described in the FSER.

The staff considered the capabilities of the proposed liquid radwaste treatment system to meet the demands of the plant resulting from anticipated operational occurrences and has concluded that the system's capacity and design flexibility are adequate to meet the anticipated needs of the plant. The design of the systems and structures housing these systems meets the applicable criteria given in RG 1.143, Revision 1.

Specifically, the base mat and outside walls of the housing structures are seismic Category I to a height necessary to retain spilled liquids within the building.

The staff reviewed the seismic design criteria including the quality group classifications used for the gaseous waste management system components and the structures housing the radwaste system and concluded that the design of the system and the structures meets the applicable criteria specified in RG 1.143.

For the solid radioactive waste management system, SSAR Sections 11.4.1.2 and 11.4.2.1 and GE's submittals dated June 7 and 29, 1990, and June 2, 1993, state that the quality group classification, seismic design, and other design features (such as heat tracing concentrate piping and tanks, and flushing connections for all components and piping that contain slurries) meet the guidelines of RG 1.143 and BTP ETSB 11-3, Position B.V. The staff concluded that the solid radwaste management system design for the ABWR is acceptable.

Potential Impact Number: 25507 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

In its evaluation of the proposed resolution related to the requirements of 10 CFR 50.34(f)(1)(x) which addresses a TMI Action Plan issue titled "Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Study and Verify Qualification of Accumulators on ADS Valves (TMI Item II.K.3(28))," the staff discussed the quality group classification of ADS accumulators and their associated safety-related nitrogen supply system as a factor providing assurance that the ADS system will perform its required functions. The staff indicated that the ADS valve accumulators, including the associated equipment to the ADS valves, and the safety-related portions of the nitrogen gas supply system are designed to seismic Category I, quality group B or C, as appropriate, and quality assurance B requirements. Based on its evaluation, the staff concluded that (1) GE's criteria for sizing each nitrogen gas accumulator and the provision for supply of makeup nitrogen gas to the accumulator is appropriate, and (2) the nitrogen accumulators and the associated equipment, including the safety-related portions of the nitrogen gas supply system, will perform their intended functions during

and following an accident situation. Therefore, the ABWR design adequately addresses the requirements of this TMI item.

Potential Impact Number: 25511 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

The staff reviewed the main steam supply system (MSSS) in accordance with SRP Section 10.3. From the main steam line (MSL) header, in addition to the four steamlines to the turbine, two steamlines will supply steam to the power cycle auxiliary equipment. One of the branch steamlines will supply steam to the turbine gland sealing system (TGSS) and to two reheaters. The other branch line will supply steam to the offgas system, the steam jet air ejectors (SJAE), the condenser sparger, and two other reheaters. Each of these steamlines contains a power-operated pneumatic gate shutoff valve. These valves are Quality Group (QG) B, seismic Category I and are equipped with air operators and spring closure mechanisms. These valves are analyzed to demonstrate structural integrity under SSE loading conditions. SSAR Section 5.4.9 and Table 3.2-1 state that the MSLs from the reactor vessel, out to and including the outboard MSIVs, are designed to QG A standards. From the outboard MSIVs to the turbine stop valves, the steamlines and associated equipment are designed to QG B standards. The MSLs that will extend from the reactor vessel up to and including the seismic interface restraint (which is downstream of the outboard MSIVs) are seismic Category I. Downstream of the seismic interface restraints, the main steam (MS) piping and equipment is classified as nonnuclear safety-related. This includes the shutoff valves on the two branch steamlines (inside the TB) that will supply steam to the power cycle auxiliary equipment (QG D).

SSAR Section 3.2.5.3, however, states that the MSLs from the containment outboard isolation valves, up to and including the turbine stop valves and all branch lines 6.3 cm (2.5 in) and larger and up to and including the first valve and its supports, are designed using an appropriate dynamic seismic system analysis to withstand the safe shutdown earthquake (SSE) design loads in combination with appropriate loads within the limits specified. In the DSER (SECY-91-235), the staff stated that the design for the MSLs downstream of the outboard MSIVs up to the turbine stop valves, the connecting lines up to the turbine bypass valves, and all other connecting lines up to and including the first shutoff valves, did not comply with the staff's requirement for seismic and QG

classifications as stated in SRP Section 10.3, Criterion III.3.b, which requires that the subject portions of the MSSS be designed to seismic Category I and QG B requirements. Therefore, the staff identified the lack of seismic Category I classification of the subject portions as DSER Open Item 3 (SECY-91-153). Although the subject piping portions will not be seismic Category I, by letter dated April 1, 1992, GE committed to apply the quality assurance criteria of 10 CFR Part 50, Appendix B, to these portions. The combination of appropriate dynamic seismic system analysis, QG B classification, and application of the Appendix B criteria to these portions are adequate and acceptable. Therefore, this open item was resolved in the DFSER (this issue is further discussed in Section 3.2.1 of the FSER).

The MSSS includes all components and piping from the outermost containment isolation valve up to and including the turbine stop valves. The safety-related portions of the system are designed to QG B from the outermost containment isolation valve and connecting piping up to and including the first normally closed valve. Those portions of the MSSS necessary to mitigate the consequences of an accident are designed to the quality standards commensurate with the importance to its safety function. The scope of the review included layout drawings, P&IDs, and descriptive information for the system. The staff concluded that the MSSS for the ABWR from the reactor to the TB is acceptable as described in the FSER.

Potential Impact Number: 25515 **SRP Section Number:** 5.4.7

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The design of interlocks will not automatically close the isolation valves in the event of an RCS pressurization during shutdown cooling. This design of interlocks prevents a loss of decay heat removal capability due to inadvertent closure of the isolation valves during SCS operation.

Potential Impact Number: 25516 **SRP Section Number:** 5.4.7

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

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Summary:

Issue 99, in NUREG-0933, addressed the staff's concerns about the inadvertent closing of the SCS suction valves when the SCS is in use. In existing plants, the auto-closure interlocks (ACIs) on suction isolation valves of the SCS were installed to guard against an operator error, namely, failure to isolate the SCS from the RCS before raising the RCS pressure above the design pressure of the SCS; however, ACIs have been a frequent cause of loss-of-SCS events. As the result of a regulatory analysis in NUREG-0933, the staff recommended removing ACIs to improve the reliability of the SCS. Over the past several years, efforts have increased to improve the reliability of the SCS. In 1988, the staff broadened the scope of this issue to include risk associated with midloop operation. In GL 88-17, "Loss of Decay Heat Removal," dated October 17, 1988, the staff asked licensees to address numerous safety concerns and to improve the operational safety for such plant conditions as reduced RCS water inventory.

ABB-CE discussed this issue in CESSAR-DC Section 20.2.27. To address the concerns of this issue regarding the reliability of the SCS, the System 80+ design contains the following instrumentation consistent with the guidelines in GL 88-17:

- two independent level indications for continuous management of RCS level
- two independent channels of core exit thermocouples (CETs) for continuous measurement of core exit temperature
- indication and alarms in the control room for each of the level and temperature measurement instruments given in the preceding two items with each instrument of an independent pair having its own separate power supply
- additional instrumentation, indication, and alarms for plant parameters (such as SCS pump suction pressure and motor current, vortex monitoring equipment, and SCS flow) enabling operators in the control room to continuously monitor the SCS when it is in use for RCS cooling
- no ACI on the SCS suction isolation valves

The SCS design is consistent with the guidelines for removal of ACIs. Additional improvements are discussed in Section 5.4.3.2 of this report. Also, the guidelines in GL 88-17 cover broader areas, such as instrumentation, equipment, procedures, analyses, TS, and RCS perturbation for operation at low RCS water inventory.

Potential Impact Number: 25517 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 9

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Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the spent fuel pool cooling and cleanup (FPC) system and its makeup system in accordance with SRP Section 9.1.3. The FPC system is a non-safety-related system designed to remove decay heat generated by spent fuel assemblies in the spent fuel storage pool; to maintain water quality and clarity; and to remove corrosion products, fission products, and other impurities from the pool water. Normal makeup water is supplied to the storage pool by the non-safety-related makeup water (condensate) system (MUWC). Backup to the normal makeup will also be available from the nonsafety-related suppression pool cleanup (SPCU) system. Additionally, an emergency safety-related, seismic Category I makeup water source to the spent fuel pool will be provided by FPC connections to the residual heat removal (RHR) system, which draws water from a safety-related water source, that is, the suppression pool. The system includes redundant safety-related isolation devices to facilitate isolation of the safety-related makeup portion of the system from the nonsafety-related SPCU system. The SPCU system makeup uses the same FPC system safety-related piping segment that the RHR makeup uses. FPC system piping from the spent fuel pool to the RHR system is classified as safety related. GE stated that the FPC system, except for the filter demineralizers, is designed to seismic Category I, Quality Group (QG) C standards. Although the entire system is not classified as seismic Category I, QG C standards, the nonsafety-related portions of the system that could affect any SSCs important to safety if they failed during a seismic event, are designed to ensure their integrity under seismic loading.

The ABWR FPC system meets the alternate design guidance provided in SRP Sections 9.1.3.II.1.a and b. The staff's review of the FPC system included all components and piping of the system from the inlet to and the exit from the storage pool, the seismic Category I makeup source and piping used for the fuel pool makeup, the cleanup system filter/demineralizers, and the regenerative process to the point of discharge to the radwaste system. The scope of the FPC system review included layout drawings, process flow diagrams, piping and instrumentation diagrams, and descriptive information for the system and supporting systems that are essential to safe operation. The cooling portion of the system and the emergency primary makeup system are designed to seismic Category I, QG C requirements, since they are necessary to remove decay heat from the spent fuel and to prevent fuel damage that could lead to unacceptable releases of radioactivity. The staff concluded that the design of the FPC system and its makeup systems meet the guidelines of SRP 9.1.3 and are acceptable as described in the FSER.

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Potential Impact Number: 25518 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the makeup water system (condensate) MUWC system (also known as the condensate storage and transfer system) in accordance with SRP Section 9.2.6. The MUWC system is not safety-related, except as noted below, because it will not affect the integrity of the reactor coolant system pressure boundary, prevent achieving and maintaining safe shutdown, or affect the capability to prevent or mitigate the consequences of accidents, which could result in unacceptable offsite radiological exposures.

As stated in SSAR Table 3.2-1, for the MUWC, reactor core isolation cooling (RCIC), and high pressure core flooding (HPCF) systems, certain parts of MUWC system piping, including supports and valves, will be designed to seismic Category I and Quality Group B standards and will be located in seismic Category I, flood-protected and tornado-missile-protected structures. The safety-related portions include those forming part of the containment boundary and those system piping portions that interface with the safety-related RCIC and HPCF systems, up to and including the isolation/suction valves for the systems from the MUWC. The non-safety-related portions of the system that could affect any SSCs important to safety if they fail during a seismic event are designed to ensure their integrity under seismic loading conditions resulting from an SSE. The level instrumentation in the MUWC system that facilitate the automatic switchover of the HPCF and RCIC pumps suction from the CST to the suppression pool and their power supplies are safety-related.

The MUWC system includes all components and piping associated with the system to the points of connection with other systems. The staff determined that the SSAR design criteria and bases for the MUWC and the requirements for a sufficient water supply to safety-related systems during normal and emergency conditions are acceptable. The staff thus concluded that the MUWC system design is acceptable as described in the FSER.

Potential Impact Number: 25519 **SRP Section Number:** 3.2.2

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Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the makeup water system (purified) distribution system (MUWP) in accordance with SRP Section 9.2.3. The MUWP system will supply demineralized makeup quality water and will include a piping distribution system from the source to the components that require this water. The MUWP system will supply makeup water to systems such as the makeup water (condensate) (MUWC) system, the reactor building cooling water (RCW) system, the turbine building cooling water (TCW) system, the diesel generator cooling water (DGCW) system, the liquid waste management system (LWMS), the standby liquid control system (SLCS), and other plant auxiliary systems. The MUWP system is not safety-related, except as noted below, because it will not affect the capability of the reactor coolant system pressure boundary, the capability to achieve and maintain safe shutdown, or the capability to prevent or mitigate the consequences of accidents that could result in unacceptable offsite radiological exposures. The MUWP system enters the primary containment through one penetration. The system piping through this penetration has a locked-closed manual valve outside the containment and a check valve inside the containment. The portions of the system penetrating the containment (including these two valves) are designed to seismic Category I, Quality Group B requirements. The staff concluded that the MUWP system is acceptable as described in the FSER.

Potential Impact Number: 25520 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the reactor building cooling water (RCW) system in accordance with SRP Section 9.2.2. The function of the RCW system is to remove heat from plant auxiliaries (some of which are required for safe shutdown) during normal operation and after a LOCA. The RCW system consists of safety-related and non-safety-related portions. Portions of the system that are necessary for safe shutdown, accident prevention, or accident mitigation including piping (including valves forming part of the primary containment boundary and other safety-related portions of the system piping up to and including the isolation valves that isolate the system from its non-safety-related

portions) are designed to the requirements of seismic Category I, Quality Groups B or C, and 10 CFR Part 50, Appendix B. The safety-related portions include the RCW system pumps, heat exchangers, surge tanks, and the division isolation valves. Instrumentation and controls performing safety-related functions (e.g., surge tank level switches) are located in the safety-related portions of the system. Electric modules (e.g., sensors, power supplies, signal processors) and cables performing safety-related functions are all designed to seismic Category I and Quality Group B requirements. Non-safety-related portions of the system that can adversely impact safety-related SSCs if they fail during a seismic event, are designed to ensure their integrity under seismic loads resulting from an SSE. The safety-related portions are located in seismic Category I structures. The staff reviewed the SSAR design criteria, design bases, and safety classification for the RCW system against the requirements for supplying adequate cooling water for the safety-related ECCS components and reactor auxiliary equipment for all conditions of plant operation. The staff concluded that the system is acceptable as described in the FSER.

Potential Impact Number: 25521 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the HVAC normal cooling water (HNCW) system in accordance with SRP Section 9.2.2. The HNCW system is not safety related except for portions of the system that penetrate the primary containment, the portions of the system that are part of the secondary containment boundary, and the associated isolation valves. The function of the HNCW system is to provide chilled water to the drywell cooler cooling coils and cooling coils of other non-safety-related air conditioners, primarily in the reactor, control, radwaste, and service buildings. The HNCW system is not safety-related because it is not required to ensure (1) integrity of the RCS pressure boundary, (2) capability to achieve and maintain safe shutdown, or (3) the ability to prevent or mitigate offsite radiological exposures during accidents. The HNCW system joins the primary containment through two penetrations: one for the supply line and the other for the return line. The supply line penetration has one motor-operated isolation valve outside the containment and a check (isolation) valve inside the containment. The return line penetration has two motor-operated isolation valves, one inside and one outside the containment. Isolation valves and piping for the primary containment penetrations are safety-related and are designed to seismic Category I, Quality Group

B, and 10 CFR Part 50, Appendix B, standards. Piping for penetrations for secondary containment is designed to seismic Category I and 10 CFR Part 50, Appendix B, standards. The rest of the HNCW system is not safety-related, as stated above, and is designed to non seismic Category I standards. However, the non-safety-related portions of the system whose failure during a seismic event could affect any structure, system, or component important to safety, are designed to ensure their integrity under seismic loads resulting from an SSE. Appropriate American Society of Heating, Refrigeration, and Air Conditioning (ASHRAE), American Society of Mechanical Engineers (ASME), Tank Equipment Manufacturers Association (TEMA), and Hydraulic Institute (HI) standards are used for all tests. The staff reviewed the SSAR design criteria, design bases, and safety classification for the HNCW system against the requirements for supplying adequate cooling water for auxiliary equipment for all conditions of plant operation. The staff concluded that the system is acceptable as described in the FSER.

Potential Impact Number: 25522 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the HVAC emergency cooling water (HECW) system in accordance with SRP Section 9.2.2. The HECW is a closed cooling water system whose function is to provide cooling water to the main control room (MCR) air conditioners, reactor building essential electrical equipment room (diesel generator zone) coolers, and control building essential electrical equipment room coolers. The HECW system is required to operate at normal power, reactor shutdown, and after any postulated abnormal reactor conditions, including a LOCA. The HECW system has no primary or secondary containment penetrations. GE states that the entire HECW system is safety-related. GE indicated that the only non-safety-related portions of the HECW divisions are the chemical addition tank (only one such tank is provided to serve the entire system) and the piping from the tank to the safety-related valves which isolate the tank from the safety-related portions of the system. The HECW system and the cooling water lines from the RCW system are designed to the requirements of seismic Category I, 10 CFR Part 50, Appendix B, and Quality Group C. Portions of the HECW system that are necessary for safe shutdown, accident prevention, or accident mitigation are designed to seismic Category I and Quality Group B requirements. The staff reviewed the SSAR design criteria, design bases, and safety classification for the

HECW system against the requirements for supplying adequate cooling water for the safety-related ECCS components and reactor auxiliary equipment for all conditions of plant operation. The staff concluded that the system is acceptable as described in the FSER.

Potential Impact Number: 25527 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff stated that to meet the requirements of GDC 1 and 2, the seismic design and quality group classification of sampling lines for the process sampling system (PSS) should conform to the classification of the system to which each sampling line and component is connected. GE indicated that the seismic design and group classification of sampling lines and their components will conform to the classification of the system into which they are connected. The staff finds this response acceptable.

Potential Impact Number: 25529 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The staff reviewed the radioactive drain transfer system (DTS) in accordance with SRP Section 9.3.3. The DTS is designated as non-safety-related (except for containment penetrations and piping in the drywell) and designed to collect radioactive or potentially radioactive effluents in equipment or floor sumps and then transfer the effluents to the liquid waste management system (LWMS) for processing. The drains, piping, pumps, instrumentation, and valves of the DTS system are classified as non-safety-related, except for the containment (drywell) penetrations and containment isolation valves, which are safety Class 2, and designed in accordance with seismic Category I and Quality Group (QG) B criteria, and the reactor building penetrations that meet ASME Code III, Section 3 (sic.), requirements.

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The staff indicated in DSER (SECY-91-355) Open Item 77 that the check valves that provide backflow protection for sumps in the ECCS equipment rooms should be classified as safety Class 3 and designed to seismic Category I and QG C criteria. GE revised SSAR Table 3.2-1 and Figure 11-2 to include the designation of these valves as being non-safety designed to seismic Category I and QG C criteria. The staff finds that these modifications ensure that the check valves will provide adequate protection from backflow into areas containing safety-related equipment. DFSER Confirmatory Item 9.3.8-1 is resolved. In addition, the staff reviewed Amendment 27 of the SSAR and concludes that GE clarified the description of the containment (drywell) sump isolation valves. The valves meet seismic Category I, safety Class 2, and QG B, requirements.

The staff concluded that the DTS system is acceptable as described in the FSER.

Potential Impact Number: 25531 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

The staff reviewed the containment isolation system (CIS). The containment isolation system includes containment isolation valves (CIVs) and associated piping and penetrations necessary to allow normal or emergency passage of fluids through the primary containment while preserving the capability to prevent or limit the escape of fission products from the containment boundary in the event of a LOCA. Penetrations, piping, isolation valves, and rupture discs will maintain their structural integrity for all accident conditions. The design meets seismic Category I and safety Class 2 requirements.

The staff concluded that the system is acceptable as described in the FSER.

Potential Impact Number: 25533 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

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The staff reviewed the reactor water cleanup system (CUW) description and piping and instrumentation diagrams in accordance with SRP Section 5.4.8. The majority of the system (including all safety-related components) is located in the nondivisional quadrant of the secondary containment portion of the reactor building. The system is classified as non-safety related with the exception of the containment isolation valves. System piping and components in the drywell, up to and including the outboard containment isolation valves, form part of the RCPB are classified as seismic Category I, Quality Group (QG) A. The remainder of the system is classified as nonseismic, QG C. Low-pressure piping in the backwash and precoat area downstream of the block valves is classified as QG D. Based on this QG and seismic classification design information, the staff concludes that the CUW system meets the guidelines of RGs 1.26 and 1.29 and, therefore, the requirements of GDC 1 and GDC 2 as they relate to the CUW design meeting the standards commensurate with the system's safety function and the system being able to withstand the effects of natural phenomena. The Advisory Committee on Reactor Safeguards (ACRS) expressed concerns regarding the system quality group classifications and potential for reverse feedwater flow into secondary containment through a CUW system pipe break. GE modified the system design to ensure that reverse flow from the feedwater system to the secondary containment and steam tunnel would not occur as a result of a CUW line break in these areas. The modified arrangement also allows use of two motor-operated valves in the nuclear boiler system (NBS) which are available to provide additional long-term manual isolation. These NBS valves are classified as safety-Class 2, Quality Group B, and seismic Category I. The staff concluded that GE has met the requirements of GDC 1 by designing, in accordance with the guidelines of RG 1.26, the portion of the CUW extending from the reactor vessel to the outermost primary containment isolation valves to Quality Group A and by designing, in accordance with position C.2.C of RG 1.26, the remainder of the CUW outside primary containment (excluding the precoat unit) to Quality Group C. The staff concluded that the system is acceptable as described in the FSER.

Potential Impact Number: 25534 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

GE uses American Nuclear Society (ANS) SC 1, 2, 3, and non-nuclear safety (NNS) as defined in American National Standards Institute (ANSI)/ANS 52.1-1983 for the

classification of system components as an alternative method of meeting RG 1.26. SSAR Tables 3.2-2 and 3.2-3 provide a correlation between (1) ABWR SC 1, 2, 3 and NNS, (2) NRC QG A, B, C, and D in RG 1.26, and (3) ASME Code Section III classes. All pressure-retaining components classified as QG A, B, or C are constructed in accordance with ASME Code, Section III, Class 1, 2, or 3 rules, respectively. Construction as defined in ASME Code, Section III, Subsections NB/NC/ND-1110(a) and used herein is an all-inclusive term comprising materials, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of components. Components classified as QG D are designed to the applicable standards identified in SSAR Table 3.2-3.

Because the staff has not endorsed ANSI/ANS 52.1-1983, it cannot rely on those safety classifications for determining the acceptability of non-pressure-retaining SSCs. Therefore, in performing the review of SSAR Table 3.2-1 for non-pressure-retaining components, the staff concentrated on an evaluation of QA in accordance with Appendix B to 10 CFR Part 50 and seismic classifications.

SSAR Table 3.2-1 identifies all ABWR safety-related mechanical and electrical equipment as (1) safety Class 1, 2, or 3, (2) seismic Category I, and (3) designed to the quality assurance requirements of 10 CFR Part 50, Appendix B. As discussed in Sections 3.2.1 and 3.2.2 of the FSER, the staff concludes that Table 3.2-1 is acceptable. On the basis of these evaluations, the staff concludes that GE meets GDC 1, 30, and 10 CFR Part 50, Appendix B, as they relate to qualifying safety-related mechanical and electrical equipment to appropriate quality standards commensurate with the importance of the safety function to be performed.

Potential Impact Number: 25535 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

In Section 3.2.2 of the DSER (SECY-91-153), the staff stated its concern about the QA requirements of the new and spent fuel storage racks and the storage container for defective fuel. This was Outstanding Issue 2 in SECY-91-153. The staff's position is that the new and spent fuel storage racks are important to safety and, as a minimum, should meet the applicable QA requirements of Appendix B to 10 CFR Part 50, in addition to being classified as seismic Category I. This position is consistent with RG

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1.29. The storage containers for defective fuels are designed to the same QA requirements as the fuel storage racks. However, because these storage containers are stored in the seismic Category I fuel storage racks, they do not have to be seismic Category I. SSAR Table 3.2-1 defines QA Category E, which is applicable to the new and spent fuel storage racks and the storage container for defective fuel. Category E states that elements of Appendix B to 10 CFR Part 50 are generally applied to these components, commensurate with the importance of the component's function. This commitment satisfies the applicable guidelines in RG 1.29 and is acceptable. Therefore, DSER Outstanding Issue 3 is resolved.

Potential Impact Number: 25539 **SRP Section Number:** 5.4.7

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

In accordance with NRC Bulletin 88-08, the staff is requesting that licensees and applicants review systems connected to the reactor coolant system (RCS) to determine if any sections of such piping that cannot be isolated can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves.

In response to this request, GE stated that the ABWR systems connected directly to the RCS or the RPV are the nuclear boiler system and the emergency core cooling systems. In the nuclear boiler system, GE reviewed the feedwater subsystem that supplies makeup water to the RPV to confirm that design requirements for temperature stratifications of feedwater piping were satisfactorily defined in system specifications and piping cycle diagrams. In the design of the ECCS, both the residual heat removal (RHR) system and high-pressure core flooder (HPCF) have piping that are directly connected to the RPV. In the unisolable sections of RHR piping, leaking toward the RPV cannot occur because the pressure will always be higher on the reactor side during normal plant operation when the upstream pumps are not operating.

In the DFSER, the staff concluded that GE had adequately addressed the potential problems described in Bulletin 88-08. Subsequently, however, the staff noted that GE did not address the potential problem described in Supplement 3 to Bulletin 88-08, although Supplements 1 and 2 were addressed adequately. It involved the development of potential cyclic stratified flow and associated thermal striping that may

occur because of possible leakage past the valve disk and out the valve stem packing gland. This flow stratification and striping may occur when the pressure on the upstream side of the valve is less than the RPV system pressure during normal operation. Supplement 3 reported an incident in which this phenomenon induced thermal fatigue, which led to a through-wall pipe crack in a foreign reactor. The staff reviewed the SSAR and requested that GE provide additional information to address this concern. GE then identified the ABWR piping sections that could be susceptible to unacceptable thermal stresses, owing to this phenomenon. They include sections of the RHR system and the HPCF system. For the affected piping sections, GE will require that either (1) the gate valve in each of the unisolable piping sections be located at a distance equal or greater than 25 pipe diameters from the RPV nozzle or (2) stress analysis be performed to show that stresses and fatigue from potential stratification and thermal striping are acceptable per the ASME Code. The SSAR incorporated these new requirements by including an additional note on the P&ID's of the RHR system (SSAR Figure 5.4-10) and the HPCF system (SSAR Figure 6.3-7). This is acceptable; therefore, the staff concludes that GE has adequately addressed the potential problem described in Supplement 3 to Bulletin 88-08.

Potential Impact Number: 25540 **SRP Section Number:** 5.4.7

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

More generally, by NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," the staff is requesting 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 stresses from temperature stratification or temperature oscillations that could be induced by leaking valves.

Potential Impact Number: 25541 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 3

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Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

GE provided a certified design commitment that the ABWR piping, its appurtenances, and its supports must satisfy the ASME Code class, seismic category, and QG requirements commensurate with their classification. The COL applicant will review the ASME Code-required design documents for installed components to verify the completion of a certified stress report and related Code-required documents.

On the basis of this commitment for a COL applicant to verify the existence of Code-required documents, the staff concludes there is reasonable assurance that the piping and its subcomponents will be adequately designed, fabricated, and examined in accordance with the applicable ASME Code requirements.

Potential Impact Number: 25542 **SRP Section Number:** 4.2

Document: RG 1.70 **Revision:** 3
Title: STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS

Summary:

The description of the thermal design should include such items as maximum fuel and cladding temperatures, clad-to-fuel gap conductance as a function of burnup and operating conditions, and fuel cladding integrity. Cladding mechanical properties and thermal-physical properties should consider the effects of design temperature and irradiation on the properties.

Burnup and irradiation effects are required to be considered when developing material and mechanical properties models for the fuel and cladding. Presently, the specific criteria in SRP section 4.2 does not explicitly address burnup and irradiation effects.

Potential Impact Number: 25545 **SRP Section Number:** 5.4.2.1

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

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Summary:

ABB-CE states in CESSAR-DC Section 20.2.52 that it will comply with the following recommendations in GL 85-02, to the extent that they are applicable to the System 80+ design: prevention and detection of loose parts; SG tube in-service inspection (ISI); secondary water chemistry and impurity control; primary-to-secondary coolant leakage limit; use of Nitrogen-16 and area radiation monitors; primary coolant iodine activity limit; safety injection (SI) signal reset logic. ABB-CE also states that Inconel 690 will be used for SG tubes to provide increased resistance to corrosion. The staff finds that ABB-CE's proposed resolution to Issue A-4 for the System 80+ design is acceptable; however, ABB-CE will be subject to the staff's proposed applicable regulations on SG tube integrity which are addressed in Section 15.3.9 of this report.

Potential Impact Number: 25546 **SRP Section Number:** 5.4.2.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

The specifications for makeup water to the feedwater and condensate system are presented in Table 9.2.3-1 of the CESSAR-DC. In Amendments J and L, ABB-CE modified this table so that all water chemistry limits complied with the EPRI evolutionary plant requirements. On this basis, DSER Open Item 5.4.2-5 is resolved.

Potential Impact Number: 25547 **SRP Section Number:** 5.4.2.1

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

In the DSER, the staff maintained that the secondary water chemistry guidelines contained within the applicant's submittal in CESSAR Tables 10.3.5-1, 10.3.5-2, and 10.3.5-3 should also contain the recently published EPRI guidelines for the makeup water to steam generators as given below (Open Item 5.4.2-5): WATER CHEMISTRY EPRI GUIDELINE: pH- 7.0 (min), 7.5 (max); Conductivity at 25 deg C- 0.1 S/cm;

Sodium <3.0 ppb; Silica 10 ppb. In the FSER, the staff verified that the secondary water chemistry will comply with EPRI guidelines.

Potential Impact Number: 25548 **SRP Section Number:** 5.4.2.1

Document: EPRI NP-6239 **Revision:** 2 , 12/1/88
Title: PWR SECONDARY WATER CHEMISTRY GUIDELINES

Summary:

EPRI FORWARD

The importance of establishing and maintaining appropriate water chemistry conditions in PWR secondary systems cannot be overemphasized. During the 1970s, several unanticipated steam generator corrosion problems significantly reduced plant availability and increased personnel radiation exposure during inspection and maintenance. Formation of locally corrosive environments as a result of impurity ingress during operation contributed to these problems. Corrosion related turbine failures have also occurred. An appropriate water chemistry control program will limit this corrosion.

To assist the electric utility industry in establishing an acceptable water chemistry program, EPRI published the "PWR Secondary Water Chemistry Guidelines" in October 1982 (EPRI report NP 2704-SR). The first revision, published in June 1984 (EPRI report NP-5056-SR), incorporated experiences and technical information available at publication. The revised guidelines have received wide acceptance and use and have contributed to improved PWR secondary water chemistry.

The second revision incorporates significant changes based on plant experiences with alternative water chemistries and recent laboratory data relating to steam generator corrosion.

Potential Impact Number: 25549 **SRP Section Number:** 3.2.2

Document: RG 1.11 **Revision:** 0 , 3/10/71
Title: INSTRUMENT LINES PENETRATING

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Summary:

General Design Criteria 55 and 56 require that each line that penetrates primary reactor containment and is part of the reactor coolant pressure boundary or that is connected directly to the containment atmosphere have one automatic valve inside and one automatic valve outside containment "unless it can be demonstrated that the design is acceptable on some other defined basis." This guide describes a suitable basis which may be used to implement General Design Criteria 55 and 56 for demonstrating the acceptability of a particular group of these lines, namely, instrument lines.

Position C.1.d states that sensing lines for instruments that are part of the protection system should be conservatively designed up to and including the isolation valve and of a quality at least equivalent to the containment. These portions of the lines should be located and protected so as to minimize the likelihood of their being damaged accidentally. They should be protected or separated to prevent failure of one line from inducing failure of any other line. Provisions should be included to permit periodic visual inservice inspection, particularly of those portions of the lines outside containment up to and including the isolation valve.

Position C.2 states that sensing lines for instruments that are not part of the protection system: should meet the provisions of position C.1 (including position C.1.d described above), or should be provided with one automatic isolation valve inside and one automatic valve outside containment. The valve outside should be located as close to containment as practical.

Potential Impact Number: 25550 **SRP Section Number:** 3.2.2

Document: RG 1.141 **Revision:** 0
Title: CONTAINMENT ISOLATION PROVISIONS FOR FLUID SYSTEMS

Summary:

This guide describes a method acceptable to the NRC staff for complying with the Commission's requirements with respect to containment isolation of fluid systems. The requirements and recommendations for containment isolation of fluid systems that penetrate the primary containment of light-water-cooled reactors as specified in ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems," are generally acceptable and provide an adequate basis for complying with the pertinent containment

isolation requirements of Appendix A to 10 CFR Part 50, subject to the positions detailed in section C of the Reg. Guide.

The positions which appear most relevant to quality/safety classification of provisions for containment isolation of fluid systems appear to be positions C.4 and C.6.

Position C.4 states that Section 4.4.8 of ANSI N271-1976 gives general design requirements for closed systems. In addition, all branch lines and their isolation valves in closed systems both inside and outside the containment should meet the design criteria of Section 3.5 or Section 3.6.7 if the branch lines constitute one of the containment isolation barriers.

Position C.6 states that Section 4.14 of ANSI N271-1976 states: "The piping between isolation barriers or piping which forms part of isolation barriers shall meet the requirements of 3.7 and applicable requirements for isolation barriers." Piping between isolation barriers should meet the applicable requirements of Section 3.5 or Section 3.7.

Potential Impact Number: 25551 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In Q430.36, the staff asked GE to identify the systems and the relevant interface requirements for containment isolation valves (CIVs) not within the ABWR scope and discuss essential and nonessential systems in accordance with RG 1.141, "Containment Isolation Provisions for Fluid Systems," Revision 0. GE stated that all isolation valves are within the scope of the ABWR standard plant. However, it did not provide containment isolation provisions for the essential and nonessential systems. This was identified as Open Item 15 in the DSER (SECY-91-355).

In response to DSER Open Item 15, GE stated that containment isolation provisions for the isolation valves were specified in SSAR Table 6.2-7, which gives information on CIVs on a system-by-system basis. However, the table did not identify the system lines as essential or nonessential to address the Three Mile Island (TMI) action plan requirements (NUREG-0737, Item II.E.4.2). RG 1.141 contains guidance on the classification of essential versus nonessential systems. Each nonessential penetration

(except instrument lines) is required to meet GDC 54, 55, 56, and 57 and will be isolated automatically by the containment isolation signal. Essential systems, such as systems related to engineered safety features (ESF) or systems needed for safe shutdown of the plant, may include remote-manual CIVs.

The staff finds that SSAR Table 6.2-7 contains the valve information on a system-by-system basis with containment isolation provisions for both ESF and non-ESF systems. It concludes that the valve actuation for the ESF system reflects the function of the essential system, and is acceptable. Therefore, DSER Open Item 15 is resolved.

Potential Impact Number: 25552 **SRP Section Number:** 3.2.2

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In CESSAR-DC Section 6.2.4, ABB-CE fails to commit to the following standards and regulations specified in SRP Section 6.2.4:

- RG 1.11: instrument line containment isolation

- RG 1.26: quality group classifications

- RG 1.141: containment penetration system classification

- BTP CSB 6-4: containment purging during normal operation

In the DSER, the staff stated that ABB-CE must clarify whether or not the CESSAR-DC will conform to each of the regulations and standards listed above. This was designated as DSER Open Item 6.2.4-1. In Amendments N and Q, ABB-CE committed in CESSAR-DC Sections 6.2.4.1.2 and 9.4.6.1, and Table 6.2.4.1 to complying with these requirements. On this basis, DSER Open Item 6.2.4.1 is resolved.

In CESSAR-DC Table 6.2.4-1, ABB-CE lists all CIVs with the associated location, arrangement, position, actuation, and test information for each valve. Designated as DSER Open Item 6.2.4-2, this table and its accompanying notes and text should include the following information:

- identification of essential and nonessential systems as described in RG 1.141

In subsequent amendments, ABB-CE placed the above requested information in CESSAR-DC Table 6.2.4-1.

Potential Impact Number: 25553 **SRP Section Number:** 4.2

Document: C&S: ANS 5.4
Title: METHOD FOR CALCULATING THE FRACTIONAL RELEASE OF VOLATILE FISSION PRODUCTS FROM OXIDE FUEL

Summary:

The standard provides an analytical method for calculating the release of volatile fission products from oxide fuel pellets during normal reactor operation. When used with nuclide yields, the method gives the "gap activity," which is the inventory of volatile fission products that could be available for release from the fuel rod if the cladding were breached.

Potential Impact Number: 25554 **SRP Section Number:** 4.2

Document: NRC LETTER From Tourigay, E.G. to Woody, C.O.
Title: AMENDMENT NO. 21 TO FACILITY OPERATING LICENSE NO. NPF-16, DOCKET NO. 50-389

Summary:

The ANS 5.4 release model has been developed by a panel of experts and adopted as an ANSI/ANS Standard "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel" under steady-state temperature conditions. The NRC has given a de facto approval for the use of the ANS 5.4 model to determine the radioactive release fractions that exist in the gap of fuel rods under steady-state operating conditions and, thus, this model is found acceptable for this licensing application.

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Potential Impact Number: 25555 **SRP Section Number:** 3.7.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This task responds to recommendations from the Piping Review Committee. It proposes to revise Regulatory Guides 1.61 and 1.84, and SRP Section 3.9.2 to incorporate ASME Code Case N-411-1 and/or new positions on pipe damping for high-frequency loads and for time-history analyses. This is a regulatory potential impact issue.

Potential Impact Number: 25556 **SRP Section Number:** 3.7.2

Document: C&S: ASME N-411 **Revision:** 1 , 2/20/89
Title: ALTERNATIVE DAMPING VALUES FOR RESPONSE SPECTRA ANALYSIS OF CLASSES 1, 2, AND 3 PIPING, SECTION III, DIVISION 1

Summary:

This code case allows damping values which are slightly less restrictive (i.e., slightly higher damping values) than the ones provided in Appendix N to ASME Code Section III, Division 1. The Code Case has not been completely accepted in its entirety, but if certain restrictions, provided in Regulatory Guide 1.84, are applied, Code Case N-411-1 may be used as an alternate source of damping values for seismic analysis of Class 1, 2, and 3 piping sections under Section III, Division 1. Neither this Code Case nor Regulatory Guide 1.84 appear in SRP 3.7.1 Revision 2. The SRP only allows damping values from Regulatory Guide 1.61 or from test data, approved on a case-by-case basis.

Potential Impact Number: 25557 **SRP Section Number:** 3.7.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

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Summary:

In section 3.9.2.2 , "Seismic Subsystem Analysis," the staff found the use of ASME Code Case N-411-1, as conditioned by Regulatory Guide 1.84, to be acceptable. Specifically:

Regulatory Guide 1.61 contains recommended values of damping to be used in the seismic analysis of structures, systems, and components. In addition, Regulatory Guide 1.84, Revision 25 (May 1988), conditionally endorses ASME Code Case N-411-1. The damping values used by General Electric are the same as those specified in either Regulatory Guide 1.61 or ASME Code Case N-411-1, pursuant to Regulatory Guide 1.84. These criteria are acceptable.

Potential Impact Number: 25558 **SRP Section Number:** 5.4.7

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

SECY 93-087 addressed the mid loop issue and although no changes to the positions of SECY 90-016 were indicated, SECY 93-087 provides insight into the NRC's current approach to the more general issue of assessing shutdown and low-power risks. Specifically, SECY 93-087 states that the staff has requested EPRI and ALWR applicants to assess shutdown and low-power risk, identifying design-specific vulnerabilities and weaknesses and documenting their consideration and incorporation of design features that minimize such vulnerabilities.

Potential Impact Number: 25559 **SRP Section Number:** 5.4.7

Document: STAFF REQ. MEMO 9007160185
Title: SECY-90-016 - EVOLUTIONARY LIGHT WATER REACTOR CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

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Summary:

The Commission (with all Commissioners agreeing) has approved the staff's proposed position, with the ACRS recommendation of April 26, 1990, that four additional specific requirements be considered for mid-loop operation.

Potential Impact Number: 25560 **SRP Section Number:** 5.4.7

Document: STAFF REQ. MEMO 9007160185
Title: SECY-90-016 - EVOLUTIONARY LIGHT WATER REACTOR CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

The Commission approved the staff's position on intersystem LOCA, provided that all elements of the low-pressure system are considered (including instrument lines, pump seals, heat exchanger tubes, and valve bonnets).

Potential Impact Number: 25561 **SRP Section Number:** 5.4.7

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

SECY 93-087 addressed the intersystem LOCA issue and although no changes to the positions of SECY 90-016 were indicated, other than the extension to passive plants, SECY 93-087 provides additional information and clarification.

Potential Impact Number: 25562 **SRP Section Number:** 3.7.2

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Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff's review of the seismic analysis of the seismic Category I SSCs includes the seismic analysis methods and acceptance criteria used for the ABWR seismic Category I structures, the RPV (reactor pressure vessel) and containment internal structures design. The radwaste building, which is not a seismic Category I structure and does not house any safety related equipment and systems, is included in this review because GE elected to design this structure for the SSE seismic loads to ensure that the embedded portion of the building retains its structural integrity during and after and SSE, and so to prevent unacceptable leakage of the radwaste material outside the building.

The seismic analyses of the turbine building and the condenser, which is located inside the turbine building but has a separate foundation, are also included in this review although these two structures are not of seismic Category I. They are within the scope of this review because a portion of the MSL (main steamline) system, which is required to be designed for the SSE based on the discussion in Section 3.2.1 of this report, is inside the turbine building, and one end of this portion of the MSL system and branch lines is anchored at the turbine building and the condenser.

Potential Impact Number: 25563 **SRP Section Number:** 3.7.3

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

The CEFSER states that:

"By DSER Open Item 3.2.1-1, the staff requested that ABB-CE clarify the use of the term 'restrained as required.' Amendment N to CESSAR-DC Sections 3.7.3.13(A) and 3.7.3.13(C) stated that the attached non-Category I piping, up to the first anchor beyond the interface, is designed in such a manner that during an earthquake of SSE intensity it will not cause a failure of the Category I piping. These commitments are in

conformance with RG 1.29, and thus, acceptable. On this basis, DSER Open Item 3.2.1-1 is resolved."

Potential Impact Number: 25564 **SRP Section Number:** 3.7.3

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

The CEFSER states (in 3.1.1) that:

"The elimination of the OBE from the analysis and design of ASME Code Class 1, 2, and 3 components and core support structures requires all current OBE design-related checks to be performed for the SSE. With regard to primary stress effects (seismic inertial stresses), the elimination of the OBE from Service Level B analysis could have a potential impact on design in those cases where the load combination includes other dynamic loadings (i.e., operational transients) in Service Level B but not in Service Level D. The staff explored this possibility specifically for piping systems and found that when the OBE is established at one-third of the SSE, the load combinations. Although cases have been found where the OBE controlled the design, they were typically found when other normal operating loads were large and did not result in a different hardware configuration. Therefore, for primary stresses in piping systems, the staff finds that eliminating the OBE from piping stress load combinations will not reduce existing safety margins because the load combination with the SSE loading is generally controlling."

Potential Impact Number: 25565 **SRP Section Number:** 3.7.3

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

"Changes to Appendix A to 10 CFR Part 100 are being proposed to redefine the OBE such that the OBE criterion can be satisfied without the need to perform explicit design

and response analyses, if the OBE is equal to or less than one-third of the safe-shutdown earthquake (SSE). The proposed amendment to 10 CFR Part 100, Appendix A, would also allow, as an option, that the OBE be eliminated from design certification when the OBE is established at less than or equal to one-third of the SSE. In this manner, the OBE serves the 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. In addition, the staff notes that Appendix A to 10 CFR Part 100 requires that the maximum vibratory ground acceleration of the OBE be at least one-half the maximum vibratory ground acceleration of the SSE. Therefore, deviations from this regulation must be evaluated to ensure that 10 CFR 50.12 exemption criteria are satisfied when the OBE is eliminated from design analysis and is redefined only as an inspection level earthquake with the maximum vibratory ground acceleration at one-third of that of the SSE."

Potential Impact Number: 25566 **SRP Section Number:** 3.7.3

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

"With the elimination of the OBE, and in order to maintain the equivalent qualification requirements in IEEE 344-1987 to qualify equipment with the equivalent of five OBE events followed by one SSE event, the staff recommended, in SECY-93-087, that equipment be qualified with five half-SSE events followed by one full SSE event. Alternatively, the staff recommended a number of fractional peak cycles equivalent to the maximum peak cycles for five half-SSE events may be used in accordance with Appendix D of IEEE 344-1987 when followed by one full SSE. In CESSAR-DC Section 3.7.3.2 and Section 3.10, ABB-CE commits to the above staff recommendations as stated in SECY-93-087. Thus, the CESSAR-DC criteria for equipment seismic qualification, when the OBE is eliminated from design, are acceptable."

Potential Impact Number: 25567 **SRP Section Number:** 3.7.3

Document: FINAL SER CE80 CH 3

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Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

On the number of earthquake cycles:

"By Amendment R to CESSAR-DC Section 3.7.3.2, ABB-CE revised the number of earthquake cycles for the System 80+ design to be a total of two SSE events with 10 maximum stress cycles per event. In SECY-93-087, the staff recommended the elimination of the OBE in the design process on the basis that it would not result in a significant decrease in the overall plant safety margin. The detailed basis for the staff's recommendation is discussed in Section 3.1.1 of this report, which also defines acceptable number of earthquake cycles or alternatives in fatigue evaluations. The revised CESSAR-DC Section 3.7.3.2 (Amendment R) is consistent with the staff position discussed in Section 3.1.1 of this report. This is, therefore, acceptable."

Potential Impact Number: 25568 **SRP Section Number:** 3.7.3

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

On the number of earthquake cycles:

"In order to ensure adequate design considerations for the fatigue effects of earthquake cycles, it is necessary to establish a bounding load definition and number of earthquake cycles to account for the more frequent occurrences of lesser earthquakes and their aftershocks. In CESSAR-DC Section 3.7.3.2, ABB-CE uses a cyclic load basis for fatigue analysis of earthquake loading for ASME Code Class 1, 2, and 3 components and core support structures equal to two SSE events with 10 maximum stress cycles per event (20 full cycles of the maximum SSE stress range). This is equivalent to the cyclic load basis of one SSE and five OBE events as currently recommended in the Standard Review Plan (SRP) (NUREG-0800) Section 3.9.2. Alternatively, an equivalent number of fractional vibratory cycles to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE 344-1987."

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Potential Impact Number: 25569 **SRP Section Number:** 3.7.3

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

"In SECY-93-087, the staff discussed the safety impact of eliminating the OBE as it pertains to civil structures, piping systems, and equipment seismic qualification. The staff made several recommendations to ensure that eliminating the OBE would not result in a significant decrease in the overall plant safety margin. In its July 21, 1993, SRM, the Commission approved and agreed with the following staff positions:

Use two SSE events with 10 maximum stress cycles per event to account for earthquake cycles in the fatigue analyses of piping systems performed until the new guidance is issued; alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D to IEEE 344-1987.

The effects of anchor displacements in the piping caused by an SSE be considered with the Service Level D limit."

Potential Impact Number: 25570 **SRP Section Number:** 3.7.3

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

On elimination of OBE as design requirement:

"Changes to Appendix A to 10 CFR Part 100 are being proposed to redefine the OBE such that the OBE criterion can be satisfied without the need to perform explicit design and response analyses, if the OBE is equal to or less than one-third of the safe-shutdown earthquake (SSE). The proposed amendment to 10 CFR Part 100,

Appendix A, would also allow, as an option, that the OBE be eliminated from design certification when the OBE is established at less than or equal to one-third of the SSE. In this manner, the OBE serves the 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."

Potential Impact Number: 25571 **SRP Section Number:** 6.3

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

This TMI item required all GE plants to review data on ECC system outages to determine if cumulative outage time limitations should be incorporated in the Technical Specifications. The advanced SER stated that cumulative outage times were not required to be in the ABWR technical specifications, but would be implemented in the plant administrative procedures. The DFSER, also reported that SSAR Section 1.9 established an action item for the COL applicant to report ECCS outages in annual summary reports to the NRC. The staff verified that GE established a COL action item to prepare and submit annual reports on ECCS unavailability that also include information on outage dates, lengths, and causes; ECCS components involved; and any corrective action taken.

Potential Impact Number: 25572 **SRP Section Number:** 14.2

Document: NRC LETTER From Ziemann, DL to Miraglia, FJ
Title: SHEARON HARRIS NUCLEAR POWER PLANTS UNITS 1, 2, 3 AND 4
SPECIAL LOW POWER TEST PROGRAM - TMI ACTION PLAN ITEM I.G.1

Summary:

Although not transmitted to all PWR applicants and permit holders, this letter provides guidance for PWR OL applicants with respect to the requirements of TMI Action Plan Item I.G.1. The letter discusses PWR special low power testing consisting of natural circulation tests which are acceptable to comply with TMI Item I.G.1 and the associated operator training and procedure validation to be conducted during such testing. The

letter indicates that an enclosed Westinghouse letter describing a revised special low power test program complies with the staff's objectives with respect to TMI Item I.G.1 and is thus provided for all PWR OL applicants. The enclosed Westinghouse letter NS-EPR-2465 dated July 8, 1981 to H. R. Denton describes the special low power testing program reviewed and accepted by the staff and indicated as acceptable for all PWR OL applicants.

Potential Impact Number: 25573 **SRP Section Number:** 6.3

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

In accordance with NRC Bulletin 88-08, the staff is requesting that licensees and applicants review systems connected to the reactor coolant system (RCS) to determine if any sections of such piping that cannot be isolated can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves.

In response to this request, GE stated that the ABWR systems connected directly to the RCS or the RPV are the nuclear boiler system and the emergency core cooling systems. In the nuclear boiler system, GE reviewed the feedwater subsystem that supplies makeup water to the RPV to confirm that design requirements for temperature stratifications of feedwater piping were satisfactorily defined in system specifications and piping cycle diagrams. In the design of the ECCS, both the residual heat removal (RHR) system and high-pressure core flooder (HPCF) have piping that are directly connected to the RPV. In the unisolable sections of RHR piping, leaking toward the RPV cannot occur because the pressure will always be higher on the reactor side during normal plant operation when the upstream pumps are not operating.

In the DFSER, the staff concluded that GE had adequately addressed the potential problems described in Bulletin 88-08. Subsequently, however, the staff noted that GE did not address the potential problem described in Supplement 3 to Bulletin 88-08, although Supplements 1 and 2 were addressed adequately. It involved the development of potential cyclic stratified flow and associated thermal striping that may occur because of possible leakage past the valve disk and out the valve stem packing gland. This flow stratification and striping may occur when the pressure on the upstream side of the valve is less than the RPV system pressure during normal

operation. Supplement 3 reported an incident in which this phenomenon induced thermal fatigue, which led to a through-wall pipe crack in a foreign reactor. The staff reviewed the SSAR and requested that GE provide additional information to address this concern. GE then identified the ABWR piping sections that could be susceptible to unacceptable thermal stresses, owing to this phenomenon. They include sections of the RHR system and the HPCF system. For the affected piping sections, GE will require that either (1) the gate valve in each of the unisolable piping sections be located at a distance equal or greater than 25 pipe diameters from the RPV nozzle or (2) stress analysis be performed to show that stresses and fatigue from potential stratification and thermal striping are acceptable per the ASME Code. The SSAR incorporated these new requirements by including an additional note on the P&ID's of the RHR system (SSAR Figure 5.4-10) and the HPCF system (SSAR Figure 6.3-7). This is acceptable; therefore, the staff concludes that GE has adequately addressed the potential problem described in Supplement 3 to Bulletin 88-08.

Potential Impact Number: 25574 **SRP Section Number:** 6.3

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

More generally, by NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," the staff is requesting 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 stresses from temperature stratification or temperature oscillations that could be induced by leaking valves.

Potential Impact Number: 25575 **SRP Section Number:** 6.3

Document: FINAL SER ABWR CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

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Since the RCIC system is a part of the ECCS, full-flow testing capability is provided using the safety-related suction source (i.e., suppression pool).

The RCIC system includes a full-flow test line with water return to the suppression pool for periodic testing. The periodic tests will be performed according to ASME Code, Section XI, as required by the TSs. The staff requires, for any emergency core cooling system (ECCS), that the TSs include a system functional test at least every refueling outage, with simulated automatic actuation and verification of proper automatic valve position to verify that the RCIC pump will develop a minimum flow of 182 m3/hr (800 gpm).

The RHR system consists of three independent loops - subsystems A, B, and C. Each loop contains a motor-driven pump, heat exchanger, piping, valves, instrumentation and controls. Each loop can take suction from either the RPV or the suppression pool and will be capable of discharging water to either the RPV or back to the suppression pool through a full-flow test line.

Potential Impact Number: 25576 **SRP Section Number:** 6.3

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff's proposed applicable regulations, as described in Section 1.6 of this report, for inservice testing are as follows:

All pumps and valves of the standard design subject to the test requirements set forth in 10 CFR 50.55a(f) are subject to the following additional limitations and modifications.

(1) Piping design must incorporate provisions for full flow testing at maximum design flow of pumps and check valves;

(2) Check valve testing must incorporate the use of advanced non-intrusive techniques to address degradation and performance characteristics;

(3) Provisions must be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced non-intrusive techniques;and

(4) Provisions must be incorporated to test motor-operated valves under design basis differential pressure.

Potential Impact Number: 25577 **SRP Section Number:** 6.3

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

ABB-CE also stated in CESSAR-DC Section 3.9.6.1 that the System 80+ piping system design can accommodate IST at a flow rate at least as large as the maximum design flow for the pump.

In CESSAR-DC, Section 6.3.4, ABB-CE describes the testing requirements for the SIS. The SIS is designed and will be installed to permit the SIS pumps to be tested at full-flow conditions with the reactor at power, which is consistent with the EPRI URD requirements. The COL applicant will perform preoperational testing of the SIS to verify that the system and components perform according to design criteria. The COL applicant will periodically test and inspect the SIS components and subsystems in accordance with the ASME Code, Section XI, which constitutes compliance with GDC 36, to ensure the SIS will operate properly in the event of an accident. The COL applicant will also test such SIS components as pumps and automatic valves to confirm acceptable performance of each active component in the SIS.

Potential Impact Number: 25578 **SRP Section Number:** 6.3

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

In its SRM of June 26, 1990, on SECY-90-016, the Commission approved the staff's recommendations. The staff's proposed applicable regulation, as described in Section 1.6 of this report, for IST, is as follows:

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(1) Piping design must incorporate provisions for full flow testing at maximum design flow of pumps and check valves;

(2) Check valve testing must incorporate the use of advanced non- intrusive techniques to address degradation and performance characteristics;

(3) Provisions must be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation which cannot be detected through the use of advanced non-intrusive techniques; and

(4) Provisions must be incorporated provisions to test MOVs under design-basis differential pressure.

Potential Impact Number: 25579 **SRP Section Number:** 6.3

Document: STAFF REQ. MEMO 9007160185
Title: SECY-90-016 - EVOLUTIONARY LIGHT WATER REACTOR CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

Inservice Testing of Pumps and Valves:

The Commission has approved the staff's position as supplemented in their April 27, 1990 response to the ACRS comments. The Commission notes that due consideration should be given to the practicality of designing testing capability, particularly for large pumps and valves.

Potential Impact Number: 25580 **SRP Section Number:** 6.3

Document: STAFF REQ. MEMO 9308270107
Title: SECY-93-087 - POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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I.N Inservice Testing of Pumps and Valves:

The Commission has no objection to the staff's position, but understands that further elaboration on this issue will be forthcoming from the staff.

Potential Impact Number: 25581 **SRP Section Number:** 6.3

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In SECY 93-087 the staff clarified its position on inservice testing of pumps and valves as follows:

- The staff recommended that full flow testing be conducted at maximum design flow with analysis to extrapolate to design pressure if it is not practicable to conduct the inservice pump testing at design flow and pressure.

- For valves the staff recommended, a qualification test (under design-basis differential pressure) be conducted prior to installation and inservice valve tests be conducted under the maximum practicable differential pressure and flow when it is not practicable to achieve design-basis differential pressure during an inservice test.

Potential Impact Number: 25582 **SRP Section Number:** 6.3

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

In response to the staff's policy concern about the adequacy of minimum-flow systems for safety-related pumps, GE states in SSAR Section 3.9.6.1 that safety-related pumps and piping configurations can accommodate inservice testing at a flow rate at least as large as the maximum design flow for the pump. In addition, GE also stated in the

SSAR that it would evaluate the sizing of each minimum recirculation flow path to ensure that its use under all analyzed conditions will not result in degradation of the pump and to periodically measure the minimum recirculation flow rate to verify that it is in accordance with the design specification.

Potential Impact Number: 25583 **SRP Section Number:** 6.3

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

In response to DSER Confirmatory Item 3.9.6.1-1 regarding the adequacy of minimum-flow systems for safety-related pumps, ABB-CE stated in Section 3.9.6.1 of the CESSAR-DC, Amendment R, that safety-related pumps and piping configurations can accommodate IST at a flow rate at least as large as the maximum design flow for the pump. With respect to minimum pump flow operation, the sizing of each minimum recirculation flow path is evaluated to ensure that its use under all analyzed conditions will not degrade the pump. The flow rate through minimum recirculation flow paths can also be periodically measured to verify that it is in accordance with the design specification.

Potential Impact Number: 25584 **SRP Section Number:** 6.3

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In response to the staff's questions regarding the SI operation at low-flow conditions as described in NRC Bulletin 88-04, ABB-CE stated that the arrangement of the SIS mini-flow recirculation lines precludes pump deadhead operation resulting from pump-to-pump interactions as discussed in NRC Bulletin 88-04. The only cross-connection in the SIS arrangement ties two discharge (mini-flow) recirculation lines together downstream of an orifice and a check valve, creating a flow path to the vented IRWST. This piping arrangement will not result in the operation of one SI pump

to cause other SI pumps to operate at a flow rate that is lower than the required minimum for pump protection. ABB-CE also confirmed with information from pump manufacturers and utilities, that the minimum recirculation flow previously established for SI pumps applies to the System 80+ design and is adequate to protect pumps at low flow conditions. The design minimum SI pump flow ranges from 322 to 397 L/minutes (85 to 105 gpm) (9.7 percent to 12 percent of best efficiency flow). The staff agrees that the proposed SIS arrangement will avoid pump deadhead operation. However, the staff required that ABB-CE discuss design criteria for the minimum SI recirculation flow, and submit pump operating data or test results demonstrating that the SI pump would be operable for an extended time at the proposed flow range. ABB-CE should also demonstrate in completing the ITAACs that the mini-flow bypass has adequate capability. This was designated as DSER Open Item 6.3.3-1.

Potential Impact Number: 25585 **SRP Section Number:** 4.3

Document: FINAL SER CE80 CH 4
Title: REACTOR

Summary:

If a control rod were ejected, the worth of a control rod would be no worse than that worth assumed in the accident analysis.

Therefore, the CEA and soluble boron worths are acceptable for use in the accident analysis.

ABB-CE has satisfied the requirements of GDC 28 (with respect to postulated reactivity accidents) by satisfying the following:

(a) regulatory position in RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

(b) fuel enthalpy limit of 280 cal/gm as shown in Section 15.3.4 of this report, and

(c) criteria on the capability to cool the core as stated in SRP Section 4.2, and

(d) using calculational methods that have been found acceptable for reactivity insertion accidents as shown in Table 15.1-3 of this report.

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Potential Impact Number: 25586 **SRP Section Number:** 6.3

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

To protect against an intersystem LOCA (ISLOCA), designers of future ALWR plants should reduce the possibility of a LOCA outside 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 normal RCS operating pressure.

Potential Impact Number: 25587 **SRP Section Number:** 6.3

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

The Commission approved interfacing-systems loss-of-coolant accident (ISLOCA) position (SECY 90-016) requires that the evolutionary advanced light water reactor (ALWR) designs (including the System 80+) reduce the possibility of a LOCA outside containment by designing to the extent practicable all systems and subsystems connected to the RCS to withstand the full RCS pressure.

Potential Impact Number: 25588 **SRP Section Number:** 3.2.2

Document: NRC MEMORANDUM 11/16/89, from E.S. Beckjord to F.P. Gillespie
Title: 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'

Summary:

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Enclosure 4 of this memorandum specifies recommended SRP Section 3.2.2 changes related to resolution of Generic Issue 70 including implementation information regarding which PWR applications should classify PORVs as safety-related.

Potential Impact Number: 25590 **SRP Section Number:** 5.4.2.1

Document: RG 1.65 **Revision:** 0
Title: MATERIALS AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS

Summary:

High-strength low-alloy reactor stud bolting may be obtained through closely controlled quenching and tempering procedures on grades of steel such as American Iron and Steel Institute (AISI) 4140 and 4340 (Ref. 1). These steels are approved by the ASME as bolting materials and are listed under Section 11, "Material Specifications,"/2/ of the ASME Code as SA-540 Grade B-23 and B-24 bar, SA-193 Grade B-7 bar, SA-320 Grade 193 bar, and SA-194 Grade 7 (nuts for bolting). Additional requirements supplementing these basic materials specifications are found in SA-614 entitled "Specification For Special Requirements for Bolting Materials For Nuclear and Other Special Applications." Proper control of the tempering procedure is necessary to obtain the desired balance of mechanical properties. The above-mentioned stud materials when tempered to a maximum tensile strength of 170 ksi (Ref. 2) are relatively immune to stress corrosion cracking (SCC). Above this strength level the alloy becomes increasingly susceptible to SCC. Therefore, design conservatism should be exercised in determining the sizing of the studs so that the minimum specified strength level of the material selected will not result in a measured ultimate tensile strength exceeding 170 ksi. _____

1. Bolting Materials

a. Reactor vessel closure stud bolting should be fabricated from materials which have adequate toughness throughout the life cycle of the reactor. Stud bolting should meet the requirements set forth in Subsection NB, "Requirements for Class 1 Components," Section III of the ASME Code. Bolting materials should meet the requirements of one of the following ASME specifications: (1) SA-540 Grade B-23 and B-24 bar (AISI 4340). (2) SA-193 Grade B-7 bar (AISI 4140, 4142, 4145). (3) SA-194 Grade 7 (nuts for bolting) (AISI 4140, 4142, 4145). (4) SA-320 Grade L43 bar (AISI 4340). b. The requirements of the specification in paragraph C.1.a. should be supplemented by the

following: (1) The maximum measured ultimate tensile strength of the stud bolting material should not exceed 170 ksi.

Potential Impact Number: 25591 **SRP Section Number:** 4.5.1

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In Chapter 1, paragraph 5.3.1.1.5 of the URD, EPRI states that although the stated controls on grinding on stainless steel may be desirable for use on PWR components, they are not required for avoiding IGSCC. ABB-CE intends to avoid fabrication processes which would severely cold work the surface of austenitic stainless steel components. Grinding is done with resin-bonded or rubber bonded, aluminum oxide or silicon carbide wheels which were not previously used on materials other than austenitic alloys. This will offer reasonable assurance of stainless steel component integrity, and that stainless steel components in contact with reactor coolant are not likely to crack. On this basis, DSER Open Item 5.2.3-11 was resolved.

In FSER Chapter 6, Section 6.1.1, the staff indicates that the COL applicant will perform grinding with resin-bonded or rubber-bonded aluminum oxide or silicon carbide wheels that have been used only on Type 300-series stainless alloys. Grinding wheels bonded with rubber compositions that include halides or sulfur will not be used on austenitic stainless steels.

Potential Impact Number: 25592 **SRP Section Number:** 4.5.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

Section 3.3 of Chapter 1 of the EPRI Utility Requirements Document states that the plant design life for the ALWR will be 60 years. This proposed design life raises questions relative to the margins available in the current ASME fatigue design curves. These margins were established almost 30 years ago and were obtained from best-fit

curves of fatigue test data by applying a factor of either 2 on stress or 20 on cycles, whichever was more conservative at each point. These factors were originally intended to cover such effects as environment, size effect, and scatter of data. However, on the basis of limited data currently available, the staff concludes that these margins may not be sufficient to account for variations in the original fatigue test data as a result of various environmental effects. In its letter dated May 17, 1991, the staff requested a commitment in Section 4.7.3 of Chapter 1 to consider such effects in the designs of applicable ASME Code Class 1 systems, components, and equipment. In its letter dated August 1, 1991, EPRI stated that if additional data or research results yield findings requiring changes to the current ASME fatigue design curves, the code consensus process will provide the proper vehicle to address such findings. The staff does not agree with all of the discussions in this response and concludes that the above commitment is not completely acceptable. The ASME Code curves may not be revised for many years. Therefore, the staff's position is that until these curves are revised, all ALWR applicants and all licensees applying for license renewal should propose appropriate fatigue design curves that will be reviewed by the staff. For the Evolutionary Requirements Document, a commitment to this position would be sufficient. Pending such a commitment, the staff will evaluate this issue during its review of an individual application for FDA/DC. On the basis of the above discussion, this issue is closed.

The staff is assessing all available data relative to environmental effects on fatigue for austenitic stainless steel and ferritic steel. The objective of this effort is to propose interim fatigue curves that may be used in lieu of the current ASME curves until ASME has acceptably addressed this concern. Pending staff implementation of these interim curves, the position stated above will be in effect.

Potential Impact Number: 25593 **SRP Section Number:** 6.3

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

Paragraph (1)(vii) of 10 CFR 50.34(f) requires performance of a feasibility and risk assessment study to determine the optimum ADS design and modifications that would eliminate the need for manual activation to ensure adequate core cooling for BWRs.

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Potential Impact Number: 25594 **SRP Section Number:** 6.3

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

Paragraph (1)(x) of 10 CFR 50.34(f) requires the performance of a study to ensure that the ADS valves, accumulators, and associated equipment and instrumentation of BWRs will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety-related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves.

Potential Impact Number: 25595 **SRP Section Number:** 6.3

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

Paragraph (1)(xi) of 10 CFR 50.34(f) requires provision of an evaluation of depressurization modes, other than full actuation of the ADS, such as early depressurization with one or two SRVs, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown of BWRs. Slower depressurization would reduce the possibility of exceeding vessel integrity limits by rapidly cooling down.

Potential Impact Number: 25596 **SRP Section Number:** 13.6

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

In addition to requiring communication with offsite law enforcement authorities, in 10 CFR 73.55(f), the staff requires intraplant security communications. The intraplant security communications requirement (10 CFR 73.55[f][1]) specifies that plant security

officers must be capable of continuous communication with the central and secondary alarm stations. In the CESSAR, the applicant addressed a security radio system for offsite communications, but not for onsite communications with security officers.

Based upon EPRI guidance for ALWR communications systems, the staff maintained that the plant designer should ensure that effective communication is not impeded by problems of transmission through thick concrete walls, high-noise areas, personnel protective equipment, too few communication lines, or interference with or from other electronic or electrical equipment. The staff asked the applicant to detail consideration given to assuring the capability for security officers to be in continuous intraplant communication at System 80+ standard plants. In response, the applicant stated that plant security is expected to depend on wireless intraplant communications, but that the plant owner/operator will choose the system. The staff does not consider this response adequate since the applicant does not address design provisions that may be necessary to allow this capability, such as shielding of instrument transmitters and logic cabinets from radio frequency interference (RFI), fiber-optic cabling, and radio repeaters within buildings, and does not provide reasons for deviating from the EPRI ALWR URD. NRC Information Notice 83-83 stated: "As newer plants are built that use more solid state equipment...more cases of RFI by portable radio transmitters are likely to result.... If plant operations make the use of portable radio transmitters near RFI-sensitive equipment either necessary or likely in an emergency, then administrative prohibitions are not adequate and the licensee should consider hardware fixes." This was designated as DSER Open Item 9.5.2-3.

Subsequently, in Amendment J to CESSAR-DC, ABB-CE added the following statement in Section 13.6: "The security communications subsystem shall meet the following requirements:....Each on-site security officer, watchman, or armed response individual shall be provided with continuous communications with an individual in each continuously manned alarm station. This may be accomplished by using multi-frequency radio or microwave-transmitted two-way voice communications." Requiring continuous wireless communication between security officers, ensures adequate communication for the security organization. This additional information is acceptable to the staff. On this basis, DSER Open Item 9.5.2-3 is resolved.

Potential Impact Number: 25601 **SRP Section Number:** 4.3

Document: FINAL SER ABWR CH 4
Title: REACTOR

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Summary:

The moderation ratio increase tends to make the end-of-cycle low power (low temperature) moderator temperature reactivity coefficient less negative and possibly slightly positive. But the increase is not sufficient to cause an operational problem during startup or shutdown or a reactivity insertion problem, even assuming the maximum potential integrated positive reactivity is available for rapid insertion into the core.

Potential Impact Number: 25603 **SRP Section Number:** 6.8

Document: NRC MEMORANDUM 11/16/89, from E.S. Beckjord to F.P. Gillespie
Title: 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'

Summary:

This NRC memo transmits the draft of Generic Letter 90-06 issued to resolve Safety Issues 70, "Power Operated Relief Valve and Block Valve Reliability," and 94, "Additional Low-Temperature Overpressure Protection for Light Water Reactors," from the Office of Nuclear Regulatory Research to the Office of Nuclear Reactor Regulation. The memo states that the Generic Letter is for all PWR Licensees and CP holders. An enclosure to the memo (Enclosure 4) indicates that the Generic Letter will be applicable to all PWR operating reactors and all other PWR Plants (custom or standard) for which issuance of the OL is expected before 1994.

Potential Impact Number: 25604 **SRP Section Number:** 9.3.4

Document: FINAL SER CE80 CH 9
Title: AUXILIARY SYSTEMS

Summary:

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In the DSER, the staff asked that ABB-CE provide its basis for classifying the RCP seal injection function of the CVCS as non-safety related. This was designated as DSER Open Item 9.3.4-1.

Subsequently, ABB-CE added an additional system, the dedicated seal injection system (DSIS), to provide RCP seal injection. They also provided clarification on the classification of the system. As discussed in more detail in section 3.2.2 of this report, the staff's concerns have been addressed. DSER Open Item 9.3.4-1 is, therefore, resolved.

Potential Impact Number: 25605 **SRP Section Number:** 9.3.4

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

The staff required that ABB-CE submit adequate test data to demonstrate the ABB-CE KSB seal integrity during an SBO 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. This is DSER Open Item 20.2-7.

Since ABB-CE submitted a diverse and reliable DSIS design for RCP seal injection, and includes the DSIS in its D-RAP and O-RAP to ensure the high availability of the DSIS, the staff concludes that the design of DSIS in combination with the normal means (CCWS and charging pump seal injection) of RCP seal cooling will provide adequate RCP seal cooling and is acceptable. Therefore, DSER Open Item 20.2-7 is resolved.

Potential Impact Number: 25607 **SRP Section Number:** 5.4.7

Document: NRC GENERIC LETTER 89-04
Title: GUIDANCE OF DEVELOPING ACCEPTABLE INSERVICE TESTING PROGRAMS

Summary:

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In cases where only the minimum-flow return line is available for pump testing, regardless of the test interval, the staff's position is that flow instrumentation which meets the requirements of IWP-4110 and 4120 must be installed in the mini-flow return line. Installation of this instrumentation is necessary to provide flow rate measurements during pump testing so this data can be evaluated with the measured pump differential pressure to monitor for pump hydraulic degradation.

NRC Bulletin 88-04, dated May 5, 1988, advised licensees of the potential for pump damage while running pumps in the minimum-flow condition. The above guidelines for meeting the Code or performing alternative testing is not intended to superseded the thrust of this Bulletin. Licensees should ensure that if pumps are tested in the low flow condition, the flow is sufficient to prevent damage to the pump.

Potential Impact Number: 25608 **SRP Section Number:** 6.3

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

Issue A-43, in NUREG-0933, concerns the availability of adequate recirculation cooling water following a LOCA when long-term recirculation of cooling water from the PWR containment sump (or boiling-water reactor (BWR) RHR suction intake) must be initiated and maintained to prevent core melt following a postulated LOCA. This water must be sufficiently free of LOCA-generated debris and potential air ingestion so that pump performance is not impaired, thereby seriously degrading long-term recirculation flow capability.

The technical concerns evaluated under Issue A-43 are as follows:

- sump hydraulic performance under post-LOCA conditions resulting from potential vortex formation and air ingestion, and subsequent pump failure

- possible transport of large quantities of LOCA-generated insulation debris resulting from a pipe break to the sump debris screen(s), and the potential for sump screen (or suction strainer) blockage to reduce net positive suction head (NPSH) margin below that required for the recirculation pumps to maintain long-term cooling

- capability of RHR and containment spray system (CSS) pumps to continue pumping when subjected to possible air, debris, or other effects, such as particulate ingestion on pump seal and bearing systems

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Potential Impact Number: 25609 **SRP Section Number:** 6.3

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In 1985, the NRC issued RG 1.82, Revision 1, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," which contains guidance on the sizing criteria for ECCS strainers. Recent events at operating reactors involving the clogging of ECCS strainers have led the staff to conclude that the guidance in RG 1.82, Revision 1, may not be conservative enough to eliminate this concern. To address this issue for operating reactors, the NRC issued Information Notice (IN) 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR"; IN 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment"; Supplement 1 to IN 93-34; and Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers." The staff is still working on resolving this issue for operating reactors.

Potential Impact Number: 25610 **SRP Section Number:** 6.3

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

Because an event that occurs during shutdown and at low power has safety significance, the staff has determined that concerns with the shutdown operation should be satisfactorily addressed before the NRC issues a final design approval for the System 80+ design. ABB-CE was required to demonstrate adequate treatment of shutdown risk by submitting information, including (1) documentation showing adequate assessment of shutdown and low-power risk, with identification of design-specific vulnerabilities and weaknesses and (2) documentation showing consideration and

incorporation of design features that minimize shutdown and low-power risk vulnerabilities. The staff has reviewed the shutdown risk report submitted by ABB-CE and found it acceptable.

In addition, the staff asked ABB-CE to describe any deviations and a justification of the deviations from GL 88-17 recommendations regarding instrumentation, operating procedures, operator training programs, and TS implementations, addressing the concerns for reduced inventory operation.

Potential Impact Number: 25611 **SRP Section Number:** 6.3

Document: FINAL SER ABWR CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

In the FSER for the ABWR the staff reviewed the ABWR ADS as part of the ECCS design and verified its ability to reliably reduce reactor pressure to allow low pressure ECCS flows to enter the vessel in time to cool the core and satisfy the ECCS acceptance criteria.

Potential Impact Number: 25612 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 9
Title: AUXILIARY SYSTEMS

Summary:

The requirements of RG 1.72, "Spray Pond Piping Made from Fiberglass- Reinforced Thermosetting Resin," Revision 2, apply to the design of a spray pond as an UHS. The staff established Open Item 50 in the DSER because these requirements were not referenced in the SSAR submitted before the staff completed the DSER (SECY-91-235). SSAR Section 9.2.5.8 states that the COL applicant will submit information to show that all applicable requirements of RG 1.72 are met. This is acceptable and resolved Open Item 50.

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Potential Impact Number: 25613 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The seismic interface restraint shall provide a structural barrier between the seismic Category I portion of the MSL in the RB and the non-seismic Category I portions of the MSL in the turbine building. The seismic interface restraint shall be located inside the seismic Category I building. The classification of the MSL in the turbine building as non-seismic Category I is needed for consistency with the classification of the turbine building. Therefore, the staff positions related to the quality and safety guidelines imposed on the MSL from the outermost isolation valve up to the turbine stop valve are equivalent to the staff guidelines in SRP Section 3.2.2, Appendix A to 10 CFR Part 50, and RG 1.29.

Potential Impact Number: 25614 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The commitment to dynamically analyze the above components for the SSE is in Section 3.2.5.3, "Main Steamline Leakage Path" of the SSAR. The components are designed by using appropriate dynamic seismic analyses to withstand the SSE design loads in combination with other appropriate loads, within the limits specified. The mathematical model for the dynamic seismic analyses of the main steamlines and the branch line piping includes the turbine stop valves and piping to the turbine casing. The dynamic input for the analysis and design of the main steamlines are derived from a time history analysis (or an equivalent method) of the turbine building as described in SSAR Section 3.7.

In the DFSER, the staff noted that, to demonstrate the structural integrity of the main steam piping under SSE loading conditions, a dynamic analysis method should be used for this portion of the main steam piping. In addition, GE should discuss how the turbine building response spectra input to the main steam piping analysis will be generated when, as discussed in the following paragraph, no dynamic analysis method

of the turbine building is proposed. The staff position as delineated in SRP Section 3.7.2.5 for the generation of floor response spectra (FRS) is that the development of the FRS is acceptable if a time history approach is used. Alternative methods, other than the time history approach, used for generating FRS shall be submitted to the staff for review and approval on a case-by-case basis. This was DFSER Open Items 3.2.1-1 and 3.7.2-6. The resolution of this issue is discussed in the "Turbine Building" portion of Section 3.7.2 in the FSER.

SSAR Sections 3.7.2.8 and 3.7.3.13 state that equipment, structures, and piping in the ABWR that are non-seismic Category I but that could damage seismic Category I items if their structural integrity failed, are analyzed and designed to ensure their integrity is maintained under seismic loading from the SSE. At the interface between seismic and non-seismic Category I piping systems, the seismic analysis for the seismic Category I system 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. In the DFSER, the staff noted that this commitment is in conformance with RG 1.29. However, as a part of the resolution of this issue, the staff noted that the turbine building and any other applicable structure or equipment should be seismically analyzed for the SSE (RG 1.60 ground response spectrum anchored to 0.3g peak acceleration) to ensure that an earthquake will not adversely affect the structural integrity of the main steam piping, bypass line, valves, and instruments mounted on these pipes, and the main condenser. In Amendment 20 to the SSAR, GE added Section 3.7.3.16, which described a static, in lieu of dynamic, analysis for the turbine building. This proposed analysis was in accordance with the Uniform Building Code (UBC) Zone 2A, but was substantially less conservative than the specified SSE for which the main steam and bypass lines should be analyzed. The staff asked GE to provide a clear set of criteria for the turbine building, which is designed to UBC criteria, so that it will neither suffer no loss of function at the specified SSE nor adversely affect safety-significant piping, pipe-mounted equipment, or the condenser itself. This issue was DFSER Open Items 3.2.1-2 and 3.7.2-7. The resolution of this issue is discussed in the "Turbine Building" portion of Section 3.7.2 in the FSER.

Potential Impact Number: 25615 **SRP Section Number:** 3.2.2

Document: FINAL SER ABWR CH 1
Title: INTRODUCTION AND GENERAL DISCUSSION

Summary:

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The reactor coolant system (RCS) includes the nuclear boiler system; the main steam, feedwater, recirculation system; the reactor core isolation cooling (RCIC) system; the residual heat removal system; and the reactor water cleanup system. The design is different from current BWR designs in that 10 reactor internal pumps (RIPs) located within the reactor vessel will forcibly circulate reactor coolant. This will eliminate large piping connections to the reactor vessel below the core and also eliminate reactor recirculation system piping. Eighteen safety/relief valves in six groups will provide RCS over-pressure protection.

Potential Impact Number: 25616 **SRP Section Number:** 9.3.4

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

In response to DSER Confirmatory Item 3.9.6.1-1 regarding the adequacy of minimum-flow systems for safety-related pumps, ABB-CE stated in Section 3.9.6.1 of the CESSAR-DC, Amendment R, that safety-related pumps and piping configurations can accommodate IST at a flow rate at least as large as the maximum design flow for the pump. This approach is responsive to the guidelines on this issue contained in SECY 90-016 and meets Item 1 of the staff's recommendations, as an applicable regulation, on pump and valve IST issues as described in Sections 1.6 and 3.9.6 of this report. This is acceptable. In addition, ABB-CE also stated that the COL applicant will ensure that the pump specified for each application is not susceptible to inadequate minimum flow rate and inadequate thrust bearing capacity. With respect to minimum pump flow operation, the sizing of each minimum recirculation flow path is evaluated to ensure that its use under all analyzed conditions will not degrade the pump. The flow rate through minimum recirculation flow paths can also be periodically measured to verify that it is in accordance with the design specification.

Potential Impact Number: 25617 **SRP Section Number:** 6.3

Document: NRC GENERIC LETTER 89-04
Title: GUIDANCE OF DEVELOPING ACCEPTABLE INSERVICE TESTING PROGRAMS

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Summary:

In cases where only the minimum-flow return line is available for pump testing, regardless of the test interval, the staff's position is that flow instrumentation which meets the requirements of IWP-4110 and 4120 must be installed in the mini-flow return line. Installation of this instrumentation is necessary to provide flow rate measurements during pump testing so this data can be evaluated with the measured pump differential pressure to monitor for pump hydraulic degradation.

NRC Bulletin 88-04, dated May 5, 1988, advised licensees of the potential for pump damage while running pumps in the minimum-flow condition. The above guidelines for meeting the Code or performing alternative testing is not intended to superseded the thrust of this Bulletin. Licensees should ensure that if pumps are tested in the low flow condition, the flow is sufficient to prevent damage to the pump.

Potential Impact Number: 25618 **SRP Section Number:** 5.4.7

Document: NRC GENERIC LETTER 92-02 Revision Date: 3/6/92
Title: RESOLUTION OF GENERIC ISSUE 79, UNANALYZED REACTOR VESSEL (PWR) THERMAL STRESS DURING NATURAL CONVECTION COOLDOWN

Summary:

The U.S. Nuclear Regulatory Commission (NRC) is providing this letter to inform addressees of (1) the NRC's resolution of Generic Issue 79, "Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown" and (2) the conclusions reached by the staff as the result of the evaluations performed to resolve this generic issue. No new requirements are being established and no specific action or written response is required.

Potential Impact Number: 25619 **SRP Section Number:** 5.4.7

Document: FINAL SER CE80 CH 5
Title: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Summary:

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During the course of the review, the staff asked ABB-CE to submit the results of an analysis to demonstrate that the System 80+ plant can achieve cold shutdown during natural circulation conditions per the assumptions specified in BTP RSB 5-1. To satisfactorily resolve this item, the staff required ABB-CE to conduct an NCC test to confirm that the operator in the control room can control the plant cooldown by using the methods consistent with the methods assumed in the analysis. In response, ABB-CE addressed the operating margins relative to the NCC analysis, and identified the instrumentation available to the operator in the control room for carrying out the NCC operation (among other items). The staff completed its review and found that multiple channels and safety-grade instrumentation will be available to the operator for monitoring and controlling the NCC conditions from the control room.

Potential Impact Number: 25620 **SRP Section Number:** 6.3

Document: NRC GENERIC LETTER 83-36
Title: NUREG-0737 TECHNICAL SPECIFICATIONS

Summary:

Licensees are required to perform a feasibility and risk assessment study to determine the optimum approach for modifying automatic depressurization system (ADS) actuation logic to eliminate the need for manual actuation to assure adequate core cooling. The BWR Owners' group has submitted an evaluation to the staff. The staff has identified the acceptable options for modifications of ADS logic. Each licensee was requested to select appropriate modifications approved by the staff. Technical Specifications changes resulting from the modifications will be reviewed on a case by case basis.

Potential Impact Number: 25621 **SRP Section Number:** 3.5.1.6

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

In the SSAR Section 3.5.1.6, GE originally stated that "aircraft hazards are not a design-basis event . . .," since the resultant event probability is < 10-7. In the DFSER,

the staff identified it as a site-specific action item for the COL applicant to address. This was DFSER COL Action Item 3.5.1.6-1. Subsequently, in SSAR Section 3.5.4.3, Amendment 33, GE stated that the COL applicant shall provide an analysis that demonstrates that the probability of aircraft impacting the ABWR standard plant and causing consequences greater than those permitted in 10 CFR Part 100 guidelines is < 10-7. This is acceptable.

Potential Impact Number: 25622 **SRP Section Number:** 3.5.1.6

Document: FINAL SER ABWR CH 2
Title: SITE CHARACTERISTICS

Summary:

The staff accepted the following COL actions for the ABWR design:

The COL applicant should provide site-specific information on identifying potential hazards in the site vicinity and evaluating potential accidents, should provide information on industrial, military, and transportation facilities and routes to establish the presence and magnitude of potential external hazards, and should identify potential accident situations in the vicinity of the plant and give the reasons why these potential accidents were or were not accommodated in the design.

Potential Impact Number: 25624 **SRP Section Number:** 3.9.4

Document: NRC GENERIC LETTER 80-95
Title: FINAL EDITION OF NUREG-0619, 'BWR FEEDWATER NOZZLE AND CONTROL ROD DRIVE RETURN LINE NOZZLE CRACKING'

Summary:

The letter 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 the NRC's Generic Technical Activity A-10, an "Unresolved Safety Issue" pursuant to Section 210 of the Energy Reorganization Act of 1974. Public comments received in response to the "For Comment" edition of April 1980, have been incorporated where applicable.

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NUREG-0619 (microfiche No. 07219-166) describes the technical issues, the technical studies and 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.

This potential impact was identified during the SRP-UDP integration task.

Potential Impact Number: 25630 **SRP Section Number:** 3.9.6

Document: NRC GENERIC LETTER 89-04 SUP 1
Title: GUIDANCE ON DEVELOPING ACCEPTABLE INSERVICE TESTING PROGRAMS

Summary:

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to notify addressees that it 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 and includes the positions from Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," supplemented with current considerations for using these positions.

NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," discusses OM-6 and OM-10, which may be implemented by licensees pursuant to 10 CFR 50.55a (f)(4)(iv). NUREG-1482, through the staff's endorsement in the generic letter supplement, gives the requisite approval for 10 CFR 50.55a (f)(4)(iv) for updating an IST program to the requirements of OM-6 and OM-10 (and OM-1 through reference in OM-10) provided the licensee documents the use of OM-6 and OM-10 in the IST program. The NUREG, through the generic letter supplement per (f)(4)(iv), also gives approval to implement selected portions of OM-6 and OM-10 as discussed in NUREG-1482.

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Potential Impact Number: 25634 **SRP Section Number:** 5.4.2.2

Document: NRC GENERIC LETTER 95-03
Title: CIRCUMFERENTIAL CRACKING OF STEAM GENERATOR TUBES

Summary:

The NRC issued this Generic Letter to notify addressees about the recent steam generator tube inspection findings at Maine Yankee Atomic Power Station and the safety significance of these findings. In addition, the NRC requested that all addressees implement the actions described in this Generic Letter.

Inspection practices should furnish assurance that steam generator tube degradation will be reliably detected so that the potential for the rupturing of a tube is maintained at an acceptably low level. If licensees conclude that unexpected levels of tube degradation may exist in their steam generators, they should implement compensatory measures to minimize the chance that tube integrity is compromised and to ensure that the plant can safely respond to a tube failure. Such measures should have the objective of maintaining a safe operating posture through a defense-in-depth philosophy of (1) prevention of uncontrolled tube degradation, (2) early detection of tube degradation, and (3) mitigation of the consequences of failed tubes.

To verify compliance with regulatory requirements (10 CFR Part 50 Appendix B; technical specifications) and to maintain an appropriate degree of defense-in-depth measures, the NRC has concluded that it is appropriate for PWR licensees to take the measures enumerated in this Generic Letter.

Potential Impact Number: 25637 **SRP Section Number:** 8.1

Document: RG 1.118 **Revision:** 3 , 4/1/95
Title: PERIODIC TESTING OF ELECTRIC POWER AND PROTECTION SYSTEMS

Summary:

This regulatory guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with respect to the periodic testing of the electric power and protection systems.

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Regulatory Guide 1.118 revision 3 was issued in April of 1995. The differences between revision 3 and revision 2 should be evaluated for impacts on this SRP Section.

Potential Impact Number: 25640 **SRP Section Number:** 17.3

Document: NRC GENERIC LETTER 85-06
Title: QUALITY ASSURANCE GUIDANCE FOR ATWS EQUIPMENT THAT IS NOT SAFETY-RELATED

Summary:

On June 1, 1984, the Commission approved publication of a final rule, 10 CFR 50.62, regarding the reduction of risk from anticipated transients without scram (ATWS) events for light-water cooled nuclear power plants (49 FR 26036). At the same time, the Commission directed the staff to complete and issue in the form of a generic letter explicit quality assurance (QA) guidance for non-safety-related equipment encompassed by the ATWS rule. Section 50.62(d) of the ATWS rule requires that each licensee develop and submit (to the Director of the Office of Nuclear Reactor Regulation) a proposed schedule for meeting the requirements of the rule within 180 days after issuance of the QA guidance. This letter 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.

The QA controls in Appendix B to 10 CFR 50 describe one form of a comprehensive management control system for a complex task. While Appendix B describes only one such system, licensees and applicants have expressed a desire to minimize proliferation of different kinds of management control systems for their plants. The NRC staff concurs with this desire not to establish new and separate management control systems for non-safety-related ATWS equipment.

Potential Impact Number: 25641 **SRP Section Number:** 5.2.2

Document: CODE OF FED. REGS 10CFR50

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Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.61

Summary:

For each pressurized water nuclear power reactor for which the analysis indicates that no reasonably practicable flux reduction program will prevent the value of RT(PTS) from exceeding the PTS screening criterion before the expiration date of the operating license, or the projected expiration date if a change in the operating license has been requested, or the end of a renewal term if a request for license renewal has been submitted, the licensee shall submit a safety analysis to determine what, if any, modifications to equipment systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed.

Potential Impact Number: 25650 **SRP Section Number:** 3.11

Document: RG 1.151 **Revision:** 0 , 7/1/83
Title: INSTRUMENT SENSING LINES

Summary:

The requirements of ISA-S67.02, "Nuclear Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants," 1980, provide a basis which is generally acceptable to the NRC staff for the design and installation of safety-related instrument sensing lines in nuclear power plants.

However, regulatory position C.4 of Regulatory Guide 1.151 recommends that freezing temperatures be added to the environmental and installation conditions listed in Section 5.2.1(5) of ISA-S67.02 that instrument sensing lines should be able to withstand and continue to perform their intended function. Additionally, regulatory position C.5 makes the statement that the special considerations in Section 5.2.2 of ISA-S67.02 that should be addressed in the design and installation of instrument sensing lines be supplemented with the following:

a. Instrument sensing lines that can be exposed to freezing temperatures and that contain and/or can be expected to contain a condensable mixture or fluid that can freeze should be provided with an environmental control system (such as heating and

ventilation or heat tracing) to protect the lines from freezing during extremely cold weather.

b. The environment associated with those instrument sensing lines in paragraph a. that are safety related should be monitored and alarmed so that appropriate corrective action can be taken to prevent loss of, or damage to, the lines from freezing in the event of loss of the environmental control system.

c. The environmental control system recommended in paragraph a., and for which paragraph b. applies, should be electrically independent of the monitoring and alarm system so that a single failure in either system, including their power sources, does not affect the capability of the other system.

d. The environmental control and monitoring systems of paragraph a. and paragraph b. should be designed to standards commensurate with their importance to safety and with administrative controls that are implemented to address events or conditions that could render either one or both of the systems inoperable.

Potential Impact Number: 25653 **SRP Section Number:** 3.11

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

In the SSAR, GE stated that the design of the information systems important to safety will be in conformance with the guidelines of Revision 3 of RG 1.97. However, the footnote for Subsection 50.49(b)(3) references Revision 2 for selection of the types of post-accident monitoring equipment. In issuing Revision 3, the NRC staff stated that the conformance with Revision 3 would not alter the implementation of Subsection 50.49. Therefore, conformance with Revision 2 is not required because conformance with Revision 3 meets the underlying purpose of the rule. As a result, an exemption from Subsection 50.49(b)(3) is justified by the special circumstances set forth in Subsection 50.12(a)(2)(ii).

Potential Impact Number: 25654 **SRP Section Number:** 3.11

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Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

The CESSAR-DC states that the design of the information systems important to safety conform to the guidelines of Revision 3 of RG 1.97. However, the footnote for 10 CFR 50.49(b)(3) references Revision 2 for selection of the types of postaccident monitoring equipment. In issuing Revision 3, the NRC staff stated that conformance to Revision 3 would not alter the implementation for 10 CFR 50.49. In accordance with 10 CFR 50.12(2)(ii), the Commission will consider granting an exemption if application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. Therefore, conformance to Revision 2 is not required because conformance to Revision 3 meets the underlying purpose of the rule. Therefore, an exemption from 10 CFR 50.49(b)(3) is justified by the special circumstances set forth in 10 CFR 50.12 (a)(2)(ii).

Potential Impact Number: 25655 **SRP Section Number:** 17.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.62

Summary:

The Anticipated Transient Without Scram (ATWS) rule defines ATWS and requires that light-water-cooled plants have certain features which were determined to acceptably reduce the risk from ATWS events. The design features required are:

(1) PWRs-turbine trip and automatic auxiliary feedwater initiation for ATWS conditions [paragraph (c)(1) of the rule];

(2) CE and B&W PWRs-diverse scram system [paragraph (c)(2)];

(3) BWRs-alternate rod insertion system [paragraph (c)(3)], recirculation pump trips for ATWS conditions [paragraph (c)(5)], and automatically initiated (for specified plants) standby liquid control system (SLCS) of specified minimum reactivity control capacity [paragraph (c)(4)].

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The rule requires that certain of the above features be reliable, diverse, and/or independent of the reactor trip system.

Furthermore, the rule requires [in paragraph (c)(6)] that information sufficient to demonstrate to the Commission the adequacy of items required by the rule be submitted to the Commission as specified in 10 CFR 50.4. The rule also indicates that QA guidance was to be issued for nonsafety-related components.

Potential Impact Number: 25656 **SRP Section Number:** 6.2.6

Document: FINAL SER CE80 CH 6
Title: ENGINEERED SAFETY FEATURES

Summary:

ABB-CE committed to containment leakage testing for the System 80+ plant in accordance with Appendix J to 10 CFR Part 50 with the following exception:

Leaks occurring during the Type A test that could affect the test results will not prevent completion of this test if: (a) the leaks are isolated for the balance of the test; (b) the leaking component had a "pre-maintenance" local leak rate test whose results, when added to those from the Type A test, are in conformance with the acceptance criteria of Appendix J; or (c) a "postmaintenance" local leak rate test of the leaking component(s) is performed and the results, when added to those from the Type A test, conform to the acceptance criteria of Appendix J.

The exception does not conform to the requirements of Appendix J to 10 CFR Part 50. Section III.A.1.(a) of Appendix J requires that a Type A test, defined as a test to measure the primary containment overall integrated leakage rate be terminated if, during this test, potentially excessive leakage paths are identified which would either interfere with satisfactory completion of the test or which would result in the Type A tests not meeting the applicable acceptance criteria of Section III.A.4(b) or III.A.5(b). Section III.A.1(a) further requires that, after terminating a Type A test due to potentially excessive leakage, the leakage through the potentially excessive leakage paths be measured using local leakage testing methods and repairs and/or adjustments to the affected equipment be made. The Type A test shall then be conducted. ABB-CE proposes that, when leakage is found during a Type A test, the test not be terminated. Instead, ABB-CE proposes that leaks be isolated and the Type A test continued. After completion of the modified Type A test (i.e., a Type A test with the leakage paths

isolated), local leakage rates of those paths isolated during the modified Type A test will be measured before or after the maintenance to those paths.

ABB-CE proposes that the adjusted "as-found" leakage rate for the Type A test be determined by adding the local leakage rates measured before maintenance to those previously isolated leakage paths, to the containment integrated leakage rate determined in the modified Type A test. This adjusted "as-found" leakage rate is to be used in determining the scheduling of the periodic Type A tests in accordance with Section III.A.6 of Appendix J.

Finally, ABB-CE proposes that the acceptability of the modified Type A test be determined by calculating the adjusted "as-left" containment overall integrated leakage rate and comparing this to the acceptance criteria of Appendix J. The adjusted "as-left" Type A leakage rate is determined by adding the local leakage rates measured after any maintenance to those previously isolated leakage paths, to the leakage rate determined in the modified Type A test.

The differences between the proposed leak testing and the requirements in Section III.A.1(a) of Appendix J are that: (1) the potentially excessive leakage paths will be repaired and/or adjusted after completion of the Type A test rather than before the test; and (2) the Type A test leakage rate is partially determined by calculation rather than by direct measurement. With respect to the first issue, the staff does not identify any significant difference in the end result (i.e., the "as-left" local leakage rates will be maintained within an acceptable range). With respect to the second issue, the measured "as-left" local leakage rates will represent a relatively small correction to the containment overall integrated leakage rate measured in the modified Type A test. Accordingly, there will be insignificant difference between the calculated "as-left" containment leakage rate (i.e., a modified Type A test) and one which would be directly measured in compliance with the requirements of Section III.A.1.(a).

The staff concludes that the special circumstances described in 10 CFR 50.12(a)(2)(ii) exist in that the regulation need not be applied in this particular circumstance to achieve the underlying purpose because ABB-CE has proposed acceptable alternatives that accomplish the intent of the regulation. On this basis, the staff concludes that a partial exemption from the requirements of Section III.A.1.(a) of Appendix J is justified.

Potential Impact Number: 25657 **SRP Section Number:** 6.2.6

Document: SECY 93-087

APPENDIX II
Potential Impacts

Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

Appendix J to 10 CFR Part 50 requires that Type C tests be performed during each reactor shutdown for refueling, but in no case at intervals greater than 2 years.

In SECY 93-087, Issue II.H, the staff recommended that the interval for Type C leakage rate tests be changed to 30 months from 24 months. The Commission endorsed this recommendation in the SRM dated July 21, 1993.

Potential Impact Number: 25658 **SRP Section Number:** 6.2.6

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

Appendix J to 10 CFR Part 50 requires that Type C tests be performed during each reactor shutdown for refueling, but in no case at intervals greater than 2 years. In response to a proposal by EPRI in the Evolutionary Requirements Document that the maximum interval between Type C leakage rate tests be 30 months rather than the 24 months currently required by Appendix J, the NRC staff concluded that the leakage rate test interval would be evaluated in the context of a standard design certification review.

Potential Impact Number: 25660 **SRP Section Number:** 9.3.5

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

Future evolutionary ALWR designs can reduce the possibility of a loss-of-coolant accident (LOCA) outside containment by designing (to the extent practicable) all

systems and subsystems connected to the reactor coolant system (RCS) to an ultimate rupture strength at least equal to the full RCS pressure.

Potential Impact Number: 25661 **SRP Section Number:** 17.1

Document: C&S: ANSI N45.2.9
Title: Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants

Summary:

SRP Section 17.1 cites ANSI N45.2.9 (date not specified). The standard is cited in conjunction with Regulatory Guide 1.88, which cites the 1974 version of the standard. The latest version of the standard is ASME NQA-1 as endorsed by Regulatory Guide 1.28.

Potential Impact Number: 25662 **SRP Section Number:** 8.1

Document: C&S: IEEE 387
Title: IEEE STANDARD CRITERIA FOR DIESEL-GENERATOR UNITS APPLIED AS STANDBY POWER SUPPLIES FOR NUCLEAR POWER GENERATING STATIONS

Summary:

SRP Section 8.1 cites IEEE Std. 387 in Table 8-1 in association with Regulatory Guide 1.9. The citation of the standard is non-date specific. Regulatory Guide 1.9 endorses the 1984 version of the standard.

Potential Impact Number: 25663 **SRP Section Number:** 9.5.1

Document: NRC GENERIC LETTER 82-21
Title: TECHNICAL SPECIFICATIONS FOR FIRE PROTECTION AUDITS

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Potential Impacts

Summary:

Generic Letter 82-21 provides guidance for the types of audits to be performed of the Fire Protection Program. This GL was listed as a reference in GL 86-10. An earlier impact was discarded as not containing useful information relative to SRP 9.5.1. Reconsider incorporating GL 82-21 as guidance or reference in BTP CMEB 9.5-1 for QA Program Guidance.

Potential Impact Number: 25664 **SRP Section Number:** 6.5.5

Document: NRC MEMORANDUM 09/21/95, from Borchart, R to Hermann R
Title: REVIEW OF REVISED SRP SECTIONS 5.2.3, 5.4.1.1, 5.4.8, 6.1.2, 6.2.7, 6.5.2, 6.5.4, 6.5.5

Summary:

The attachment to the subject memorandum identifies an error in SRP Section 6.5.5. Specifically,utilizing Regulatory Guide 1.3 assumptions in conjunction with the procedures in the SRP may result in double accounting of the plateout removal mechanism when determining the fission product iodine available for leakage from containment.

Potential Impact Number: 25666 **SRP Section Number:** 6.5.2

Document: NRC MEMORANDUM 09/21/95, from Borchart, R to Hermann R
Title: REVIEW OF REVISED SRP SECTIONS 5.2.3, 5.4.1.1, 5.4.8, 6.1.2, 6.2.7, 6.5.2, 6.5.4, 6.5.5

Summary:

The attachment to the subject memorandum identifies an error in SRP Section 6.5.2. Specifically,utilizing Regulatory Guide 1.3 or 1.4 assumptions in conjunction with the procedures in the SRP may result in double accounting of the plateout removal mechanism when determining the fission product iodine available for leakage from containment.

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Potential Impact Number: 25667 **SRP Section Number:** 3.7.4

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 recommendation.

In SECY 93-087 the staff proposed additional guidance and positions for the design of structures, systems, and components (SSCs) 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.

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 Impact Number: 25668 **SRP Section Number:** 3.7.4

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

Design certification, using a single-earthquake SSE design, is predicated on the adequacy of pre-earthquake planning and post-earthquake damage inspections that are to be implemented by the COL applicant. The COL applicant will be required to demonstrate to the NRC staff as a part of its application the procedures it plans to use

for pre-earthquake planning and post-earthquake actions. For design certification, the NRC staff reviewed the criteria developed by EPRI in reports 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 for ensuring that these actions can be taken. The staff concluded in its review of EPRI NP-5930 that the report was adequate for use by the COL applicant with several enumerated exceptions: [5 specific items were listed, all of which constitute requirements that the seismic instrumentation should meet.]

Potential Impact Number: 25669 **SRP Section Number:** 3.7.4

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

Design certification, using a single-earthquake SSE design, is predicated on the adequacy of pre-earthquake planning and post-earthquake damage inspections that are to be implemented by the COL applicant. The COL applicant will be required to demonstrate to the NRC staff as a part of its application the procedures it plans to use for pre-earthquake planning and post-earthquake actions. For design certification the NRC staff reviewed the criteria developed by EPRI in reports 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 for ensuring that these actions can be taken. The staff concluded from its review of EPRI-5930 that the report was adequate for use by a COL applicant with an enumeration of exceptions (5 specific items are listed, all of which constitute requirements that the seismic instrumentation should meet).

For the System 80+, the following criteria for structures are used to ensure that when the OBE is eliminated from design, the structures will continue to be designed appropriately for earthquake effects: ... (2) CESSAR-DC Sections 3.7.4 states that seismic instrumentation will be installed so that prompt action can be taken when the response spectra level and the cumulative absolute velocity (CAV) experienced at the site exceed the shutdown level. It also gives information on pre-earthquake planning and post-earthquake operator action. The acceptability review of seismic instrumentation is discussed in Section 3.7.4 of this report.

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Potential Impact Number: 25670 **SRP Section Number:** 3.7.4

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Locations: Appendix A, Appendix A, 59 FR 52255

Summary:

By proposed rulemaking published in 59 FR 52255, the Commission stated their intent to revise their regulations to remove from 10 CFR Part 100, Appendix A, design criteria not associated with site suitability or establishment of the SSE ground motion and place them into 10 CFR Part 50. This includes requirements for seismic instrumentation. Criteria on earthquake engineering would be included in a new Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50.

The proposed regulation would allow the value of the OBE to be set at (I) one-third or less of the SSE, where OBE requirements are satisfied without an explicit response or design analyses being performed, or (ii) a value greater than on-third of the SSE, where analysis and design are required.

The proposed regulation would treat plant shutdown associated with vibratory ground motion exceeding the OBE, or significant plant damage, as a condition in every operating license. Draft Regulatory Guide DG-1034, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Actions," is being developed to provide guidance acceptable to the NRC staff for determining whether or not vibratory ground motion exceeding the OBE or significant plant damage had occurred and the timing of nuclear power plant shutdown. The guidance is based on criteria developed by the Electric Power Research Institute (EPRI).

The NRC is currently developing the following draft regulatory guides to implement the proposed regulation:

1) DG-1033, Third Proposed Revision 2 to Regulatory Guide 1.12, "Nuclear Power Plant Instrumentation for Earthquakes," describing seismic instrumentation type and location, operability, characteristics, installation, actuation, and maintenance that are acceptable to the NRC staff.

2) DG-1034, as discussed above.

3) DG-1035, "Restart of a Nuclear Power Plant Shut Down by as Seismic Event," establishing guidelines that are acceptable to the NRC staff for performing inspections

and tests of nuclear power plant equipment and structures prior to restart of a plant that has been shutdown because of a seismic event.

Potential Impact Number: 25675 **SRP Section Number:** 8.1

Document: RG 1.63 **Revision:** 3
Title: ELECTRIC PENETRATION ASSEMBLIES IN CONTAINMENT STRUCTURES FOR NUCLEAR POWER PLANTS

Summary:

Provides design, construction, testing, qualification, installation, and external circuit protection guidance for nuclear power plant containment electrical penetration assemblies and modifies the provisions of IEEE Standard 317-1983. The Reg. guide is specified as guidance for SRP sections 8.3.1 and 8.3.2.

Potential Impact Number: 25676 **SRP Section Number:** 8.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.63

Summary:

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. RG 1.155 references NUMARC-8700, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors. The latter document also provides acceptable guidance for meeting the requirements of the Station Blackout Rule. Guidance has not been provided for the reviewer of the sections in SRP Chapter 8 that integrates the guidance found in these documents.

Potential Impact Number: 25677 **SRP Section Number:** 8.2

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Document: RG 1.155 **Revision:** 0
Title: STATION BLACKOUT

Summary:

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. RG 1.155 references NUMARC-8700, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors. The latter document also provides acceptable guidance for meeting the requirements of the Station Blackout Rule. Guidance has not been provided for the reviewer of the sections in SRP Chapter 8 that integrates the guidance found in these documents.

Potential Impact Number: 25678 **SRP Section Number:** 8.2

Document: NUMARC 8700
Title: GUIDELINES AND TECHNICAL BASES FOR NUMARC INITIATIVES ADDRESSING STATION BLACKOUT AT LIGHT WATER REACTORS

Summary:

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. RG 1.155 references NUMARC-8700, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors. The latter document also provides acceptable guidance for meeting the requirements of the Station Blackout Rule. Guidance has not been provided for the reviewer of the sections in SRP Chapter 8 that integrates the guidance found in these documents.

Potential Impact Number: 25679 **SRP Section Number:** 8.2

Document: C&S: ANSI C37.04
Title: Standard Rating Structure for AC High-Voltage Circuit Breakers Rated on a Symmetrical Basis

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Summary:

ANSI C37.04 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; and many others is provided. The current revision of the standard is 1979. ANSI C37.04 has not been endorsed by a regulatory guide.

Potential Impact Number: 25681 **SRP Section Number:** 3.8.1

Document: FINAL SER CE80 CH 1
Title: INTRODUCTION AND GENERAL DISCUSSION

Summary:

In accordance with 10 CFR 52.48, the staff used the current regulations in 10 CFR Parts 20, 50, 73, and 100 in reviewing ABB-CE's application for design certification. During this review, the staff recognized that if certain regulations were applied to the System 80+ design, the use of those regulations would not serve the underlying purpose of the rule or would not be needed for achieving the fundamental intent of the rule. These exemptions are discussed in the sections of this report listed below.

In the SRM pertaining to SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor Designs," the Commission approved the staff's recommendation to proceed with design-specific rulemakings as part of individual design certifications to resolve selected technical and severe accident issues as "applicable regulations" for the advanced boiling water reactor and System 80+ standard designs. Among these issues are staff positions that deviate from or are not embodied in current regulations applicable to the System 80+ design.

These issues were addressed in various Commission papers, such as SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced LWR designs." The "applicable regulations" that are specific to the System 80+ design are identified and evaluated in this report. The proposed design certification rule will then include these "applicable regulations" for the System 80+ design for the purposes of 10 CFR 52.48, 52.54, 52.59, and 52.63. These "applicable regulations" are discussed in the sections of this report, including Section 3.1.1, "Exemption from operating-basis earthquake design requirement."

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Potential Impact Number: 25682 **SRP Section Number:** 3.8.1

Document: FINAL SER ABWR CH 1
Title: INTRODUCTION AND GENERAL DISCUSSION

Summary:

In the SRM pertaining to SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor Designs," the Commission approved the staff's recommendation to proceed with design-specific rulemakings as part of design certification rulemakings to establish selected technical and severe accident issues as "applicable regulations" for the ABWR and System 80+ standard designs. These issues included staff positions that deviated from or were not embodied in current regulations applicable to the standard designs.

These issues were addressed in various Commission papers, such as SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolution and Advanced LWR Designs." The "applicable regulations" that are specific to the ABWR design are identified and evaluated in this report, including Section 3.1.1, Exemption from operating basis earthquake design requirement.

Potential Impact Number: 25683 **SRP Section Number:** 3.8.1

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff's review of the controlling load combinations finds that, in general, the load combinations with the SSE control the design of steel structures although there may be specific cases where the load combinations with the OBE control the design. Similarly, an examination of the pertinent load combinations for concrete structures, including the containment structure, should lead to the same conclusion that the OBE loads, in most cases, do not control the outcome of the structural design.

In the design of the containment, the staff reviewed the extent to which the elimination of the OBE from the load combinations would lead to a reduction of the safety margin.

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An examination of the nuclear structural design practice and the SRP load combination equations, however, show that the major dynamic load for the overall design of structures is either the OBE or the SSE. All other potential dynamic loads are conservatively accounted for in the definition of equivalent dead and live loads or only produce local effects that are handled by local reinforcement details. Therefore, the staff concludes that no reduction in safety margins of concrete and steel structures results from the elimination of the OBE as a design requirement.

Potential Impact Number: 25684 **SRP Section Number:** 3.8.1

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

The staff's review of the controlling load combinations finds that, in general, the load combinations with the SSE control the design of steel structures although there may be specific cases where the load combinations with the OBE control. Similarly, an examination of the pertinent load combinations for concrete structures should lead to the same conclusion that the OBE loads, in most cases, do not control the outcome of the structural design.

All other potential dynamic loads are conservatively accounted for in the definition of equivalent dead and live loads, except those that produce local effects that are handled by local reinforcement details. Therefore, it can be concluded that no reduction in safety margins of concrete and steel structures results from the elimination of the OBE as a design requirement.

Potential Impact Number: 25685 **SRP Section Number:** 13.1.1

Document: NUREG 0711
Title: HUMAN FACTORS ENGINEERING PROGRAM REVIEW MODEL

Summary:

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NUREG-0711, 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.

Potential Impact Number: 25686 **SRP Section Number:** 13.1.2 - 13.1.3

Document: NUREG 0711
Title: HUMAN FACTORS ENGINEERING PROGRAM REVIEW MODEL

Summary:

NUREG-0711, published in July 1994, presents a model for performing design certification reviews of human factors engineering program elements. Element 5 is "Staffing" and Element 6 is "Human Reliability Analysis." The NUREG contains important factors for NRC staff reviewers to consider.

Potential Impact Number: 25687 **SRP Section Number:** 13.2.1

Document: NUREG 0711
Title: HUMAN FACTORS ENGINEERING PROGRAM REVIEW MODEL

Summary:

NUREG-0711, 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.

Potential Impact Number: 25688 **SRP Section Number:** 13.2.2

Document: NUREG 0711
Title: HUMAN FACTORS ENGINEERING PROGRAM REVIEW MODEL

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Potential Impacts

Summary:

NUREG-0711, 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.

Potential Impact Number: 25689 **SRP Section Number:** 13.5.2.1

Document: NUREG 0711
Title: HUMAN FACTORS ENGINEERING PROGRAM REVIEW MODEL

Summary:

NUREG-0711, 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.

Potential Impact Number: 25690 **SRP Section Number:** 15.1.1 - 15.1.4

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

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.

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.

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Potential Impact Number: 25691 **SRP Section Number:** 15.4.9

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

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. 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 Impact Number: 25692 **SRP Section Number:** 3.8.4

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

The staff's review of the controlling load combinations finds that, in general, the load combinations with the SSE control the design of steel structures although there may be specific cases where the load combinations with the OBE control. Similarly, an examination of the pertinent load combinations for concrete structures should lead to the same conclusion that the OBE loads, in most cases, do not control the outcome of the structural design.

All other potential dynamic loads are conservatively accounted for in the definition of equivalent dead and live loads, except those that produce local effects that are handled

by local reinforcement details. Therefore, it can be concluded that no reduction in safety margins of concrete and steel structures results from the elimination of the OBE as a design requirement.

Potential Impact Number: 25693 **SRP Section Number:** 3.8.4

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff's review of the controlling load combinations finds that, in general, the load combinations with the SSE control the design of steel structures although there may be specific cases where the load combinations with the OBE control the design. Similarly, an examination of the pertinent load combinations for concrete structures, including the containment structure, should lead to the same conclusion that the OBE loads, in most cases, do not control the outcome of the structural design.

In the design of the containment, the staff reviewed the extent to which the elimination of the OBE from the load combinations would lead to a reduction of the safety margin. An examination of the nuclear structural design practice and the SRP load combination equations, however, show that the major dynamic load for the overall design of structures is either the OBE or the SSE. All other potential dynamic loads are conservatively accounted for in the definition of equivalent dead and live loads or only produce local effects that are handled by local reinforcement details. Therefore, the staff concludes that no reduction in safety margins of concrete and steel structures results from the elimination of the OBE as a design requirement.

Potential Impact Number: 25695 **SRP Section Number:** 9.5.2

Document: NUREG 0700
Title: GUIDELINES FOR CONTROL ROOM DESIGN REVIEWS

Summary:

These guidelines for control room design reviews provide the NRC staff guidance that should be followed in accomplishing the detailed control room design review described

in Task I.D.1 of NUREG-0660, the NRC Action Plan. Guidance includes recommendations for planning the review process, review procedures, the assessment of identified human engineering discrepancies to select those that should be corrected, implementation of design improvements, and reporting. Detailed human engineering guidelines are included which provide benchmarks for the human factors quality of specific control room design features.

Section 6.2 provides criteria for communications systems and equipment and related design features of the control room to support reliable human performance.

Microfiche No. 09940:204

Potential Impact Number: 25697 **SRP Section Number:** 15.6.5.D

Document: NRC GENERIC LETTER 86-17
Title: AVAILABILITY OF NUREG-1169. 'TECHNICAL FINDINGS RELATED TO GENERIC ISSUE C-8; BOILING WATER REACTOR MAIN STEAM ISOLATION VALVE LEAKAGE AND LEAKAGE TREATMENT METHODS'

Summary:

This potential impact was duplicated from PI# 15082.

This letter is to inform the licensees that NUREG 1169 was published in June, 1986. This NUREG presents the results of the NRC staff and contractor efforts to assess Generic Issue C-8 'Main Steam Isolation Valves Leakage and Leakage Control System Failures.' NUREG-1169 represents technical findings associated with Generic Issue C-8 and the staff expects it to be used in the regulatory resolution of the issue. Neither this letter nor NUREG-1169 constitute a change in the existing generic staff positions or allowable MSIV leakage rates, nor the specifications for MSIV leakage control systems. Therefore, no action is being requested of applicants or licensees.

This potential impact was identified on the basis that NUREG-1169 provides substantial background on MSIVLCS issues including evaluations related to the elimination of MSIVLCS.

Potential Impact Number: 25702 **SRP Section Number:** 15.6.5.D

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Potential Impacts

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

This potential impact was duplicated from PI# 22102.

"2.3.1 BWR Main Steamline Isolation Valves and Leakage Control and Classification of Main Steamline of Boiling-Water Reactor

In Section 2.3.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI proposes to eliminate the BWR main steam isolation valve leakage control system and provide an alternative leakage pathway (i.e., the main steamline and the condenser) to the main condenser downstream of the isolation valves in the event of a loss-of-coolant accident (LOCA). This issue is also related to Section 3.3.2 in Appendix B to Chapter 1, Section 3.4.1.5 in Chapter 2, Sections 5.3.3 and 5.4.1.5 in Chapter 3, Section 1.2.3 in Chapter 5, and Section 3 in Chapter 13 of the Evolutionary Requirements Document. In its letters dated May 17 and August 29, 1991, the staff stated its positions on the quality group and seismic classifications of the BWR main/extraction steam system, main steamline, main turbine system, condenser, and related non-seismic structures that resulted from the proposed elimination of the main steam isolation valve leakage control system. The objective of these staff positions is to establish adequate measures to ensure that the above systems and structures will maintain their structural integrity during and following an SSE. Although this issue involves information in Chapters 1, 2, 3, 5, 6, and 13, for the sake of continuity, the basis for the staff's positions and the staff's evaluation of all of EPRI's responses are discussed [in this section]."

Potential Impact Number: 25703 **SRP Section Number:** 15.6.5.D

Document: DRAFT FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

This potential impact was duplicated from PI# 22103.

GE offers an alternative plan for plateout and holdup of radionuclides following a LOCA and intends not to include an MSIV leakage control system in ABWR designs. Staff

presents their review of this alternative in ABWR SER section 15.4.4.2. 'Staff concerns related to the GE proposal were identified as Open Item 142 in the DSER. Subsequently the staff has completed its evaluation and has accepted the BWROG and GE proposals. In order to implement these proposals, the staff has had to deviate from the current guidelines provided in the SRP. The bases for these deviations by the staff are given in [15.4.4.2.1 and 15.4.4.2.2].'

Potential Impact Number: 25704 **SRP Section Number:** 15.6.5.D

Document: DRAFT FINAL SER ABWR CH 10
Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

This potential impact was duplicated from PI# 22104.

GE offers an alternative plan for plateout and holdup of radionuclides following a LOCA and intends not to include an MSIV leakage control system in ABWR designs. Staff has identified this issue as COL Action Item 10.3.1-1 and commits to review this proposed method to maintain offsite exposure within 10CFR100 limits on a case by case basis.

Potential Impact Number: 25705 **SRP Section Number:** 15.6.5.D

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

This potential impact was duplicated from PI# 22471.

'The staff concludes that the above described approach {performing dynamic seismic analysis of the condenser, a portion of the main steam bypass line, and the piping between the turbine stop valve and the turbine inlet} to resolve the BWR main steam line issue for both evolutionary and passive ALWRs provides reasonable assurance that the main steam piping from the outermost isolation valve up to the turbine stop

valve, the main steam drain and by-pass lines up to the condenser, and the main condenser will retain their pressure and structural integrity during and following a SSE. The staff recommends that the Commission approve the above-described approach to resolve the main steamline classification for both evolutionary and passive ALWRs.'

Potential Impact Number: 25706 **SRP Section Number:** 15.6.5.D

Document: DRAFT FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

This potential impact was duplicated from PI# 22473.

"GE has eliminated the main steam isolation valve leakage control system for the ABWR plant design. Instead, it proposes to rely on the use of an alternate leakage path that takes advantage of the large volume and surface area in the main steam piping, bypass line, and condenser to hold up and plate out the release of fission products following core damage. In this manner, the main steam piping, bypass line, and condenser are used to mitigate the consequences of an accident and are required to remain functional during and after an SSE. ...

The resolution of this item will be discussed in Section 3.7.2 of this report. This is Open Item 3.2.1-1."

Potential Impact Number: 25707 **SRP Section Number:** 15.6.5.D

Document: DRAFT FINAL SER ABWR CH 14
Title: VERIFICATION PROGRAMS

Summary:

This potential impact was duplicated from PI# 22476.

"For the ABWR plant design, GE proposes to eliminate the main steam isolation valve leakage control system and to rely on the use of an alternate leakage path that takes advantage of the large volume and surface area in the main steam piping, bypass line,

and condenser to hold up and plate out the release of fission products following core damage. In this manner, the main steam piping, bypass line, and condenser will be used to mitigate the consequences of an accident and will be required to remain functional during and after a safe shutdown earthquake (SSE)."

Potential Impact Number: 25708 **SRP Section Number:** 15.2.8

Document: DRAFT SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

This potential impact was duplicated from PI# 22920.

In accordance with the requirements of GDC 17, loss of offsite power (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. For some transient analyses (such as the loss of condenser vacuum event analysis), the applicant did not postulate the unavailability of offsite power as part of the event, but rather took it as an additional single failure to show limiting fault criteria were not exceeded. The applicant is required to discuss each of the transient and accident analyses in the CESSAR to justify that the analyses meet the GDC 17 requirements given above. If the existing analyses do not meet the GDC 17 requirements, the applicant should reanalyze the transient and accident analyses in accordance with GDC 17 and submit the results for the staff to review.

Potential Impact Number: 25709 **SRP Section Number:** 15.2.8

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 17

Summary:

This potential impact was duplicated from PI# 22921.

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"An onsite electric power system 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 anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained."

Potential Impact Number: 25710 **SRP Section Number:** 15.2.8

Document: DRAFT SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

This potential impact was duplicated from PI# 22922.

In CESSAR Section 15.2, the applicant analyzes various transients resulting from a decrease in heat removal by the secondary system and identifies the limiting cases for the consideration of integrity of RCS primary system boundary and fuel rod cladding to withstand the consequences of transients. With consideration of a single failure, the applicant noted that the loss of condenser vacuum (LOCV) event with a loss of offsite power (LOOP) following a turbine trip is the most limiting case with respect to both peak RCS pressure and low DNBR. With consideration of the worst single failure of a LOOP, the limiting fuel cladding damage resulting from the LOCV event is less than 1.8 percent of total fuel rods. As required by GDC 17 (discussed in Section 15.1), a LOOP should be included in the analysis and the LOCV event with LOOP should be analyzed as a moderate-frequency event, and therefore, the calculated DNBR should meet the 95/95 DNBR limit as discussed in Section 4.4 of this DSER. Since the fuel failure was predicted, the staff concludes that the results of the LOCV are not acceptable. The

applicant should provide acceptable results using the approved methods for the LOCV event analysis for the staff to review.

Potential Impact Number: 25711 **SRP Section Number:** 15.1.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 17

Summary:

This potential impact was duplicated from PI# 23026.

"An onsite electric power system 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 anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained."

Potential Impact Number: 25712 **SRP Section Number:** 15.1.5

Document: ADVANCE SER CE CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

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This potential impact was duplicated from PI# 23027.

"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 meet the GDC 17 requirements given above. If the existing analyses did meet the GDC 17 requirements, ABB-CE should have reanalyzed the transient and accident analyses in accordance with GDC 17 and submit the results for the staff to review. This was identified as DSER Open Item 15.1-5.

By letter dated December 18, 1992, ABB-CE agreed to comply with GDC 17, which requires that the LOOP not be treated as a single failure. ABB-CE included the results of reanalysis in the CESSAR-DC (Amendment R) Section 6.3.3 and Chapter 15. The staff reviewed the submittal and found that ABB-CE considered the LOOP in all the events analyzed and applied the acceptance criteria specified in the related SRP sections for the event with or without LOOP. The staff concludes that ABB-CE's approach complies with the requirements of GDC 17. On this basis, DSER Open Item 15.1-5 is resolved."

Potential Impact Number: 25718 **SRP Section Number:** 15.2.6

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

This potential impact was duplicated from PI# 24274.

The staff's review of acceptability of the ODYNA and REDYA computer codes for ABWR transient analyses is documented in Section 15.1 of the ABWR FSER.

Potential Impact Number: 25719 **SRP Section Number:** 15.2.6

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

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Summary:

This potential impact was duplicated from PI# 24275.

Tables 15.3 and 15.4 of the System 80+ FSER lists topical reports and letters and NRC approval letters for analyses methods and computer codes used in transient and accident analyses.

Potential Impact Number: 25720 **SRP Section Number:** 15.1.5.A

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

This potential impact was duplicated from PI# 24506.

In conducting its evaluation on the radiological consequences of a locked rotor event, ABB-CE used appropriate guidance from draft NUREG-1465 on the revised source term.

Potential Impact Number: 25721 **SRP Section Number:** 15.2.6

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

This potential impact was duplicated from PI# 24523.

NRC staff approved analysis methods for transient and accident analyses are discussed in Section 15.1 of the System 80+ FSER.

Potential Impact Number: 25722 **SRP Section Number:** 15.2.6

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Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

This potential impact was duplicated from PI# 24524.

NRC staff review of the ODYNA and REDYA computer codes is discussed in Section 15.1 of the ABWR FSER.

Potential Impact Number: 25723 **SRP Section Number:** 15.4.2

Document: DRAFT SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

This potential impact was duplicated from PI# 22920.

In accordance with the requirements of GDC 17, loss of offsite power (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. For some transient analyses (such as the loss of condenser vacuum event analysis), the applicant did not postulate the unavailability of offsite power as part of the event, but rather took it as an additional single failure to show limiting fault criteria were not exceeded. The applicant is required to discuss each of the transient and accident analyses in the CESSAR to justify that the analyses meet the GDC 17 requirements given above. If the existing analyses do not meet the GDC 17 requirements, the applicant should reanalyze the transient and accident analyses in accordance with GDC 17 and submit the results for the staff to review.

Potential Impact Number: 25724 **SRP Section Number:** 15.4.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, Criterion 17

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Summary:

This potential impact was duplicated from PI# 22921.

"An onsite electric power system 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 anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained."

Potential Impact Number: 25725 **SRP Section Number:** 15.4.2

Document: DRAFT SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

This potential impact was duplicated from PI# 22922.

In CESSAR Section 15.2, the applicant analyzes various transients resulting from a decrease in heat removal by the secondary system and identifies the limiting cases for the consideration of integrity of RCS primary system boundary and fuel rod cladding to withstand the consequences of transients. With consideration of a single failure, the applicant noted that the loss of condenser vacuum (LOCV) event with a loss of offsite power (LOOP) following a turbine trip is the most limiting case with respect to both peak RCS pressure and low DNBR. With consideration of the worst single failure of a LOOP, the limiting fuel cladding damage resulting from the LOCV event is less than 1.8 percent of total fuel rods. As required by GDC 17 (discussed in Section 15.1), a LOOP

should be included in the analysis and the LOCV event with LOOP should be analyzed as a moderate-frequency event, and therefore, the calculated DNBR should meet the 95/95 DNBR limit as discussed in Section 4.4 of this DSER. Since the fuel failure was predicted, the staff concludes that the results of the LOCV are not acceptable. The applicant should provide acceptable results using the approved methods for the LOCV event analysis for the staff to review.

Potential Impact Number: 25731 **SRP Section Number:** 10.4.4

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

This issue involved the classification of the main steam drain lines and bypass piping between the first normally-closed valve and the condenser hotwell as well as the hotwell itself in boiling water reactors. In SECY 93-087, the staff recommended that for both the evolutionary and passive ALWRs the Commission approve the position that neither the main steam drain and bypass lines from the first valve up to the condenser inlet, nor the piping between the turbine stop valve and the turbine inlet should be classified as safety-related or as seismic Category I. Rather, these lines should be analyzed using a dynamic seismic analysis to demonstrate structural integrity under SSE loading conditions. The turbine stop, control, and bypass valves and the main steam lines from the turbine control valves to the turbine shall meet all of the quality group and quality assurance guidelines specified in SRP Section 3.2.2, Appendix A. Further, that seismic analyses are performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after the SSE. In the SRM for SECY 93-087 the Commission approved the above described approach and positions for classifying the mainsteam lines and systems for boiling water reactors.

Potential Impact Number: 25733 **SRP Section Number:** 13.2.2

Document: CODE OF FED. REGS 10CFR19

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Title: NOTICES, INSTRUCTIONS, AND REPORTS TO WORKERS: INSPECTION AND INVESTIGATIONS
Location: 19.12 Instruction to workers

Summary:

Federal Register Notice 60 FR 36038 identifies final rulemaking with regard to 10 CFR 20 and associated changes to 10 CFR 19.12.

10 CFR 19.12 is revised to reflect changes to radiation protection requirements. In particular, the requirements were revised to required all individuals who are likely to receive in a year an occupational dose in excess of 100 mrem to receive training related to radiation protection and procedures to minimize exposure.

Potential Impact Number: 25734 **SRP Section Number:** 15.3.3 - 15.3.4

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 17

Summary:

Electric power systems. An onsite electric power system 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 anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Potential Impact Number: 25735 **SRP Section Number:** 15.3.3 - 15.3.4

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

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Potential Impacts

Summary:

GDC 17 in 10 CFR Part 50, Appendix A requires, in part: An onsite electric power system 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 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.

Potential Impact Number: 25736 **SRP Section Number:** 2.5.2

Document: FINAL SER CE80 CH 2
Title: SITE CHARACTERISTICS

Summary:

The COL applicant referencing the certified System 80+ design will be required to develop site-specific geological, seismological, and geotechnical data and will submit these data to the NRC staff for review. These data should be comparable to the design-basis assumptions regarding the SSE, including the verification of the ground-motion response spectra. The staff will compare the site-specific earthquake ground-motions to the ground motions used as input for the design certification in accordance with procedures and acceptance criteria in SRP Section 2.5.2,"Vibratory Ground Motion."

Potential Impact Number: 25737 **SRP Section Number:** 2.5.2

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Document: NRC NOTICE 94-32
Title: REVISED SEISMIC HAZARD ESTIMATES

Summary:

During the 1980's, two probabilistic seismic hazard studies were undertaken, one by the Lawrence Livermore National Laboratory (LLNL), as documented in NUREG/CR-5250, "Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains," dated January 1989, and another by the Seismicity Owners Group/Electric Power Research Institute as documented in EPRI NP-4726, "Seismic Hazard Methodology for the Central and Eastern United States," dated July 1986. These studies were performed in an effort to improve the estimation of the seismic hazard that can be expected anywhere in the United States east of te Rocky Mountains. The results of these studies produced significantly different seismic hazard predictions in terms of the induced ground motion. In 1992, LLNL conducted another study under the sponsorship of the NRC using a revised method for elicitation of seismicity parameters, ground motion estimates, and the uncertainties associated with those parameters and estimates. The results of the new study were published for comment in draft NUREG-1488 in October 1993.

The results of the new LLNL study are substantially different from those of the previous LLNL study. Also, the EPRI study has not been revised since its publication in 1986. Therefore, the staff considers the results of the new LLNL study to represent the best available information on seismic hazard estimates.

Potential Impact Number: 25738 **SRP Section Number:** 2.5.2

Document: FINAL SER ABWR CH 2
Title: SITE CHARACTERISTICS

Summary:

The Combined Operating License (COL) applicant will develop site-specific geological, seismological, and geotechnical data and will compare the site-specific SSE ground response spectra with the design ground response spectra of SSAR Section 2.3.1.2.

Potential Impact Number: 25739 **SRP Section Number:** 6.2.1.1.B

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Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 4

Summary:

Under the broad scope revision to GDC 4 (52 FR 41288, dated October 27, 1987), the NRC allows the exclusion from structural design consideration of the dynamic effects of pipe ruptures in nuclear power plants. However, it must first be demonstrated that the probability of pipe rupture is extremely low under conditions consistent with the design bases for the piping. Demonstration of low probability pipe rupture requires a deterministic fracture mechanics analysis that evaluates the stability of postulated small, through-wall flaws in piping and the ability to detect leakage through the flaws long before the flaws could grow to unstable sizes. The concept underlying such analyses is referred to as "leak-before-break."

Potential Impact Number: 25740 **SRP Section Number:** 15.3.1 - 15.3.2

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

GDC 17 in 10 CFR Part 50, Appendix A requires, in part: An onsite electric power system 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 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.

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Potential Impact Number: 25741 **SRP Section Number:** 15.3.1 - 15.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 17

Summary:

Electric power systems. An onsite electric power system 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 anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Potential Impact Number: 25742 **SRP Section Number:** 15.5.1 - 15.5.2

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

For the pressure regulator down-scale failure to occur, all three channels would have to suffer a common-mode failure before the pressure regulator would go either up or down the scale. If the pressure regulator failed down-scale, the steam control valves would close causing the reactor pressure and reactivity to increase. When reanalyzing this postulated event, GE proposed to assume that any fuel rods that achieve transition boiling fail for the purposes of the radiological dose calculation. The staff includes this postulated event in the special category of anticipated transients involving a common-mode software failure because of the uncertainty that such an event will occur during the plant lifetime. GE originally categorized this event as an accident. The staff believes that it is more appropriate to apply a special classification for such an event. The staff required that GE demonstrate that this special event will not exceed the limits of 10 percent of 10 CFR Part 100, which the staff considers appropriate for an event of such postulated frequency.

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Potential Impact Number: 25743 **SRP Section Number:** 15.5.1 - 15.5.2

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

GDC 17 in 10 CFR Part 50, Appendix A requires, in part: An onsite electric power system 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 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.

Potential Impact Number: 25744 **SRP Section Number:** 15.5.1 - 15.5.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 17

Summary:

Electric power systems. An onsite electric power system 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 anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Potential Impact Number: 25745 **SRP Section Number:** 6.2.4

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

SECY 93-087 contains staff positions and guidance for advanced light water reactors on midloop operation and risk assessments for shutdown operations.

Potential Impact Number: 25746 **SRP Section Number:** 3.10

Document: FINAL SER CE80 CH 1
Title: INTRODUCTION AND GENERAL DISCUSSION

Summary:

In accordance with 10 CFR 52.48, the staff used the current regulations in 10 CFR Parts 20, 50, 73, and 100 in reviewing ABB-CE's application for design certification. During this review, the staff recognized that if certain regulations were applied to the System 80+ design, the use of those regulations would not serve the underlying purpose of the rule or would not be needed for achieving the fundamental intent of the rule. These exemptions are discussed in the sections of this report listed below.

In the SRM pertaining to SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor Designs," the Commission approved the staff's recommendation to proceed with design-specific rulemakings as part of individual design certifications to resolve selected technical and severe accident issues as "applicable regulations" for the advanced boiling water reactor and System 80+ standard designs. Among these issues are staff positions that deviate from or are not embodied in current regulations applicable to the System 80+ design.

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These issues were addressed in various Commission papers, such as SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced LWR designs." The "applicable regulations" that are specific to the System 80+ design are identified and evaluated in this report. The proposed design certification rule will then include these "applicable regulations" for the System 80+ design for the purposes of 10 CFR 52.48, 52.54, 52.59, and 52.63. These "applicable regulations" are discussed in the sections of this report, including Section 3.1.1, "Exemption from operating-basis earthquake design requirement."

Potential Impact Number: 25747 **SRP Section Number:** 3.10

Document: FINAL SER ABWR CH 1
Title: INTRODUCTION AND GENERAL DISCUSSION

Summary:

In the SRM pertaining to SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor Designs," the Commission approved the staff's recommendation to proceed with design-specific rulemakings as part of design certification rulemakings to establish selected technical and severe accident issues as "applicable regulations" for the ABWR and System 80+ standard designs. These issues included staff positions that deviated from or were not embodied in current regulations applicable to the standard designs.

These issues were addressed in various Commission papers, such as SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolution and Advanced LWR Designs." The "applicable regulations" that are specific to the ABWR design are identified and evaluated in this report, including Section 3.1.1, Exemption from operating basis earthquake design requirement.

Potential Impact Number: 25748 **SRP Section Number:** 10.3.6

Document: NRC GENERIC LETTER 91-17 Revision Date: 10/17/91
Title: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

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This letter provides information on the U.S. Nuclear Regulatory Commission's (NRC's) resolution of Generic Safety Issue (GSI) 29, "Bolting Degradation or Failure in Nuclear Power Plants." The scope of GSI-29 includes all safety-related bolts, studs, embedments, machine/cap screws, other special threaded fasteners, and all their associated nuts and washers. The NRC has resolved this issue without developing any new requirements, based on licensees continuing to implement actions taken in response to previous NRC guidance and the industry's initiatives in this area. The GSI-29 resolution encompasses a number of industry initiatives and previous NRC actions. The bases for resolution are further documented in the enclosed NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," which was published by NRC in June 1990. Since establishing GSI-29 in 1982, the staff has issued a number of bulletins, generic letters, and information notices on bolting events judged to be safety-significant (See NUREG-1339). The bulletins and some of the generic letters required both one-time actions and continuing programs. These actions and continuing programs were considered by the staff in resolving GSI-29.

Because safety-related bolting may continue to pose some safety concerns, it is important that all licensees continue their actions in accordance with commitments made in response to NRC bulletins and generic letters on bolting events as listed in NUREG-1339.

The NRC staff further believes that licensees can significantly improve safety and reduce costs by implementing the industry's recommendations delineated in two EPRI reports. The staff suggests that, to efficiently implement these recommendations, licensees may wish to review the following industry-developed information: a. EPRI NP-5769, volumes 1 and 2; b. EPRI NP-5067, volumes 1 and 2; and c. EPRI videotapes, "Pressure Boundary Bolting Problems," parts I, II and III; and review the NRC staff report, NUREG-1339, which discusses the NRC's evaluation of, and exceptions to, EPRI NP-5769.

The staff agrees that an effective means of ensuring bolting reliability, as recommended in EPRI NP-5769, would be to develop and implement plant-specific bolting integrity programs that encompass all safety-related bolting. NUREG-1339 includes recommendations and guidelines for the content of a comprehensive bolting integrity program. EPRI NP-5769 provides additional details on bolting integrity. The plant-specific bolting integrity program could incorporate licensee's commitments for continuing actions made in response to the previously issued NRC bulletins and generic letters listed in NUREG-1339.

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Potential Impact Number: 25749 **SRP Section Number:** 3.9.5

Document: NRC GENERIC LETTER 91-17 Revision Date: 10/17/91
Title: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

This letter provides information on the U.S. Nuclear Regulatory Commission's (NRC's) resolution of Generic Safety Issue (GSI) 29, "Bolting Degradation or Failure in Nuclear Power Plants." The scope of GSI-29 includes all safety-related bolts, studs, embedments, machine/cap screws, other special threaded fasteners, and all their associated nuts and washers. The NRC has resolved this issue without developing any new requirements, based on licensees continuing to implement actions taken in response to previous NRC guidance and the industry's initiatives in this area. The GSI-29 resolution encompasses a number of industry initiatives and previous NRC actions. The bases for resolution are further documented in the enclosed NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," which was published by NRC in June 1990. Since establishing GSI-29 in 1982, the staff has issued a number of bulletins, generic letters, and information notices on bolting events judged to be safety-significant (See NUREG-1339). The bulletins and some of the generic letters required both one-time actions and continuing programs. These actions and continuing programs were considered by the staff in resolving GSI-29.

Because safety-related bolting may continue to pose some safety concerns, it is important that all licensees continue their actions in accordance with commitments made in response to NRC bulletins and generic letters on bolting events as listed in NUREG-1339.

The NRC staff further believes that licensees can significantly improve safety and reduce costs by implementing the industry's recommendations delineated in two EPRI reports. The staff suggests that, to efficiently implement these recommendations, licensees may wish to review the following industry-developed information: a. EPRI NP-5769, volumes 1 and 2; b. EPRI NP-5067, volumes 1 and 2; and c. EPRI videotapes, "Pressure Boundary Bolting Problems," parts I, II and III; and review the NRC staff report, NUREG-1339, which discusses the NRC's evaluation of, and exceptions to, EPRI NP-5769.

The staff agrees that an effective means of ensuring bolting reliability, as recommended in EPRI NP-5769, would be to develop and implement plant-specific bolting integrity programs that encompass all safety-related bolting. NUREG-1339 includes

recommendations and guidelines for the content of a comprehensive bolting integrity program. EPRI NP-5769 provides additional details on bolting integrity. The plant-specific bolting integrity program could incorporate licensee's commitments for continuing actions made in response to the previously issued NRC bulletins and generic letters listed in NUREG-1339.

Potential Impact Number: 25750 **SRP Section Number:** 6.2.7

Document: NRC GENERIC LETTER 91-17 Revision Date: 10/17/91
Title: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

This letter provides information on the U.S. Nuclear Regulatory Commission's (NRC's) resolution of Generic Safety Issue (GSI) 29, "Bolting Degradation or Failure in Nuclear Power Plants." The scope of GSI-29 includes all safety-related bolts, studs, embedments, machine/cap screws, other special threaded fasteners, and all their associated nuts and washers. The NRC has resolved this issue without developing any new requirements, based on licensees continuing to implement actions taken in response to previous NRC guidance and the industry's initiatives in this area. The GSI-29 resolution encompasses a number of industry initiatives and previous NRC actions. The bases for resolution are further documented in the enclosed NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," which was published by NRC in June 1990. Since establishing GSI-29 in 1982, the staff has issued a number of bulletins, generic letters, and information notices on bolting events judged to be safety-significant (See NUREG-1339). The bulletins and some of the generic letters required both one-time actions and continuing programs. These actions and continuing programs were considered by the staff in resolving GSI-29.

Because safety-related bolting may continue to pose some safety concerns, it is important that all licensees continue their actions in accordance with commitments made in response to NRC bulletins and generic letters on bolting events as listed in NUREG-1339.

The NRC staff further believes that licensees can significantly improve safety and reduce costs by implementing the industry's recommendations delineated in two EPRI reports. The staff suggests that, to efficiently implement these recommendations, licensees may wish to review the following industry-developed information: a. EPRI

NP-5769, volumes 1 and 2; b. EPRI NP-5067, volumes 1 and 2; and c. EPRI videotapes, "Pressure Boundary Bolting Problems," parts I, II and III; and review the NRC staff report, NUREG-1339, which discusses the NRC's evaluation of, and exceptions to, EPRI NP-5769.

The staff agrees that an effective means of ensuring bolting reliability, as recommended in EPRI NP-5769, would be to develop and implement plant-specific bolting integrity programs that encompass all safety-related bolting. NUREG-1339 includes recommendations and guidelines for the content of a comprehensive bolting integrity program. EPRI NP-5769 provides additional details on bolting integrity. The plant-specific bolting integrity program could incorporate licensee's commitments for continuing actions made in response to the previously issued NRC bulletins and generic letters listed in NUREG-1339.

Potential Impact Number: 25751 **SRP Section Number:** 6.2.4

Document: FINAL SER ABWR CH 19
Title: SEVERE ACCIDENTS

Summary:

The ABWR FSER section 19.3 contains NRC staff reviews and guidance on shutdown risk assessments.

Potential Impact Number: 25752 **SRP Section Number:** 6.2.4

Document: NRC GENERIC LETTER 88-17
Title: LOSS OF DECAY HEAT REMOVAL

Summary:

Generic Letter 88-17 includes eight expeditious actions and six programmed enhancements that were required concerning operation of PWRs during shutdown cooling conditions. The positions in this Generic Letter apply whenever there is irradiated fuel in the reactor vessel. The guidance includes requirements for procedures and administrative controls to assure containment closure can be achieved

prior to uncovering the core during a loss of decay heat removal during shutdown conditions.

Potential Impact Number: 25753 **SRP Section Number:** 6.2.4

Document: NUREG 1449
Title: SHUT DOWN AND LOW-POWER OPERATION AT NUCLEAR POWER PLANTS IN THE UNITED STATES. FINAL REPORT

Summary:

NUREG-1449 documents the NRC staff's evaluations and recommendations for regulations and guidance in key areas related to shutdown and low power operations.

Potential Impact Number: 25754 **SRP Section Number:** 6.2.1

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

SECY 93-087 contains staff positions and guidance for advanced light water reactors on midloop operation and risk assessments for shutdown operations.

Potential Impact Number: 25755 **SRP Section Number:** 6.2.1

Document: FINAL SER ABWR CH 19
Title: SEVERE ACCIDENTS

Summary:

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The ABWR FSER section 19.3 contains NRC staff reviews and guidance on shutdown risk assessments.

Potential Impact Number: 25756 **SRP Section Number:** 6.2.1

Document: NRC GENERIC LETTER 88-17
Title: LOSS OF DECAY HEAT REMOVAL

Summary:

Generic Letter 88-17 includes eight expeditious actions and six programmed enhancements that were required concerning operation of PWRs during shutdown cooling conditions. The positions in this Generic Letter apply whenever there is irradiated fuel in the reactor vessel. The guidance includes requirements for analyses that may require additional containment analyses during shutdown conditions (such as time to core uncovery and containment closure during a loss of decay heat removal capability).

Potential Impact Number: 25757 **SRP Section Number:** 6.2.1

Document: NUREG 1449
Title: SHUT DOWN AND LOW-POWER OPERATION AT NUCLEAR POWER PLANTS IN THE UNITED STATES. FINAL REPORT

Summary:

NUREG-1449 documents the NRC staff's evaluations and recommendations for regulations and guidance in key areas related to shutdown and low power operations.

Potential Impact Number: 25758 **SRP Section Number:** 6.2.1.1.A

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

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Summary:

SECY 93-087 contains staff positions and guidance for advanced light water reactors covering midloop operations and risk assessments for shutdown operations.

Potential Impact Number: 25759 **SRP Section Number:** 6.2.1.1.A

Document: FINAL SER CE80 CH 19
Title: SEVERE ACCIDENTS

Summary:

The CE80+ FSER Section 19.3 contains NRC staff reviews and guidance on the shutdown risk assessments.

Potential Impact Number: 25760 **SRP Section Number:** 6.2.1.1.A

Document: NRC GENERIC LETTER 88-17
Title: LOSS OF DECAY HEAT REMOVAL

Summary:

Generic Letter 88-17 includes eight expeditious actions and six programmed enhancements that were required concerning operation of PWRs during shutdown cooling conditions. The positions in this Generic Letter apply whenever there is irradiated fuel in the reactor vessel. The guidance includes analyses to supplement existing information and includes requirements for procedures and administrative controls to assure containment closure can be achieved prior to uncovering the core during a loss of decay heat removal during shutdown conditions.

Potential Impact Number: 25761 **SRP Section Number:** 6.2.1.1.A

Document: NUREG 1449

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Title: SHUT DOWN AND LOW-POWER OPERATION AT NUCLEAR POWER PLANTS IN THE UNITED STATES. FINAL REPORT

Summary:

NUREG-1449 documents the NRC staff's evaluations and recommendations for regulations and guidance in key areas related to shutdown and low power operation.

Potential Impact Number: 25762 **SRP Section Number:** 6.2.1.1.B

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

SECY 93-087 contains staff positions and guidance for advanced light water reactors covering midloop operations and risk assessments for shutdown operations.

Potential Impact Number: 25763 **SRP Section Number:** 6.2.1.1.B

Document: FINAL SER CE80 CH 19
Title: SEVERE ACCIDENTS

Summary:

The CE80+ FSER Section 19.3 contains NRC staff reviews and guidance on the shutdown risk assessments.

Potential Impact Number: 25764 **SRP Section Number:** 6.2.1.1.B

Document: NRC GENERIC LETTER 88-17
Title: LOSS OF DECAY HEAT REMOVAL

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Summary:

Generic Letter 88-17 includes eight expeditious actions and six programmed enhancements that were required concerning operation of PWRs during shutdown cooling conditions. The positions in this Generic Letter apply whenever there is irradiated fuel in the reactor vessel. The guidance includes analyses to supplement existing information and includes requirements for procedures and administrative controls to assure containment closure can be achieved prior to uncovering the core during a loss of decay heat removal during shutdown conditions.

Potential Impact Number: 25765 **SRP Section Number:** 6.2.1.1.B

Document: NUREG 1449
Title: SHUT DOWN AND LOW-POWER OPERATION AT NUCLEAR POWER PLANTS IN THE UNITED STATES. FINAL REPORT

Summary:

NUREG-1449 documents the NRC staff's evaluations and recommendations for regulations and guidance in key areas related to shutdown and low power operation.

Potential Impact Number: 25766 **SRP Section Number:** 2.3.1

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

SSAR Section 3.3.2 specifies that all seismic Category I structures exposed to tornado forces are designed to resist a maximum tornado wind speed of 483 km/hr (300 mi/hr) and translational wind velocity of 97 km/hr (60 mi/hr). This also implies a maximum tangential velocity of 386 km/hr (240 mi/hr). Also specified are a simultaneous atmospheric pressure drop to 13.8 kPa (2.00 lbf/in2) at the rate of 8.3 kPa/sec (1.20 lbf/in2/sec) and the radius of maximum tornado is 45.7 m (150 ft). In SECY-93-087, the staff recommended that the Commission approve its position to employ a maximum tornado wind speed of 483 km/hr (300 mph) in the design of evolutionary and passive ALWRs. In its SRM dated July 21, 1993, the Commission approved the staff's position.

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On the basis of this evaluation, the staff concludes that the ABWR design meets the Commission-approved staff recommendation for design basis tornado and is acceptable.

Potential Impact Number: 25767 **SRP Section Number:** 2.3.1

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

ABB-CE states, in the CESSAR-DC, that all seismic Category I structures exposed to tornado forces and needed for the safe shutdown of the plant are designed to resist tornado effects in accordance with the interim staff position in RG 1.76, and the tornado missile spectrum is in accordance with SRP Section 3.5.1.4. CESSAR-DC Table 2.0-1 specifies a maximum tornado wind speed of 531 km/hr (330 mph), maximum rotational tornado wind speed of 418 km/hr (260 mph), and a maximum translational tornado wind speed of 113 km/hr (70 mph). Also specified are a simultaneous atmospheric pressure drop to 16.5 kPa (2.4 psi) at the rate of 11.7 kPa/sec (1.7 psi/sec) and the radius of 45.7 m (150 ft). Because the values of these parameters are more severe than the tornado design-basis requirements specified in SECY-39-087 as approved in the July 21, 1993, SRM, the staff concludes that the ABB-CE System 80+ tornado design basis is acceptable.

Potential Impact Number: 25769 **SRP Section Number:** 6.2.1.1.C

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

SECY 93-087 contains staff positions and guidance for advanced light water reactors covering midloop operations and risk assessments for shutdown operations.

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Potential Impact Number: 25770 **SRP Section Number:** 6.2.1.1.C

Document: FINAL SER ABWR CH 19
Title: SEVERE ACCIDENTS

Summary:

The ABWR FSER Section 19.3 contains NRC staff reviews and guidance on the shutdown risk assessments.

Potential Impact Number: 25771 **SRP Section Number:** 6.2.1.1.C

Document: NUREG 1449
Title: SHUT DOWN AND LOW-POWER OPERATION AT NUCLEAR POWER PLANTS IN THE UNITED STATES. FINAL REPORT

Summary:

NUREG-1449 documents the NRC staff's evaluations and recommendations for regulations and guidance in key areas related to shutdown and low power operation.

Potential Impact Number: 25772 **SRP Section Number:** 5.2.2

Document: NRC GENERIC LETTER 82-16
Title: NUREG-0737 TECHNICAL SPECIFICATIONS

Summary:

This letter describes appropriate technical specifications (TS) to address TMI action plan item II.K.3.3.

NUREG-0660 stated that safety and relief valve failures be reported promptly and challenges be reported annually. The sections of your TSs that discuss reporting

requirements should be accordingly changed; model TSs are given in Enclosure 2. Note that an acceptable alternative would be to report challenges monthly.

Potential Impact Number: 25775 **SRP Section Number:** 5.2.2

Document: NUREG 1430
Title: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS

Summary:

Monthly Operating Reports

Routine reports of operating statistics and shutdown experience[, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

Potential Impact Number: 25776 **SRP Section Number:** 5.2.2

Document: NUREG 1431
Title: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS

Summary:

Monthly Operating Reports

Routine reports of operating statistics and shutdown experience[, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

Potential Impact Number: 25777 **SRP Section Number:** 5.2.2

Document: NUREG 1432

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Title: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS

Summary:

Monthly Operating Reports

Routine reports of operating statistics and shutdown experience[, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

Potential Impact Number: 25778 **SRP Section Number:** 5.2.2

Document: NUREG 1433
Title: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/4

Summary:

Monthly Operating Reports

Routine reports of operating statistics and shutdown experience[, including documentation of all challenges to the safety/relief valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

Potential Impact Number: 25779 **SRP Section Number:** 5.2.2

Document: NUREG 1434
Title: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS, BWR/6

Summary:

Monthly Operating Reports

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Routine reports of operating statistics and shutdown experience[, including documentation of all challenges to the safety/relief valves,] shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

Potential Impact Number: 25780 **SRP Section Number:** 3.7.1

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff evaluated the use of a single-earthquake design (i.e., elimination of operating-basis earthquake (OBE)) for structures systems, and components (SSCs). Originally, implementation of the OBE was included in applicable sections of the ABWR SSAR. After many discussions with the staff, General Electric (GE) stated that it might opt to use the single-earthquake design approach. In a letter to GE dated September 11, 1992, the staff provided preliminary guidance concerning what types of analyses and information would be required in the SSAR for the staff to approve design of SSCs for the ABWR without the OBE. This was Open Item 3.1-1 in the DFSER. The staff's evaluation of this issue is discussed in Chapter 3.1 of the ABWR FSER.

Approval to eliminate the OBE from the design basis would represent a staff position which would require modifying SRP 3.7.1 to modify references to the OBE appropriately.

Potential Impact Number: 25781 **SRP Section Number:** 3.7.1

Document: FINAL SER CE80 CH 2
Title: SITE CHARACTERISTICS

Summary:

10 CFR 100 App A and SRP 2.5.2 define OBE as at least 1/2 the SSE. For the CE80, SSE is anchored at 0.3 g, however, OBE is anchored at 0.1 g. In accordance with 10 CFR 52.48, the staff used the current regulations in 10 CFR Parts 20, 50, 73, and 100 in reviewing ABB-CE's application for design certification. During this review, the staff

recognized that if certain regulations were applied to the System 80+ design, the use of those regulations would not serve the underlying purpose of the rule or would not be needed for achieving the fundamental intent of the rule. These exemptions are discussed in the sections of this report listed below.

In the SRM pertaining to SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor Designs," the Commission approved the staff's recommendation to proceed with design-specific rulemakings as part of individual design certifications to resolve selected technical and severe accident issues as "applicable regulations" for the advanced boiling water reactor and System 80+ standard designs. Among these issues are staff positions that deviate from or are not embodied in current regulations applicable to the System 80+ design.

These issues were addressed in various Commission papers, such as SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced LWR designs." The "applicable regulations" that are specific to the System 80+ design are identified and evaluated in this report. The proposed design certification rule will then include these "applicable regulations" for the System 80+ design for the purposes of 10 CFR 52.48, 52.54, 52.59, and 52.63. These "applicable regulations" are discussed in the sections of this report, including Section 3.1.1, "Exemption from operating-basis earthquake design requirement."

Potential Impact Number: 25782 **SRP Section Number:** 3.7.1

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

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 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.

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 Impact Number: 25783 **SRP Section Number:** 4.4

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

SECY 93-087 contains staff positions and guidance for advanced light water reactors covering midloop operations and risk assessments for shutdown operations.

Potential Impact Number: 25784 **SRP Section Number:** 4.4

Document: NRC GENERIC LETTER 88-17
Title: LOSS OF DECAY HEAT REMOVAL

Summary:

Generic Letter 88-17 includes eight expeditious actions and six programmed enhancements that were required concerning operation of PWRs during shutdown cooling conditions. The positions in this Generic Letter apply whenever there is irradiated fuel in the reactor vessel. The guidance includes new analyses to supplement existing information regarding shutdown conditions.

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Potential Impact Number: 25785 **SRP Section Number:** 4.4

Document: FINAL SER CE80 CH 19
Title: SEVERE ACCIDENTS

Summary:

The ABB-CE80+ FSER Section 19.3 contains NRC staff reviews and guidance on the shutdown risk assessments.

Potential Impact Number: 25786 **SRP Section Number:** 4.4

Document: NUREG 1449
Title: SHUT DOWN AND LOW-POWER OPERATION AT NUCLEAR POWER PLANTS IN THE UNITED STATES. FINAL REPORT

Summary:

NUREG-1449 documents the NRC staff's evaluations and recommendations for regulations and guidance in key areas related to shutdown and low power operation.

Potential Impact Number: 25787 **SRP Section Number:** 3.8.3

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff's review of the controlling load combinations finds that, in general, the load combinations with the SSE control the design of steel structures although there may be specific cases where the load combinations with the OBE control the design. Similarly, an examination of the pertinent load combinations for concrete structures, including the containment structure, should lead to the same conclusion that the OBE loads, in most cases, do not control the outcome of the structural design.

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In the design of the containment, the staff reviewed the extent to which the elimination of the OBE from the load combinations would lead to a reduction of the safety margin. An examination of the nuclear structural design practice and the SRP load combination equations, however, show that the major dynamic load for the overall design of structures is either the OBE or the SSE. All other potential dynamic loads are conservatively accounted for in the definition of equivalent dead and live loads or only produce local effects that are handled by local reinforcement details. Therefore, the staff concludes that no reduction in safety margins of concrete and steel structures results from the elimination of the OBE as a design requirement.

Potential Impact Number: 25788 **SRP Section Number:** 3.8.3

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

The staff's review of the controlling load combinations finds that, in general, the load combinations with the SSE control the design of steel structures although there may be specific cases where the load combinations with the OBE control. Similarly, an examination of the pertinent load combinations for concrete structures should lead to the same conclusion that the OBE loads, in most cases, do not control the outcome of the structural design.

All other potential dynamic loads are conservatively accounted for in the definition of equivalent dead and live loads, except those that produce local effects that are handled by local reinforcement details. Therefore, it can be concluded that no reduction in safety margins of concrete and steel structures results from the elimination of the OBE as a design requirement.

Potential Impact Number: 25789 **SRP Section Number:** 3.8.3

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

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Summary:

Presently, 10 CFR 100 requires that the magnitude of the OBE be at least one-half that of the SSE. It has been an industry wide experience that such a requirement leads to a design that is governed by the OBE requirements and produces unnecessary and inconsistent margins for the SSE loading. This requirement was included in the regulation when the staff did not have substantial experience with the seismic resistance of plants that incorporated OBE design at half the SSE value. Since then a number of research programs have been conducted including a large industry effort on testing and observation of actual earthquake experience of industrial facilities; consequently, the NRC funded Piping Review Committee has concluded that the OBE at existing plants are too high, therefore, controlling the design of some safety systems, and recommended that the OBE be decoupled from the SSE. Certain interim measures, such as allowing somewhat higher damping values for piping analysis, have been taken to partially implement the Piping Review Committee recommendations (NUREG 1061, 1984). But the complete implementation of the recommendations would involve a revision of 10 CFR 100, Appendix A. Because of higher priority work, the effort on revision of this regulation has been postponed. It should be noted that the Commission has, in certain site specific cases, previously approved OBEs of less than one-half the SSE.

EPRI has requested that NRC regulations be changed to reduce the magnitude of the OBE relative to the SSE as a basis for the design. All evolutionary ALWR vendors agree with the request. GE has stated that they agree with EPRI in principle, however, the ABWR design uses an OBE that is one-half the SSE; therefore, this is a non-issue for the ABWR. EPRI has identified this as an optimization issue.

The staff agrees that the OBE should not control the design of safety systems. However, a staff position on this issue to be applied generically to all future designs has not yet been fully developed. For the evolutionary reactors, the staff will consider requests to decouple the OBE from the SSE on a design-specific basis. Such a decoupling would require an exemption to the Commission's regulations, therefore the staff recommends that the Commission approve this design-specific relief approach.

Potential Impact Number: 25790 **SRP Section Number:** 5.4

Document: NRC GENERIC LETTER 90-06
Title: RESOLUTION OF GENERIC ISSUE 70, 'POWER-OPERATED RELIEF-VALVE AND BLOCK VALVE RELIABILITY,' AND GENERIC ISSUE 94, 'ADDITIONAL

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LOW-TEMPERATURE OVERPRESSURE PROTECTION FOR LIGHT-WATER REACTORS,'

Summary:

The purpose of this generic letter is to advise pressurized water reactor (PWR) licensees and construction permit (CP) holders of the staff positions resulting from the resolution of Generic Issue 70 (GI-70) and Generic Issue 94 (GI-94) regarding power operated relief valve and block valve reliability, and low-temperature ovepressure protection systems, respectively.

Potential Impact Number: 25792 **SRP Section Number:** 3.6.1

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

SECY 93-087 provides additional guidance and discussion on the staff position regarding all systems and subsystems that have not been designed to full RCS pressure. Such systems should include the following: (1) the capability for leak testing of the pressure isolation valves; (2) valve position indication that is available in the control room when isolation valve operators are deenergized; and, (3) high-pressure alarms to warn control room operators when rising reactor coolant pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

Also, the staff provided clarification stating it will be necessary for a plant designer to demonstrate that any interfacing system for which the ultimate rupture strength (URS) is not at least equal to full RCS pressure could not practically be designed to meet such a criterion. The degree of isolation or number of barriers is not sufficient justification for using low-pressure components that can practically be designed to meet the full RCS URS criterion. In addition, piping runs should be designed to meet the full RCS URS criterion, as should all associated elements. The plant designer should make every effort to reduce the level of pressure challenge to all systems and subsystems connected to the RCS.

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Finally, in SECY 93-087 the staff recommended that the Commission approve the position that passive plants should also be reviewed for compliance with the intersystem LOCA criteria approved in the Commission's SRM for SECY 90-016 dated June 26, 1990.

Potential Impact Number: 25830 **SRP Section Number:** 9.2.2

Document: NRC GENERIC LETTER 89-13
Title: SERVICE WATER SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT

Summary:

Action II. Conduct a test program to verify the heat transfer capability of all safety-related heat exchangers cooled by service water. The total test program should consist of an initial test program and a periodic retest program. Both the initial test program and the periodic retest program should include heat exchangers connected to or cooled by one or more open-cycle systems as defined above. Operating experience and studies indicate that closed-cycle service water systems, such as component cooling water systems, have the potential for significant fouling as a consequence of aging-related in-leakage and erosion or corrosion. The need for testing of closed-cycle system heat exchangers has not been considered necessary because of the assumed high quality of existing chemistry control programs. If the adequacy of these chemistry control programs cannot be confirmed over the total operating history of the plant or if during the conduct of the total testing program any unexplained downward trend in heat exchanger performance is identified that cannot be remedied by maintenance of an open-cycle system, it may be necessary to selectively extend the test program and the routine inspection and maintenance program addressed in Action III, below, to the attached closed-cycle systems.

Action III. Ensure by establishing a routine inspection and maintenance program for open-cycle service water system piping and components that corrosion, erosion, protective coating failure, silting, and biofouling cannot degrade the performance of the safety-related systems supplied by service water. The maintenance program should have at least the following purposes: A) To remove excessive accumulations of biofouling agents, corrosion products, and silt; and B) To repair defective protective coatings and corroded service water system piping and components that could adversely affect performance of their intended safety functions.

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Potential Impact Number: 25831 **SRP Section Number:** 2.4.7

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.55a, Codes and standards

Summary:

SRP Section 2.4.7 inappropriately specifies 10 CFR Part 50, 50.55a as an acceptance criterion for this section. General Design Criterion 1 would be a more appropriate quality standards criterion for structures, systems and components important to safety.

Potential Impact Number: 25836 **SRP Section Number:** 10.4.7

Document: FINAL SER ABWR CH 19
Title: SEVERE ACCIDENTS

Summary:

The probabilistic shutdown risk evaluation performed by GE concluded that the ABWR design could be maintained in configurations during Modes 3, 4, and 5 so that the estimated conditional core damage frequency was very low. The staff noted that estimates of core damage frequency and risk at shutdown for the ABWR design have much larger uncertainties than do such estimates for internal events in Modes 1 and 2. In the SSAR, GE presented sample plant configurations that help limit risk when shutdown occurs. The staff finds that the ABWR design, through appropriate shutdown planning, contingency planning, and operator instrumentation, can be configured and maintained in Modes 3, 4, and 5 in a manner that helps to reduce the chances of core damage or releases to the environment to a point where shutdown risk does not represent a disproportionate risk to the public.

Although the staff found that the most significant events to date have occurred at pressurized water reactors (PWRs), the potential vulnerability of BWR plants to shutdown and low-power events cannot be ignored. GE submitted a shutdown risk evaluation of the ABWR design (SSAR Appendix 19Q). The evaluation covered Modes 3 (hot shutdown), 4 (cold shutdown), and 5 (refueling). It included all aspects of the nuclear steam supply system (NSSS), containment, and all systems that support the

NSSS and containment. It did not address fuel handling outside the primary containment or fuel storage in the spent fuel pool.

Potential Impact Number: 25837 **SRP Section Number:** 10.4.7

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

In its SRM of June 26, 1990, the Commission approved the staff's position and endorsed the staff's position and endorsed consideration of the four additional ACRS requirements for mid-loop operation. However, the staff remained concerned that the vendors and EPRI have not adequately evaluated the overall question regarding the vulnerability of the ALWRs during shutdown and low-power operation. This issue was discussed with regard to evolutionary designs in a memorandum to the Commission dated September 5, 1990. The staff requested that ALWR vendors and EPRI assess shutdown and low-power risk, identifying design-specific vulnerabilities and weaknesses and documenting their consideration and incorporation of design features that minimize such vulnerabilities.

Potential Impact Number: 25838 **SRP Section Number:** 9.4.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue was raised in response to a 1982 ACRS letter which outlined deficiencies in the design, installation, maintenance and testing of control room systems for maintaining operator habitability. The solution to this issue 'has been identified.'

Potential Impact Number: 25839 **SRP Section Number:** 9.4.2

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Document: DRAFT SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

During its plant-specific reviews (of proposed resolution of GSI 83, Control Room Habitability), the staff will verify conformance with the applicable RGs 1.52, 1. 78, and 1.95; industry standards [ANSI/ANS 59.2, ANSI/ASME N509, ANSI/ASME N510, ASME Code AG-1, 'Code of Nuclear Air and Gas Treatment,' and ASTM D3803, 'Standard Test Methods for Radiological Testing of Nuclear- Grade Gas-Phase Absorbents']; and evaluation methodology, and will verify that technical specifications and surveillance requirements for the control room habitability system, as a whole, to meet GDC 4, 5, and 19 have been proposed. This will demonstrate that the control room will adequately protect the control room operators and will remain habitable in accordance with Three Mile Island (TMI) Task Action Plan Item III.D.3.4 (NUREG-0737, 'Clarification of TMI Action Plan Requirements').

Potential Impact Number: 25840 **SRP Section Number:** 9.4.2

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The staff considers EPRI's proposed resolution of Issue 83 to be complete, subject to the staff's plant-specific reviews, as applicable, of individual applications for FDA/DC. During its reviews, the staff will ensure that the existing minimum requirements have been met, by conforming to the guidance of SRP Sections 6.4 and 9.4.1 (to meet GDC 19). Appropriate requirements are identified in the Evolutionary Requirements Document that include conformance with the applicable regulatory guides (RGs 1.52, 1.78, and 1.95) and industry standards (ANSI/ANS-59.2, ANSI/ ASME N509, and ANSI/ASME N510). However, during its plant-specific reviews, the staff will also verify conformance with industry standards (ASME Code AG-1, 'Code on Nuclear Air and Gas Treatment,' and American Society for Testing and Materials (ASTM) D3803, 'Standard Test Methods for Radiological Testing of Nuclear-Grade Gas-Phase Absorbents') and evaluation methodology and verify that technical specifications and surveillance requirements for the control room habitability system as a whole to meet GDC 4, 5, and 19 have been proposed. This will demonstrate that the control room will adequately protect the control room operators and will remain habitable in accordance with TMI

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Task Action Plan Item III.D.3.4 (NUREG- 0737). Additionally, the FDA/DC applicant will have to ensure that the control room habitability design meets GDC 4, 5, and 19 and that operators are protected in accordance with TMI Action Plan Item III.D. 3.4.

EPRI states that the control room HVAC system and other specific areas will be designed in accordance with National Fire Protection Association (NFPA) 90A, 'Installation of Air Conditioning and Ventilating Systems.' As stated in Section 3.3.1 of Chapter 9, by committing to NFPA 90A, EPRI has committed to use the HVAC systems for removing smoke from specific areas as a means of satisfying the smoke control provisions of NRC fire protection guidelines. The staff will evaluate the design, installation, and operation of the HVAC systems functioning in the smoke removal mode during its review of individual applications for FDA/DC.

Potential Impact Number: 25841 **SRP Section Number:** 9.4.3

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue was raised in response to a 1982 ACRS letter which outlined deficiencies in the design, installation, maintenance and testing of control room systems for maintaining operator habitability. The solution to this issue 'has been identified.'

Potential Impact Number: 25842 **SRP Section Number:** 9.4.3

Document: DRAFT SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

During its plant-specific reviews (of proposed resolution of GSI 83, Control Room Habitability), the staff will verify conformance with the applicable RGs 1.52, 1. 78, and 1.95; industry standards [ANSI/ANS 59.2, ANSI/ASME N509, ANSI/ASME N510, ASME Code AG-1, 'Code of Nuclear Air and Gas Treatment,' and ASTM D3803, 'Standard Test Methods for Radiological Testing of Nuclear- Grade Gas-Phase Absorbents']; and evaluation methodology, and will verify that technical specifications and surveillance

requirements for the control room habitability system, as a whole, to meet GDC 4, 5, and 19 have been proposed. This will demonstrate that the control room will adequately protect the control room operators and will remain habitable in accordance with Three Mile Island (TMI) Task Action Plan Item III.D.3.4 (NUREG-0737, 'Clarification of TMI Action Plan Requirements').

Potential Impact Number: 25843 **SRP Section Number:** 9.4.3

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The staff considers EPRI's proposed resolution of Issue 83 to be complete, subject to the staff's plant-specific reviews, as applicable, of individual applications for FDA/DC. During its reviews, the staff will ensure that the existing minimum requirements have been met, by conforming to the guidance of SRP Sections 6.4 and 9.4.1 (to meet GDC 19). Appropriate requirements are identified in the Evolutionary Requirements Document that include conformance with the applicable regulatory guides (RGs 1.52, 1.78, and 1.95) and industry standards (ANSI/ANS-59.2, ANSI/ ASME N509, and ANSI/ASME N510). However, during its plant-specific reviews, the staff will also verify conformance with industry standards (ASME Code AG-1, 'Code on Nuclear Air and Gas Treatment,' and American Society for Testing and Materials (ASTM) D3803, 'Standard Test Methods for Radiological Testing of Nuclear-Grade Gas-Phase Absorbents') and evaluation methodology and verify that technical specifications and surveillance requirements for the control room habitability system as a whole to meet GDC 4, 5, and 19 have been proposed. This 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 (NUREG- 0737). Additionally, the FDA/DC applicant will have to ensure that the control room habitability design meets GDC 4, 5, and 19 and that operators are protected in accordance with TMI Action Plan Item III.D. 3.4.

EPRI states that the control room HVAC system and other specific areas will be designed in accordance with National Fire Protection Association (NFPA) 90A, 'Installation of Air Conditioning and Ventilating Systems.' As stated in Section 3.3.1 of Chapter 9, by committing to NFPA 90A, EPRI has committed to use the HVAC systems for removing smoke from specific areas as a means of satisfying the smoke control provisions of NRC fire protection guidelines. The staff will evaluate the design,

installation, and operation of the HVAC systems functioning in the smoke removal mode during its review of individual applications for FDA/DC.

Potential Impact Number: 25844 **SRP Section Number:** 3.6.1

Document: DRAFT SER CE80 CH 2
Title: SITE ENVELOPE CHARACTERISTICS

Summary:

This potential impact is a duplicate of PI 22060.

10 CFR Part 100 Appendix A and SRP 2.5.2 both require the magnitude of the OBE be at least one-half the SSE. Industry experience has shown such an OBE controls the design of some safety systems. The staff has agreed in SECY 90-016 Section IV.A that the OBE should not control safety system design.

The CE80 SSE is anchored at 0.30g per RG 1. 60, however, the OBE is anchored at 0.10g. CE DSER Section 2.5.2.7 accepts this condition based on SECY-90-016. Specifically, "Since the Commission has approved decoupling of the OBE from the SSE for evolutionary reactors, per SECY-90-016, the proposed OBE for the ABB-CE System 80+ is acceptable."

Potential Impact Number: 25845 **SRP Section Number:** 3.6.1

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

This potential impact is a duplicate of PI 23920.

EPRI proposed to eliminate the OBE as a design basis event and to use alternative criteria for dynamic analysis methods.

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Potential Impact Number: 25848 **SRP Section Number:** 9.4.4

Document: FINAL SER EPRI CH 1
Title: EPRI URD-EPD: OVERALL REQUIREMENTS

Summary:

The staff considers EPRI's proposed resolution of Issue 83 to be complete, subject to the staff's plant-specific reviews, as applicable, of individual applications for FDA/DC. During its reviews, the staff will ensure that the existing minimum requirements have been met, by conforming to the guidance of SRP Sections 6.4 and 9.4.1 (to meet GDC 19). Appropriate requirements are identified in the Evolutionary Requirements Document that include conformance with the applicable regulatory guides (RGs 1.52, 1.78, and 1.95) and industry standards (ANSI/ANS-59.2, ANSI/ ASME N509, and ANSI/ASME N510). However, during its plant-specific reviews, the staff will also verify conformance with industry standards (ASME Code AG-1, 'Code on Nuclear Air and Gas Treatment,' and American Society for Testing and Materials (ASTM) D3803, 'Standard Test Methods for Radiological Testing of Nuclear-Grade Gas-Phase Absorbents') and evaluation methodology and verify that technical specifications and surveillance requirements for the control room habitability system as a whole to meet GDC 4, 5, and 19 have been proposed. This 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 (NUREG- 0737). Additionally, the FDA/DC applicant will have to ensure that the control room habitability design meets GDC 4, 5, and 19 and that operators are protected in accordance with TMI Action Plan Item III.D. 3.4.

EPRI states that the control room HVAC system and other specific areas will be designed in accordance with National Fire Protection Association (NFPA) 90A, 'Installation of Air Conditioning and Ventilating Systems.' As stated in Section 3.3.1 of Chapter 9, by committing to NFPA 90A, EPRI has committed to use the HVAC systems for removing smoke from specific areas as a means of satisfying the smoke control provisions of NRC fire protection guidelines. The staff will evaluate the design, installation, and operation of the HVAC systems functioning in the smoke removal mode during its review of individual applications for FDA/DC.

Potential Impact Number: 25849 **SRP Section Number:** 9.4.4

Document: DRAFT SER CE80 CH 20
Title: GENERIC ISSUES

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Summary:

During its plant-specific reviews (of proposed resolution of GSI 83, Control Room Habitability), the staff will verify conformance with the applicable RGs 1.52, 1. 78, and 1.95; industry standards [ANSI/ANS 59.2, ANSI/ASME N509, ANSI/ASME N510, ASME Code AG-1, 'Code of Nuclear Air and Gas Treatment,' and ASTM D3803, 'Standard Test Methods for Radiological Testing of Nuclear- Grade Gas-Phase Absorbents']; and evaluation methodology, and will verify that technical specifications and surveillance requirements for the control room habitability system, as a whole, to meet GDC 4, 5, and 19 have been proposed. This will demonstrate that the control room will adequately protect the control room operators and will remain habitable in accordance with Three Mile Island (TMI) Task Action Plan Item III.D.3.4 (NUREG-0737, 'Clarification of TMI Action Plan Requirements').

Potential Impact Number: 25850 **SRP Section Number:** 9.4.4

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This issue was raised in response to a 1982 ACRS letter which outlined deficiencies in the design, installation, maintenance and testing of control room systems for maintaining operator habitability. The solution to this issue 'has been identified.'

Potential Impact Number: 25852 **SRP Section Number:** 11.4

Document: ADVANCE SER ABWR CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

The staff in their evaluation of System 80+, NUREG-1462, "Advanced Final Safety Evaluation Report (FSAR) Related to the Design Certification of Combustion Engineering (CE) System 80+," February 1994, and the ABWR, NUREG-1503, "Final Safety Evaluation Report Related to the Advanced Boiling Water Reactor Design," July

1994, invoked the provisions of Generic Letter (GL) 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites," November 11, 1981 as acceptable review criteria for the Solid Waste Management Systems.

Potential Impact Number: 25853 **SRP Section Number:** 11.4

Document: ADVANCE SER CE CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

The staff in their evaluation of System 80+, NUREG-1462, "Advanced Final Safety Evaluation Report (FSAR) Related to the Design Certification of Combustion Engineering (CE) System 80+," February 1994, and the ABWR, NUREG-1503, "Final Safety Evaluation Report Related to the Advanced Boiling Water Reactor Design," July 1994, invoked the provisions of Generic Letter (GL) 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites," November 11, 1981 as acceptable review criteria for the Solid Waste Management Systems.

Potential Impact Number: 25855 **SRP Section Number:** 11.3

Document: ADVANCE SER CE CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

The staff in its evaluation of System 80+, NUREG-1462, "Advanced Final Safety Evaluation Report (FSAR) Related to the Design Certification of Combustion Engineering (CE) System 80+," February 1994, and the ABWR, NUREG-1503, "Final Safety Evaluation Report Related to the Advanced Boiling Water Reactor Design," July 1994, invoked Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," as review criteria. This BTP 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

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 Impact Number: 25856 **SRP Section Number:** 11.3

Document: ADVANCE SER ABWR CH 11
Title: RADIOACTIVE WASTE MANAGEMENT

Summary:

The staff in its evaluation of System 80+, NUREG-1462, "Advanced Final Safety Evaluation Report (FSAR) Related to the Design Certification of Combustion Engineering (CE) System 80+," February 1994, and the ABWR, NUREG-1503, "Final Safety Evaluation Report Related to the Advanced Boiling Water Reactor Design," July 1994, invoked Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," as review criteria. This BTP 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 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 Impact Number: 25859 **SRP Section Number:** 3.13

Document: NRC GENERIC LETTER 91-17 Revision Date: 10/17/91
Title: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

This potential impact was duplicated from PI#21826.

Generic letter 91-17 provides information on the U.S. Nuclear Regulatory Commission's resolution of Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power

Plants." The NRC has resolved this issue without developing any new requirements. The bases for resolution are further documented in the enclosed NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants".

NUREG-1339 recommends that a new section of the SRP be prepared to provide guidance to the staff for the review of future plants. It is recognized that NUREG-1339 makes an exception for closure studs which are already addressed in SRP sections 5.3.1 and 5.2.3, however, the staff noted the absence of an SRP section on general reviews of bolting and recommended that one be prepared and issued.

Potential Impact Number: 25860 **SRP Section Number:** 3.13

Document: NRC GENERIC LETTER 91-17 Revision Date: 10/17/91
Title: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

This potential impact was duplicated from PI#21671.

Generic letter 91-17 provides information on the U.S. Nuclear Regulatory Commission's resolution of Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants." The NRC has resolved this issue without developing any new requirements. The bases for resolution are further documented in the enclosed NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants".

NUREG-1339 recommends that a new section of the SRP be prepared to provide guidance to the staff for the review of future plants. It is recognized that NUREG-1339 makes an exception for closure studs which are already addressed in SRP sections 5.3.1 and 5.2.3, however, the staff noted the absence of an SRP section on general reviews of bolting and recommended that one be prepared and issued.

Potential Impact Number: 25861 **SRP Section Number:** 3.13

Document: NRC BULLETIN 82-02

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Title: DEGRADATION OF THREADED FASTENERS IN THE REACTOR COOLANT PRESSURE BOUNDARY OF PWR PLANTS

Summary:

This potential impact was duplicated from PI#19185.

The purpose of this bulletin is to: (1) notify licensees and construction permit holders about incidents of severe degradation of threaded fasteners (bolts and studs) in closures in the reactor coolant pressure boundary (RCPB), and (2) to require appropriate actions.

Potential Impact Number: 25862 **SRP Section Number:** 3.13

Document: NUREG 1339
Title: RESOLUTION OF GENERIC SAFETY ISSUE 29: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

This potential impact was duplicated from PI#23558.

In NUREG-1339, the staff noted the absence of an SRP section on general review of bolting and recommended that one be prepared and issued. Such a section, entitled, for example, "Safety-Related Bolting" would expand the limited coverage on fasteners now included in the SRP and provide a systematic method for implementation of the staff position regarding the basis for resolution of GSI 29.

Potential Impact Number: 25863 **SRP Section Number:** 3.13

Document: NRC GENERIC LETTER 91-17 Revision Date: 10/17/91
Title: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

This potential impact was duplicated from PI#23561.

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Generic letter 91-17 provides information on the U.S. Nuclear Regulatory Commission's resolution of Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants." The NRC has resolved this issue without developing any new requirements. The bases for resolution are further documented in the enclosed NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants".

NUREG-1339 recommends that a new section of the SRP be prepared to provide guidance to the staff for the review of future plants. It is recognized that NUREG-1339 makes an exception for closure studs which are already addressed in SRP sections 5.3.1 and 5.2.3, however, the staff noted the absence of an SRP section on general reviews of bolting and recommended that one be prepared and issued.

Potential Impact Number: 25864 **SRP Section Number:** 3.13

Document: NUREG 1339
Title: RESOLUTION OF GENERIC SAFETY ISSUE 29: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

This potential impact was duplicated from PI#23563.

In NUREG-1339, the staff noted the absence of an SRP section on general review of bolting and recommended that one be prepared and issued. Such a section, entitled, for example, "Safety-Related Bolting" would expand the limited coverage on fasteners now included in the SRP and provide a systematic method for implementation of the staff position regarding the basis for resolution of GSI 29.

Potential Impact Number: 25865 **SRP Section Number:** 3.13

Document: NRC GENERIC LETTER 91-17 Revision Date: 10/17/91
Title: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

This potential impact was duplicated from PI#23767.

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Generic letter 91-17 provides information on the U.S. Nuclear Regulatory Commission's resolution of Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants." The NRC has resolved this issue without developing any new requirements. The bases for resolution are further documented in the enclosed NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants".

Potential Impact Number: 25866 **SRP Section Number:** 3.13

Document: NUREG 1339
Title: RESOLUTION OF GENERIC SAFETY ISSUE 29: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

Summary:

This potential impact was duplicated from PI#23771.

In NUREG-1339, the staff noted the absence of an SRP section on general review of bolting and recommended that one be prepared and issued. Such a section, entitled, for example, "Safety-Related Bolting" would expand the limited coverage on fasteners now included in the SRP and provide a systematic method for implementation of the staff position regarding the basis for resolution of GSI 29.

NUREG-1339 was transmitted by Generic Letter 91-17. Microfiche 54927:158.

Potential Impact Number: 25867 **SRP Section Number:** 3.13

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.55a

Summary:

10 CFR 50.55a specifies industry codes and standards to be used for components of the reactor coolant pressure boundary and other classes components as defined in Regulatory Guide 1.26 and the ASME Code.

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Potential Impact Number: 25868 **SRP Section Number:** 3.13

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Locations: 50.60, 50.60, Appendix G

Summary:

10 CFR 50.60 and Appendix G provide fracture toughness requirements for specific materials and components within the reactor coolant pressure boundary of light water reactors. Threaded fasteners are specifically included.

Potential Impact Number: 25869 **SRP Section Number:** 3.13

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, General Design Criterion (GDC)

Summary:

GDC 1 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;

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.

Potential Impact Number: 25870 **SRP Section Number:** 3.13

Document: CODE OF FED. REGS 10CFR50

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Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 2

Summary:

GDC 2 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.

Potential Impact Number: 25871 **SRP Section Number:** 3.13

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 4

Summary:

GDC 4 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).

Potential Impact Number: 25872 **SRP Section Number:** 3.13

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 14

Summary:

GDC 14 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.

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Potential Impact Number: 25873 **SRP Section Number:** 3.13

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 15

Summary:

GDC 15 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.

Potential Impact Number: 25874 **SRP Section Number:** 3.13

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 30

Summary:

GDC 30 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.

Potential Impact Number: 25875 **SRP Section Number:** 3.13

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 31

Summary:

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GDC 31 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 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.

Potential Impact Number: 25876 **SRP Section Number:** 3.13

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 32

Summary:

GDC 32 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.

Potential Impact Number: 25877 **SRP Section Number:** 4.6

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Document: FINAL SER ABWR CH 19
Title: SEVERE ACCIDENTS

Summary:

The ABWR design includes a non-safety-related feedwater and condensate system, consisting of three electric pumps and associated piping, that can be used as an alternate means for make-up during shutdown operation. The CRD pump can also be used to provide inventory control during shutdown by injecting water from the condensate storage tank to the RPV through the FMCRD system. An ACIWA system is also available to supply make-up water to the RPV if no ECCS make-up water is available.

The staff finds these provisions acceptable and concludes that GE has sufficiently addressed the concerns in NUREG-1449 related to alternate make-up capability to provide core decay heat removal. The alternate inventory control features using the feedwater, the condensate system, and the CRD pump will provide alternate core cooling upon loss of normal RHR capability. The staff also finds that an ACIWA system will further enhance the capability of the ABWR to maintain core cooling in the event that no ECCS make-up is available.

Potential Impact Number: 25880 **SRP Section Number:** 3.6.1

Document: C&S: ASME III
Title: CLASS 1 AND 2 SYSTEMS

Summary:

Appendix B to BTP SPLB 3-1 specifies that SA 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.

Potential Impact Number: 25883 **SRP Section Number:** 2.5.2

Document: CODE OF FED. REGS 10CFR52

APPENDIX II
Potential Impacts

Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 10 CFR 52.17 (a)(1)

Summary:

The application (for an early site permit) must contain the information required by 10 CFR 50.33 (a)-(d), the first three sentences of 50.34(a)(1), and, to the extent approval of emergency plans is sought under paragraph (b)(2)(ii) of this section, the information required by 50.33 (g) and (j), and 50.34(b)(6)(v). In particular, the application should describe the following: ... (vi) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site (see Appendix A to 10 CFR Part 100)...

Potential Impact Number: 25884 **SRP Section Number:** 2.5.2

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: 10 CFR 52.79(b)

Summary:

An application referencing certified design must describe those portions of the design which are site-specific, such as the service water intake structure and the ultimate heat sink. An application referencing a certified design must also demonstrate compliance with the interface requirements established for the design.

Potential Impact Number: 25885 **SRP Section Number:** 5.4.2.2

Document: FINAL SER CE80 CH 16
Title: TECHNICAL SPECIFICATIONS

Summary:

The staff review of the System 80+ technical specifications (TS) was closely coupled to the development of the improved standard technical specifications (STS) under the TS

Improvement Program in accordance with the Commission Policy Statement on TS Improvement 58 FR 39132 (SECY-93-067). Since the System 80+ design evolved primarily from the System 80 design, most of the System 80+ TS were modeled after NUREG-1432, "Standard Technical Specifications for Combustion Engineering Plants" (January 1991). These improved STS contain the benefits of the accumulated operating experience from currently operating light-water reactors.

The System 80+ design, however, has some major changes from the System 80 design, particularly in the containment, which consists of a much larger free containment volume to account for severe accident concerns. Other specific changes include the reactor cavity and flooding system, the in-containment refueling water storage tank (IRWST), the Nuplex 80+ advanced control complex, the hydrogen control system, the addition of a safety depressurization system and the addition of a combustion turbine generator (CTG). The System 80+ TS address these changes and the issue of shutdown risk considerations. On this basis, the staff's review of the System 80+ TS concentrated on the differences from the System 80 STS (NUREG-1432).

The staff forwarded to ABB-CE, its comments from the System 80+ proof and review of TS, including Amendments U and V, for resolution and incorporation into the final TS. The final TS were produced in the industry format and certified as accurate by ABB-CE in Amendment W to the CESSAR-DC.

The staff evaluated the System 80+ TS to confirm that they will preserve the validity of the design plant as described in the CESSAR-DC by assuring that System 80+ plants will be operated (1) within the required conditions bounded by the CESSAR-DC, and (2) with operable equipment that is essential to prevent accidents and to mitigate the consequences of accidents postulated in the CESSAR-DC.

On the basis of its review of the System 80+ TS, as discussed above, the staff concludes that the System 80+ TS are consistent with the regulatory guidance contained in the System 80 STS (NUREG-1432) and contain design-specific parameters and additional TS requirements considered appropriate by the staff. Therefore, the staff concludes that the System 80+ TS satisfy 10 CFR 50.34 and 10 CFR 50.36.

Potential Impact Number: 25886 **SRP Section Number:** 15.1.1 - 15.1.4

Document: NRC GENERIC LETTER 80-91

APPENDIX II
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Title: USE OF THE ODYN CODE TO CALCULATE PRESSURIZATION TRANSIENTS

Summary:

The use of the ODYN code to calculate pressurization transients has been reviewed extensively by the staff and discussed with the General Electric Company. The staff found that ODYN provides acceptable best estimate calculation predictions of the core responses to pressurization transients.This letter advises licensees and applicants of BWRs of the staffs requirements for implementation of ODYN for licensing basis calculations performed by the General Electric Company. These requirements are applicable to license applications and all proposed license amendments, including core reloads for which analyses are provided by General Electric.

Transient analyses performed by General Electric supporting reload submittals received prior to February 1, 1981, will be reviewed taking into account the results of recent generic transient analyses with ODYN. Appropriate CPR penalties will be applied on a case by case basis. Transient analyses performed by General Electric 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. Generally, these will include generator load rejection/turbine trip without bypass (whichever is limiting), feedwater controller failure - maximum demand, and main steamline isolation valve closure (to satisfy ASME code pressure requirements). After January 1982, all operating BWRs with General Electric analyses must have the limiting transients recalculated with the ODYN code, even if no reload submittal has been received.

Potential Impact Number: 25887 **SRP Section Number:** 2.5.4

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

Appendix A to 10 CFR Part 100 requires, in part, that all structures, systems, and components (SSCs) of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public shall be designed to remain functional and within applicable stress and deformation limits when subject to an OBE. Changes to Appendix A to 10 CFR Part 100 are being proposed to redefine the OBE

such that the OBE criterion can be satisfied without the need to perform explicit design and response analyses, if the OBE is equal to or less than one-third of the safe-shutdown earthquake (SSE). The proposed amendment to 10 CFR Part 100, Appendix A, would also allow, as an option, that the OBE be eliminated from design certification when the OBE is established at less than or equal to one-third of the SSE. In this manner, the OBE serves the 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. In addition, the staff notes that Appendix A to 10 CFR Part 100 requires that the maximum vibratory ground acceleration of the OBE be at least one-half the maximum vibratory ground acceleration of the SSE. Therefore, deviations from this regulation must be evaluated to ensure that 10 CFR 50.12 exemption criteria are satisfied when the OBE is eliminated from design analysis and is redefined only as an inspection level earthquake with the maximum vibratory ground acceleration at one-third of that of the SSE.

Special circumstances exist for granting these exemptions from the requirements of Appendix A to Part 100 pursuant to 10 CFR 50.12. The purpose of designing SSCs necessary for continued operation without undue risk to the health and safety of the public to withstand an OBE is to ensure that these SSCs remain functional and within applicable stress and deformation limits when subjected to the effects of the OBE vibratory ground motion. However, Appendix A to Part 100 also requires that these SSCs be designed to withstand the SSE and remain functional. Thus, when these SSCs are designed to remain functional for the SSE, they will also remain functional at a lesser earthquake level (one-third the SSE) provided that all design functions of the OBE are accounted for. The basis for selecting one-third the SSE as the earthquake level at which the plant will be required to shut down and be inspected for damage was that at this level the likelihood of damage and the frequency of earthquakes occurring was judged to be low based on actual earthquake experience. It should be noted that certain design functions had been verified only for the OBE loads in the past. These design functions were the evaluations of (1) fatigue damage caused by earthquake cycles and (2) relative seismic anchor motions (SAMs) in piping systems. With the elimination of the OBE from design, these design functions would not explicitly have been verified. Consequently, for the System 80+ these design functions will be verified in conjunction with the SSE using applicable stress and deformation limits as described below.

Potential Impact Number: 25888 **SRP Section Number:** 2.5.4

APPENDIX II
Potential Impacts

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

Appendix A to 10 CFR Part 100 requires, in part, that all SSCs of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public shall be designed to remain functional and within applicable stress and deformation limits when subject to an OBE. The NRC is proposing changes to Appendix A to Part 100 to redefine the OBE to such a level such that the function of the OBE can be satisfied without performing explicit response analyses. In addition, Appendix A to Part 100 requires that the maximum vibratory ground acceleration of the OBE be at least one-half the maximum vibratory ground acceleration of the SSE. When the OBE is redefined to an inspection level earthquake for the ABWR, the maximum vibratory ground acceleration of the OBE will be established at one-third of the maximum vibratory ground acceleration of the SSE.

There exist special circumstances for granting these exemptions from the requirements of Appendix A to Part 100 pursuant to 10 CFR 50.12. The purpose of designing SSCs necessary for continued operation without undue risk to the health and safety of the public to withstand an OBE is to ensure that these SSCs remain functional and within applicable stress and deformation limits when subjected to the effects of the OBE vibratory ground motion. However, Appendix A to Part 100 also requires that these SSCs are designed to withstand the SSE and remain functional. Thus, when these SSCs are designed to remain functional for the SSE, they will also remain functional at a lesser earth -quake level (one-third the SSE) provided all design functions at the OBE are accounted for. The basis for selecting one-third the SSE as the earthquake level at which the plant will be required to shutdown and be inspected for damage was that at this level the likelihood of damage and the frequency of earthquakes occurring was judged to be low based on actual earthquake experience. It should be noted that certain design functions had been only verified for the OBE loads in the past. These design functions were the evaluations of (1) fatigue damage caused by earthquake cycles and (2) relative seismic anchor motions in piping systems. With the elimination of the OBE from design, these design functions would not have been explicitly verified. Consequently, for the ABWR these design functions will be verified in conjunction with the SSE using applicable stress and deformation limits as described in Section 3.1.1.2 of this report.

Potential Impact Number: 25889 **SRP Section Number:** 6.2.1

APPENDIX II
Potential Impacts

Document: FINAL SER CE80 CH 20
Title: GENERIC ISSUES

Summary:

The CE 80+ FSER section 19.3 contains staff reviews and guidance on shutdown risk assessments.

Potential Impact Number: 25890 **SRP Section Number:** 15.2.7

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

NRC staff approved analysis methods for transient and accident analyses are discussed in Section 15.1 of the System 80+ FSER.

Potential Impact Number: 25891 **SRP Section Number:** 15.2.7

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

Tables 15.3 and 15.4 of the System 80+ FSER lists topical reports and letters and NRC approval letters for analyses methods and computer codes used in transient and accident analyses.

Potential Impact Number: 25892 **SRP Section Number:** 15.2.7

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

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Potential Impacts

Summary:

The staff's review of acceptability of the ODYNA and REDYA computer codes for ABWR transient analyses is documented in Section 15.1 of the ABWR FSER.

Potential Impact Number: 25893 **SRP Section Number:** 15.2.7

Document: FINAL SER ABWR CH 15
Title: TRANSIENT AND ACCIDENT ANALYSIS

Summary:

The staff's review of the ODYNA and REDYA computer codes is discussed in Section 15.1 of the ABWR FSER.

Potential Impact Number: 25898 **SRP Section Number:** 3.8.2

Document: FINAL SER ABWR APPENDIX E
Title: STAFF POSITION ON SHELL BUCKLING DUE TO INTERNAL PRESSURE

Summary:

The Staff's evaluation of steel components of the containment design is described in Section 3.8.1 of the ABWR FSER and includes a discussion of the ABWR design with regard to buckling of the drywell head. Section 3.8.1 of the FSER refers to Appendix E of the FSER for the Staff's position with regard to shell buckling due to internal pressure.

Potential Impact Number: 25899 **SRP Section Number:** 2.1.1

Document: CODE OF FED. REGS 10CFR20
Title: STANDARDS FOR PROTECTION AGAINST RADIATION
Location: Subpart D

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Potential Impacts

Summary:

10 CFR 20.1301 effectively places limits on the annual average releases in gaseous and liquid effluents at the boundary of the restricted area by placing limits on the exposure an individual would receive if he were continually present at the boundary of the restricted area.

10 CFR 20.1003 defines restricted area.

Potential Impact Number: 25900 **SRP Section Number:** 3.4.2

Document: C&S: US ARMY SPM-1977
Title: SHORE PROTECTION MANUAL

Summary:

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 Impact Number: 25901 **SRP Section Number:** 3.6.1

Document: C&S: ASME III
Title: CLASS 1 AND 2 SYSTEMS

Summary:

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.

Potential Impact Number: 25902 **SRP Section Number:** 13.3

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Potential Impacts

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.47

Summary:

10 CFR 50.47(d) sets forth specific provisions which would allow a license to be issued for fuel loading and low power testing without all of the requirements of 10 CFR 50.47(a) and (b) having been met.

Potential Impact Number: 25903 **SRP Section Number:** 13.3

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Locations: 52.17, 52.17, 52.79, Appendix O

Summary:

10 CFR 52.17, et seq., contains specific provisions regarding emergency planning requirements for an early site permit applicant. 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.

Potential Impact Number: 25904 **SRP Section Number:** 13.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.33 and 50.47(a)

Summary:

10 CFR 50.33 specifies that an applicant for an operating license shall submit radiological emergency response plans of State and local governmental entities in the

U.S. that are wholly or partially within the plume exposure pathway Emergency Planning Zones (EPZs). A footnote presented in 10 CFR 50.33 states that EPZs are discussed in NUREG-0396 (EPA 520/1-78-016), "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," December 1978.

10 CFR 50.47(a) 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(a) as an acceptance criterion will provide a clearer basis for establishing compliance with the ensuing regulations.

Potential Impact Number: 25906 **SRP Section Number:** 3.8.5

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The staff's review of the controlling load combinations finds that, in general, the load combinations with the SSE control the design of steel structures although there may be specific cases where the load combinations with the OBE control the design. Similarly, an examination of the pertinent load combinations for concrete structures, including the containment structure, should lead to the same conclusion that the OBE loads, in most cases, do not control the outcome of the structural design.

In the design of the containment, the staff reviewed the extent to which the elimination of the OBE from the load combinations would lead to a reduction of the safety margin. An examination of the nuclear structural design practice and the SRP load combination equations, however, show that the major dynamic load for the overall design of structures is either the OBE or the SSE. All other potential dynamic loads are conservatively accounted for in the definition of equivalent dead and live loads or only produce local effects that are handled by local reinforcement details. Therefore, the staff concludes that no reduction in safety margins of concrete and steel structures results from the elimination of the OBE as a design requirement.

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Potential Impact Number: 25907 **SRP Section Number:** 3.8.5

Document: FINAL SER CE80 CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Summary:

The staff's review of the controlling load combinations finds that, in general, the load combinations with the SSE control the design of steel structures although there may be specific cases where the load combinations with the OBE control. Similarly, an examination of the pertinent load combinations for concrete structures should lead to the same conclusion that the OBE loads, in most cases, do not control the outcome of the structural design.

All other potential dynamic loads are conservatively accounted for in the definition of equivalent dead and live loads, except those that produce local effects that are handled by local reinforcement details. Therefore, it can be concluded that no reduction in safety margins of concrete and steel structures results from the elimination of the OBE as a design requirement.

Potential Impact Number: 25908 **SRP Section Number:** 3.8.5

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

Presently, 10 CFR 100 requires that the magnitude of the OBE be at least one-half that of the SSE. It has been an industry wide experience that such a requirement leads to a design that is governed by the OBE requirements and produces unnecessary and inconsistent margins for the SSE loading. This requirement was included in the regulation when the staff did not have substantial experience with the seismic resistance of plants that incorporated OBE design at half the SSE value. Since then a number of research programs have been conducted including a large industry effort on testing and observation of actual earthquake experience of industrial facilities; consequently, the NRC funded Piping Review Committee has concluded that the OBE at existing plants are too high, therefore, controlling the design of some safety systems,

and recommended that the OBE be decoupled from the SSE. Certain interim measures, such as allowing somewhat higher damping values for piping analysis, have been taken to partially implement the Piping Review Committee recommendations (NUREG 1061, 1984). But the complete implementation of the recommendations would involve a revision of 10 CFR 100, Appendix A. Because of higher priority work, the effort on revision of this regulation has been postponed. It should be noted that the Commission has, in certain site specific cases, previously approved OBEs of less than one-half the SSE.

EPRI has requested that NRC regulations be changed to reduce the magnitude of the OBE relative to the SSE as a basis for the design. All evolutionary ALWR vendors agree with the request. GE has stated that they agree with EPRI in principle, however, the ABWR design uses an OBE that is one-half the SSE; therefore, this is a non-issue for the ABWR. EPRI has identified this as an optimization issue.

The staff agrees that the OBE should not control the design of safety systems. However, a staff position on this issue to be applied generically to all future designs has not yet been fully developed. For the evolutionary reactors, the staff will consider requests to decouple the OBE from the SSE on a design-specific basis. Such a decoupling would require an exemption to the Commission's regulations, therefore the staff recommends that the Commission approve this design-specific relief approach.

Potential Impact Number: 25909 **SRP Section Number:** 13.5.1.1

Document: NRC GENERIC LETTER 90-03
Title: RELAXATION OF STAFF POSITION IN GENERIC LETTER 83-28, ITEM 2.2 PART 2 'VENDOR INTERFACE FOR SAFETY-RELATED COMPONENTS'

Summary:

The staff has reviewed the Vendor Equipment Technical Information Program (VETIP) program and has had the benefit of considerable discussions with licensees regarding vendor interface. It is now recognized that implementing a formal vendor interface program for every safety-related component is not practical. It is also recognized that vendors may not always be In the best position to analyze a failure because they may not be aware of the components' application, environment or maintenance history. Therefore, we conclude that the elements of VETIP provide a framework to improve the quality and availability of equipment technical information for use by utility licensees.

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Experience has shown that many vendors, in addition to NSSS vendors, do produce valuable information relating to their equipment. For this reason, the staff concludes that an adequate vendor interface program should include:

(a) A program with the NSSS vendor as described in the VETIP, which covers all the safety-related components within the NSSS scope of supply. This program should include provisions for assuring receipt by the licensee/applicant of all technical information provided by the NSSS vendor; and

(b) A program of periodic contact with the vendors of other key related components not included in (a) above.

The vendor interface program should also take into account the requirements of 10 CFR Part 50, Appendix B which requires the licensee or applicant to be responsible for establishing and executing the quality assurance program.

Potential Impact Number: 25910 **SRP Section Number:** 15.0

Document: CODE OF FED. REGS 10CFR100
Title: REACTOR SITE CRITERIA
Location: 10 CFR 100.11

Summary:

10 CFR 100.11 contains regulations on the determination of exclusion area, low population zone, and population center distance including addressing the fission product release (source term) used in the supporting calculations.

Potential Impact Number: 25911 **SRP Section Number:** 15.0

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.34(f)(2)

Summary:

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10 CFR 50.34(f)(2) references the use of the TID-14844 source term document throughout its subsections.

Potential Impact Number: 25912 **SRP Section Number:** 15.0

Document: FINAL SER CE80 CH 15
Title: TRANSIENT AND ACCIDENT ANALYSES

Summary:

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.

Potential Impact Number: 25913 **SRP Section Number:** 15.0

Document: SECY 90-016
Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

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.

Potential Impact Number: 25914 **SRP Section Number:** 15.0

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

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Summary:

SECY 93-087 contains additional information on the use by future plant applicants of a physically based source term as discussed in NUREG-1465.

Potential Impact Number: 25915 **SRP Section Number:** 13.5.2.1

Document: NUREG 1358
Title: LESSONS LEARNED FROM THE SPECIAL INSPECTION PROGRAM FOR EMERGENCY OPERATING PROCEDURES

Summary:

The results of the team inspections conducted during 1988 for the Nuclear Regulatory Commission (NRC) Emergency Operating Procedures (EOPs) Inspection Program are summarized in this report. Inspection results are discussed in relation to past staff guidance provided to the utilities for upgrading EOPs. In this report the NRC informs the nuclear power industry of concerns that were identified during the EOP team inspections and restates the staff's previous guidance in a format that more clearly illustrates the consequences of programmatic weaknesses. The information contained in this report is provided so that licensees may take advantage of the lessons learned from the EOP inspection effort. This report does not impose any new requirements related to the upgrading of EOPs.

Potential Impact Number: 25916 **SRP Section Number:** 13.5.2.1

Document: NUREG 1358 SUP 1
Title: LESSONS LEARNED FROM THE SPECIAL INSPECTION PROGRAM FOR EMERGENCY OPERATING PROCEDURES

Summary:

The results of the team inspections conducted during October 1988 - September 1991 for the Nuclear Regulatory Commission (NRC) Emergency Operating Procedures (EOPs) Inspection Program are summarized in this report. Inspection results are discussed in relation to past staff guidance provided to the utilities for upgrading EOPs. In this report, the NRC informs the nuclear power industry of concerns that were

identified during the EOP team inspections and restates the staff's previous guidance in a format that more clearly illustrates the consequences of programmatic weaknesses. The information contained in this report is provided so that licensees may take advantage of the lessons learned from the EOP inspection effort. This report does not impose any new requirements related to the upgrading of EOPs.

Potential Impact Number: 25917 **SRP Section Number:** 13.5.2.1

Document: FINAL SER ABWR APPENDIX J
Title: HUMAN FACTORS ENGINEERING PROGRAM REVIEW MODEL AND ACCEPTANCE CRITERIA FOR EVOLUTIONARY REACTORS

Summary:

Appendix J of the ABWR FSER describes the human factors engineering program review model. Element 7 of the HFE PRM addresses plant and emergency operating procedures and states that the procedures shall be developed to support and guide human interaction with plant systems and to control plant-related events and activities. Human engineering principles and criteria shall be applied along with all other design requirements to develop procedures that are technically accurate, comprehensive, explicit, easy to utilize, and validated. Item 7 of Subsection 3.7 lists NUREG-1358 as providing guidance for procedure development.

Potential Impact Number: 25923 **SRP Section Number:** 13.5.2.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.34(f)(2)(ii)

Summary:

Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts.

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Potential Impact Number: 25924 **SRP Section Number:** 13.5.2.2

Document: NUREG 0933
Title: A Prioritization of Generic Safety Issues

Summary:

This task was developed to provide assurance that plant procedures were adequate and could be used effectively. The objective was to provide procedures that would guide operators in maintaining plants in a safe state under all operating conditions, including the ability to control upset conditions without first having to diagnose the specific initiating event. This objective was to be met by: (1) developing guidelines for preparing, and criteria for evaluating, emergency operating procedures (EOPs), normal operating procedures, and other procedures that affect plant safety; and (2) upgrading the procedures, training the operators in their use, and implementing the upgraded procedures.

Potential Impact Number: 25925 **SRP Section Number:** 13.5.2.2

Document: FINAL SER CE80 CH 13
Title: CONDUCT OF OPERATIONS

Summary:

In summary, the staff finds ABB-CE's approach to procedure development as described in CESSAR-DC Section 13.5.1 acceptable since it will be addressed by the COL applicant referencing the System 80+ certified design. The COL applicant will be required to submit site-specific information described in 10 CFR 52.79(b) at the COL stage. COL Action Item 13.5-1 is found to be acceptable.

Potential Impact Number: 25926 **SRP Section Number:** 13.5.2.2

Document: FINAL SER ABWR CH 13
Title: CONDUCT OF OPERATIONS

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Summary:

The staff determined that development of detailed procedures and associated training materials is beyond the scope of the ABWR design certification and is the responsibility of the COL applicant.

Potential Impact Number: 25930 **SRP Section Number:** 13.5.2.2

Document: FINAL SER CE80 CH 13
Title: CONDUCT OF OPERATIONS

Summary:

The staff has reviewed HF-4.4, "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." The staff determined that development of detailed procedures is beyond the scope of the System 80+ certification and is the responsibility of the COL applicant. ABB-CE has included the procedure development process as a COL license information item in CESSAR-DC Section 13.5.1. CESSAR-DC Section 13.5.1 states that the methods and criteria for the development, V&V, implementation, maintenance, and revision of procedures will include considerations of I.C.1, I.C.5, and I.C.9. On this basis, COL Action Items 13.5-1 is acceptable.

Potential Impact Number: 25931 **SRP Section Number:** 13.5.2.2

Document: FINAL SER ABWR CH 13
Title: CONDUCT OF OPERATIONS

Summary:

The staff has reviewed HF-4.4, "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." It was determined that development of detailed procedures is beyond the scope of the ABWR certification and is the responsibility of the COL applicant. GE has included the procedure development process as a COL license information item in

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SSAR Section 13.5.3. SSAR Section 13.5.3.1 states that the methods and criteria for the development, V&V, implementation, maintenance, and revision of procedures will include considerations of I.C.1, I.C.5, and I.C.9. The staff finds GE's approach to be acceptable.

Potential Impact Number: 25932 **SRP Section Number:** 13.5.2.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50, Appendix B, section V

Summary:

Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Potential Impact Number: 25933 **SRP Section Number:** 13.5.2.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix B, criteria VI

Summary:

Measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality. These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed. Changes to documents shall be reviewed and approved by the same organizations that performed the original review and approval unless the applicant designates another responsible organization.

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Potential Impact Number: 25942 **SRP Section Number:** 5.2.4

Document: NUREG 0619
Title: BWR FEEDWATER NOZZLE AND CONTROL ROD DRIVE RETURN LONE NOZZLE CRACKING

Summary:

This report summarizes work performed by the NRC staff in the resolution of Generic Technical Activity A-10, "BWR Nozzle Cracking. " Generic Technical Activity A-100 is one of the generic technical subjects designated as "unresolved safety issues" pursuant to Section 210 of the Energy Reorganization Act of 1974. The report describes the technical issues, the technical studies and analyses performed by the General Electric Company and the NRC staff, the staff's technical positions based on these studies, and the staff's plans for continued implementation of its technical positions.

Potential Impact Number: 25943 **SRP Section Number:** 5.2.4

Document: NRC GENERIC LETTER 80-95
Title: FINAL EDITION OF NUREG-0619, 'BWR FEEDWATER NOZZLE AND CONTROL ROD DRIVE RETURN LINE NOZZLE CRACKING'

Summary:

This letter forwards the revised and final edition of NUREG- 0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking." The report provides the staff's resolution of the NRC's Generic Technical Activity A-10, which is an "Unresolved Safety Issue" pursuant to Section 210 of the Energy Reorganization Act of 1974.

It is expected that requirements contained in the NUREG will be met.

Potential Impact Number: 25944 **SRP Section Number:** 5.2.4

Document: NRC GENERIC LETTER 81-11
Title: BWR FEEDWATER NOZZLE CRACKING

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Summary:

This letter provided corrections and clarifications to NUREG-0619 and the actions expected in response to GL 89-95.

Potential Impact Number: 25945 **SRP Section Number:** 5.4.2.2

Document: NRC GENERIC LETTER 95-05
Title: VOLTAGE-BASED REPAIR CRITERIA FOR WESTINGHOUSE STEAM GENERATOR TUBES AFFECTED BY OUTSIDE DIAMETER STRESS CORROSION CRACKING

Summary:

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to give guidance to licensees who may wish to request a license amendment to the plant technical specifications to implement alternate steam generator tube repair criteria applicable specifically to 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 alloy 600 steam generator tubing.

This generic letter 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.

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Potential Impact Number: 25946 **SRP Section Number:** 6.4

Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

The review criterion for Issue 83, "Control Room Habitability" is to verify that the control room is designed to provide adequate protection to the operating personnel during and following an accident. 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 design must be in accordance with the requirements of GDC 2, 4, and 19 of 10 CFR Part 50, Appendix A, and ASME AG-1, "Code on Nuclear Air and Gas Treatment" (1991) and the ASME AG-1a-92 Addenda, ASME 509, and ASME N510.

SSAR Section 9.4.1.1.7 states that ESF filter trains comply with the design, testing, and maintenance provisions of RG 1.52. SSAR Appendices 9C and 9D provide information on how the ABWR design meets RG 1.52 and SRP Table 6.5.1-1, respectively. The staff's review of this section found that the ABWR conforms with the design, testing, and maintenance provisions of RG 1.52.

Based on this information, the staff finds that the ABWR is adequately designed to ensure the habitability of the control room under normal and accident conditions and meets the requirements of GDC 2 regarding the systems being capable of withstanding the effects of earthquakes; GDC 4 as it relates to maintaining environmental conditions in the control room compatible with the design limits of essential equipment located therein during normal, transient, and accident conditions; and GDC 19 with regard to providing adequate protection to permit access and occupancy of the control room under accident conditions. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

It should be noted that ASME AG-1 is not limited in scope to BWR HVAC systems. In fact, several CE System 80+ HVAC systems are also constructed to ASME AG-1 (see Section 3.9.3.1 of the CE System 80+ FSER and CESSAR-DC Section 9.4, "Air Conditioning, Heating, Cooling and Ventilation Systems").

Potential Impact Number: 25947 **SRP Section Number:** 9.4.1

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Document: FINAL SER ABWR CH 20
Title: GENERIC ISSUES

Summary:

The review criterion for Issue 83, "Control Room Habitability" is to verify that the control room is designed to provide adequate protection to the operating personnel during and following an accident. 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 design must be in accordance with the requirements of GDC 2, 4, and 19 of 10 CFR Part 50, Appendix A, and 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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It should be noted that ASME AG-1 is not limited in scope to BWR HVAC systems. In fact, several CE System 80+ HVAC systems are also constructed to ASME AG-1 (see Section 3.9.3.1 of the CE System 80+ FSER and CESSAR-DC Section 9.4, "Air Conditioning, Heating, Cooling and Ventilation Systems").

Potential Impact Number: 25948 **SRP Section Number:** 16.0

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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Location: 50.36(c)(2)

Summary:

The Commission has amended its regulations regarding technical specifications, specifically at paragraph 10 CFR 50.36(c)(2). A notice of the final rule was published in the Federal Register on July 19, 1995. The revised text of paragraph (c) of the rule now is as follows (with portions of the paragraph unaffected by the amendment only outlined for context):

(c) Technical specifications will include items in the following categories:

(1) Safety limits, limiting safety system settings, and limiting control settings...

(2) Limiting conditions for operation.

(i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met. When a limiting condition for operation of any process step in the system of a fuel reprocessing plant is not met, the licensee shall shut down that part of the operation or follow any remedial action permitted by the technical specifications until the condition can be met. In the case of a nuclear reactor not licensed under Section 50.21(b) or Section 50.22 of this part or fuel reprocessing plant, the licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the nuclear reactor or the fuel reprocessing plant. In the case of nuclear power reactors licensed under Section 50.21(b) or Section 50.22, the licensee shall notify the Commission if required by Section 50.72 and shall submit a Licensee Event Report to the Commission as required by Section 50.73. In this case, licensees shall retain records associated with preparation of a Licensee Event Report for a period of three years following issuance of the report. For events which do not require a Licensee Event Report, the licensee shall retain each record as required by the technical specifications.

(ii) A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

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(A) Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

(iii) A licensee is not required to propose to modify technical specifications that are included in any license issued before August 18, 1995, to satisfy the criteria in paragraph (c)(2)(ii) of this section.

(3) Surveillance requirements...

(4) Design features...

(5) Administrative controls...

(6) Initial notification...

(7) Written Reports...

Potential Impact Number: 25949 **SRP Section Number:** 16.0

Document: FINAL SER CE80 CH 16
Title: TECHNICAL SPECIFICATIONS

Summary:

The staff review of the System 80+ technical specifications (TS) was closely coupled to the development of the improved standard technical specifications (STS) under the TS

Improvement Program in accordance with the Commission Policy Statement on TS Improvement 58 FR 39132 (SECY-93-067). Since the System 80+ design evolved primarily from the System 80 design, most of the System 80+ TS were modeled after NUREG-1432, "Standard Technical Specifications for Combustion Engineering Plants" (January 1991). These improved STS contain the benefits of the accumulated operating experience from currently operating light-water reactors.

The System 80+ design, however, has some major changes from the System 80 design, particularly in the containment, which consists of a much larger free containment volume to account for severe accident concerns. Other specific changes include the reactor cavity and flooding system, the in-containment refueling water storage tank (IRWST), the Nuplex 80+ advanced control complex, the hydrogen control system, the addition of a safety depressurization system and the addition of a combustion turbine generator (CTG). The System 80+ TS address these changes and the issue of shutdown risk considerations. On this basis, the staff's review of the System 80+ TS concentrated on the differences from the System 80 STS (NUREG-1432).

The staff forwarded to ABB-CE, its comments from the System 80+ proof and review of TS, including Amendments U and V, for resolution and incorporation into the final TS. The final TS were produced in the industry format and certified as accurate by ABB-CE in Amendment W to the CESSAR-DC.

The staff evaluated the System 80+ TS to confirm that they will preserve the validity of the design plant as described in the CESSAR-DC by assuring that System 80+ plants will be operated (1) within the required conditions bounded by the CESSAR-DC, and (2) with operable equipment that is essential to prevent accidents and to mitigate the consequences of accidents postulated in the CESSAR-DC.

On the basis of its review of the System 80+ TS, as discussed above, the staff concludes that the System 80+ TS are consistent with the regulatory guidance contained in the System 80 STS (NUREG-1432) and contain design-specific parameters and additional TS requirements considered appropriate by the staff. Therefore, the staff concludes that the System 80+ TS satisfy 10 CFR 50.34 and 10 CFR 50.36.

Potential Impact Number: 25950 **SRP Section Number:** 10.4.4

Document: FINAL SER ABWR CH 10

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Title: STEAM AND POWER CONVERSION SYSTEM

Summary:

The MS system for the ABWR must also be capable of mitigating the radiological consequences of an accident that could result in potential offsite exposures comparable to the dose reference values specified in 10 CFR Part 100. Most of the currently licensed BWRs rely on the MSIV leakage control system to mitigate the radiological consequences of MSIV leakage following a design-basis loss-of-coolant accident (LOCA) and to stay within 10 CFR Part 100 limits if the MSIV leakage rate exceeded the technical specification limit of 0.3 m3/hr (11.5 ft3/hr). The ABWR will not have an MSIV leakage control system and, therefore, will rely on the MS system coupled with the main condenser (MC) to contain MSIV leakage, relying on plateout and holdup of radioactive iodine and to limit the radiological consequences to within 10 CFR Part 100 limits.

In the DFSER, the staff stated that in order to take credit for the MSSS and MC for containment and holdup of MSIV leakage, the MSSS and the MC and connections from the MSLs to the condenser must be capable of maintaining their integrity during and following an SSE.

Potential Impact Number: 25951 **SRP Section Number:** 10.4.4

Document: FINAL SER EPRI CH 5
Title: EPRI URD-EPD: ENGINEERED SAFETY SYSTEMS

Summary:

EPRI states that a separate MSIV leak detection and control (MSIV-LCS) will not be provided. EPRI is taking credit for the mitigative capabilities of the main steamlines, the bypass lines, and the condenser. The main steam-lines will be designed as seismic Category I from the reactor vessel to the seismic restraint located between the MSIV and the turbine stop valve and Quality Group A from the reactor vessel to and including the outboard MSIV. The staff is continuing its review of EPRI's models for holdup, plateout, and resuspension in the main steamlines, the bypass lines, and the condenser. In order to accept the mitigative capabilities of the main steamlines and the condenser, the staff would need to find these models acceptable and would require the main steamlines, up to and including the turbine stop valve, and the bypass lines to be

designed to seismic Category I and Quality Group B standards, and the condenser would be required to be seismically qualified.

Potential Impact Number: 25952 **SRP Section Number:** 13.5.2.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 10 CFR 50.65

Summary:

(a)(1) Each holder of an operating license under 50.21(b) or 50.22 shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, and components, as defined in paragraph (b), are capable of fulfilling their intended functions. Such goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience. When the performance or condition of a structure, system, or component does not meet established goals, appropriate corrective action shall be taken.

(b) The scope of the monitoring program specified in paragraph (a)(1) of this section shall include safety related and nonsafety related structures, systems, and components, as follows:

(1) Safety related structures, systems, or components that are relied upon to remain functional during and following design basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10 CFR part 100 guidelines.

(2) Nonsafety related structures, systems, or components:

(i) That are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures (EOPs); or

(ii) Whose failure could prevent safety-related structures, systems, and components from fulfilling their safety-related function; or

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(iii) Whose failure could cause a reactor scram or actuation of a safety-related system.

Potential Impact Number: 25953 **SRP Section Number:** 13.3

Document: NRC GENERIC LETTER 89-15
Title: EMERGENCY RESPONSE DATA SYSTEM

Summary:

The Commission has defined the NRC's primary role in an emergency at a licensed nuclear facility as one of monitoring the licensee to assure that appropriate recommendations are made with respect to offsite protective actions. Other aspects of our role include supporting the licensee with technical analysis and logistic support, supporting offsite authorities, including confirming the licensee's recommendations to offsite authorities, keeping other Federal agencies and entities informed of the status of the incident, and keeping the media informed of the NRC's knowledge of the status of the incident.

To fulfill the NRC's role, the NRC requires accurate, timely data on four types of parameters: (l) core and coolant system conditions must be known well enough to assess the extent or likelihood of core damage; (2) conditions inside the containment building must be known well enough to assess its status; (3) radioactivity release rates must be available promptly to assess the immediacy and degree of public danger by these pathways; and (4) the data from the plant's meteorological tower is necessary to provide insight into the potential distribution of a release.

Experience with the voice-only emergency communications link, currently utilized for data transmission, has demonstrated that excessive amounts of time are needed for the routine transmission of data and for verification or correction of data that appear questionable. Error rates have been excessive and the frequency of updates has been unreliable.

The ERDS concept is a direct electronic transmission of selected parameters (Enclosures 1 and 2) for existing facility electronic data systems which have been established by the licensees. The ERDS would be for use only during emergencies at the facilities and would be activated by the licensees upon declaration, of an ALERT or higher emergency classification to begin transmission to the NRC Operations Center. The ERDS would be supplemented with voice transmission over the existing

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Emergency Notification System (ENS) of essential data not available on licensees' systems, rather than requiring modifications to existing systems.

Potential Impact Number: 25954 **SRP Section Number:** 13.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix E to Part 50, section VI.

Summary:

The Emergency Response Data System (ERDS) is a direct near real-time electronic data link between the licensee's onsite computer system and the NRC Operations Center that provides for the automated transmission of a limited data set of selected parameters. The ERDS supplements the existing voice transmission over the Emergency Notification System (ENS) by providing the NRC Operations Center with timely and accurate updates of Q limited set of parameters from the licensee's installed onsite computer system in the event of an emergency. When selected plant data are not available on the licensee's onsite computer system, retrofitting of data points is not required. The licensee shall test the ER!)S periodically to verify system availability and operability. The frequency of ERDS testing will be quarterly unless otherwise set by NRC based on demonstrated system performance.

The system must be capable of transmitting all available ERDS parameters at time intervals of not less than 15 seconds or more than 60 seconds. Exceptions to this requirement will be considered on a case by case basis.

All link control and data transmission must be established in a format compatible with the NRC receiving system /7/ as configured at the time of licensee implementation.

Each licensee shall complete implementation of the ERDS by February 13, 1993, or before initial escalation to full power, whichever comes later. Licensees with currently operational ERDS interfaces approved under the voluntary ERDS implementation program /10/ will not be required to submit another implementation plan and will be considered to have met the requirements for ERDS under appendix E to part 50, section VI.1 and 2 of this part.

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Potential Impact Number: 25955 **SRP Section Number:** 13.3

Document: NUREG 1394 **Revision:** 1 , 6/1/91
Title: Emergency Response Data System (ERDS) Implementation

Summary:

The U.S. Nuclear Regulatory Commission has begun implementation of the Emergency Response Data System (ERDS) to upgrade its ability to acquire data from nuclear power plants in the event of an emergency at the plant. ERDS provides a direct real-time transfer of data from licensee plant computers to the NRC Operations Center. The system has been designed to be activated by the licensee during an emergency which has been classified at an ALERT or higher level. The NRC portion of ERDS will receive the data stream, sort and file the data. The users will include the NRC Operations Center, the NRC Regional Office of the affected plant, and if requested the States which are within the ten mile EPZ of the site. The currently installed Emergency Notification System will be used to supplement ERDS data.

This report provides the minimum guidance for implementation of ERDS at licensee sites. It is intended to be used for planning implementation under the current voluntary program as well as for providing the minimum standards for implementing the proposed ERDS rule.

NUREG-1394 (Rev. 1) is being issued to provide guidance that NRC staff believes should be followed to meet the requirements of the ERDS rule. NUREG- 1394 (Rev. 1) is not a substitute for the regulations, and compliance is not a requirement. However, an approach or method different from the guidance contained here will be accepted only if the substitute approach or method provides a basis for determining that the above-cited regulatory requirements have been met.

Potential Impact Number: 25956 **SRP Section Number:** 17.4

Document: SECY 95-132
Title: POLICY AND TECHNICAL ISSUES ASSOCIATED WITH THE REGULATORY TREATMENT OF NON-SAFETY SYSTEMS (RTNSS) IN PASSIVE PLANT DESIGNS

Summary:

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In SECY-95-132, 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.

Potential Impact Number: 25957 **SRP Section Number:** 17.4

Document: SECY 94-084
Title: POLICY AND TECHNICAL ISSUES ASSOCIATED WITH THE REGULATORY TREATMENT OF NON-SAFETY SYSTEMS IN PASSIVE PLANT DESIGNS

Summary:

In SECY-94-084, 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.

Potential Impact Number: 25958 **SRP Section Number:** 17.4

Document: CODE OF FED. REGS 10CFR50

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Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: **50.65**

Summary:

10 CFR 50.65 requires that power reactor licensees monitor the performance or condition of SSCs against licensee-established goals in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. Such goals are to be established commensurate with safety and, where practical, take into account industry-wide operating experience. When the performance or condition of an SSC does not meet established goals, appropriate corrective action must be taken.

Potential Impact Number: 25959 **SRP Section Number:** 17.4

Document: RG 1.160 **Revision:** 1 , 1/1/95
Title: MONITORING THE EFFECTIVENESS OF MAINTENANCE AT NUCLEAR POWER PLANTS

Summary:

This regulatory guidance is intended to provide flexibility for a licensee to structure its maintenance program in accordance with the safety significance of those SSCs within the scope of the maintenance rule.

The nuclear industry has developed a document that provides guidance to licensees regarding implementation of the maintenance rule (10 CFR 50.65), NUMARC 93-01,
"Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (May 1993) /2/. This document has been prepared by NUMARC. A verification and validation (V&V) effort was conducted by NUMARC, with NRC staff observation, to test the guidance document on several representative systems. A number of changes were made to the NUMARC guidance document based

on the results of the V&V effort. The NRC staff reviewed this document and found that it provides acceptable guidance to licensees. Licensees may use methods other than those provided in NUMARC 93-01 to meet the intent of the maintenance rule. However, the NRC will determine acceptability of other methods on a case-by-case basis.

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Potential Impact Number: 25960 **SRP Section Number:** 8-A

Document: RG 1.153 **Revision:** 0
Title: CRITERIA FOR POWER, INSTRUMENTATION, AND CONTROL

Summary:

Provides guidance for the design, reliability, qualification, and testability of power, instrumentation, and control portions of safety systems for nuclear power plants. The Reg. Guide was not specified as guidance for SRP Sections 8.3.1 and/or 8.3.2 in the previous revision of SRP Section 8.1 Table 8-1 but was added under the SRP-UDP (See ROC 359 for SRP Section 8.1).

The requirements contained in IEEE Std 603-1980 provide a method acceptable to the NRC staff for complying with the Commission's regulations with regard to the design, reliability, qualification, and testability of the power,* instrumentation, and control portions of safety systems as modified and supplemented by the following:

1. The term "safety system" used throughout IEEE Std 603-1980 should be understood to be synonymous with "safety-related system," in which the term "safety related" has the meaning stated in Section 50.49(b)(1) of 10 CFR Part 50.

2. For displays for manually controlled actions covered in Section 5.8.1 of IEEE Std 603-1980, the provisions for Type A instruments in Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," should be followed instead of IEEE Std 497-1977.

3. Instead of the first sentence in Section 6.3.1(1) of IEEE Std 603-1980, the first sentence in Section 4.7.4.1 of IEEE Std 279 should be used: "Alternate channels, not subject to failure resulting from the same single event, shall be provided to limit the consequences of this event to a value specified by the design bases."

4. Section 6.3.1 of IEEE Std 603-1980 references Figure 7, a decision chart for applying the requirements of the section. Figure 1 of this guide should be used in lieu of Figure 7 of IEEE Std 603-1980.

5. Section 3 of IEEE Std 603-1980 lists additional standards that are referenced in the standard. Those referenced standards not endorsed by a regulatory guide or

incorporated into the regulations contain valuable information; if used, they should be used in a manner consistent with current regulatory practice.

* The term power includes electric, pneumatic, and hydraulic power.

Potential Impact Number: 25961 **SRP Section Number:** 8-A

Document: C&S: IEEE 279
Title: STANDARD CRITERIA FOR SAFETY SYSTEMS FOR NUCLEAR POWER GENERATING STATIONS

Summary:

Per 10 CFR 50.55a, paragraph (h), for construction permits issued after January 1, 1971, protection systems must meet the requirements set forth in editions or revisions of the Institute of Electrical and Electronics Engineers Standard: "Criteria for Protection Systems for Nuclear Power Generating Stations, (IEEE 279) in effect on the formal docket date of the application for a construction permit. Protection systems may meet the requirements set forth in subsequent editions or revisions of IEEE 279 which become effective.

For purposes of this regulation the proposed IEEE 279 became "in effect" on August 30, 1968, and the revised issue IEEE 279-1971 became "in effect" on June 3, 1971. Copies may be obtained from the Institute of Electrical and Electronics Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. Copies are available for inspection at the Commission's Technical Library, 11545 Rockville Pike, Two White Flint North, Rockville, Maryland 20852-2738.

Where an application for a construction permit is submitted in four parts pursuant to the provisions of Section 2.101(a-1) and Subpart F of 10 CFR Part 2, "the formal docket date of the application for a construction permit" for purposes of 50.55a(h) shall be the date of docketing of the information required by Section 2.101(a-1) (2) or (3), whichever is later.

Potential Impact Number: 25962 **SRP Section Number:** 8-A

Document: C&S: IEEE 603

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Title: CRITERIA FOR SAFETY SYSTEMS FOR NUCLEAR POWER GENERATING STATIONS

Summary:

Per Regulatory Guide 1.153, 10 CFR Part 50 contains, among other things, requirements for the design, reliability, qualification, and testability of the protection system, systems that accomplish safety functions, and other systems (known as auxiliary supporting features) that are essential to the operation of the protection system and the accomplishment of the safety functions. These systems, taken collectively, have been designated as safety systems...

In addition, Section 50.55a, "Codes and Standards," of 10 CFR Part 50 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" (also designated ANSI N42.7-1972).

Per Regulatory Guide 1.153, the requirements contained in IEEE Std 603-1980 provide a method acceptable to the NRC staff for complying with the Commission's regulations with regard to the design, reliability, qualification, and testability of the power (the term power includes electric, pneumatic, and hydraulic power), instrumentation, and control portions of safety systems as modified and supplemented by the guide's regulatory positions. See Impact 25960 for details of the regulatory positions.

Potential Impact Number: 25964 **SRP Section Number:** 17.3

Document: NRC GENERIC LETTER 88-18
Title: PLANT RECORD STORAGE ON OPTICAL DISKS

Summary:

Appendix B of 10 CFR 50, under criterion XVII, "Quality Assurance Records," establishes requirements for a record keeping system, and chapter 17 of the Standard Review Plan (NUREG-0800) expounds on these requirements, The purpose of the record keeping system is to ensure that records are available when needed. ANSI N45.2.9-1974, "Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants," as endorsed by Regulatory Guide 1.88 (Rev. 2) and Basic Requirements 17, Supplement 175-1, and Appendix 17A-1 of ANSI/ASME-NQA-1 1983 edition, "Quality Assurance Program Requirements for

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Nuclear Facilities," as endorsed by Regulatory Guide 1.28 (Rev. 3) both describe quality assurance controls regarding records. Appropriate quality controls for an optical disk document imaging system include the following.

- The optical disk technology does not allow deletion or modification of record images.

- The image of each record is written onto two optical disks.

- The legibility of each record image is verified to ensure that the image is legible on both disks. If the image is illegible, the hard copy record is maintained as the record copy.

- One optical disk is stored in the document imaging system for on-line retrieval.

- The second (backup) optical disk is stored in a records storage facility meeting the requirements of ANSI N45.2.9-1974 for single copy storage or in a separate remote location.

- To ensure permanent retention of records, the records stored on an optical disk are acceptably copied onto a new optical disk before the manufacturer's certified useful life of the original disk is exceeded. This includes verification of the records so copied.

- Periodic random inspections of images stored on optical disks are performed to verify that there has been no degradation of image quality.

- If the optical disk document imaging system in use is to be replaced by an incompatible new system, the records stored on the old system's disks are acceptably converted into the new system before the old system is taken out of service. This includes verification of the records so copied.

Potential Impact Number: 25965 **SRP Section Number:** 3.8.1

Document: FINAL SER ABWR CH 3
Title: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Summary:

The containment is designed as a reinforced-concrete cylindrical shell structure with an internal steel liner made of carbon steel, except for wetted surfaces where stainless

steel or carbon steel with stainless steel cladding will be used. It is divided by the diaphragm floor and the reactor pedestal into an upper drywell chamber, a lower drywell chamber and a suppression chamber.

Potential Impact Number: 25966 **SRP Section Number:** 8-A

Document: RG 1.9 **Revision:** 3 , 7/1/93
Title: SELECTION, DESIGN, AND QUALIFICATION OF DIESEL-GENERATOR UNITS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

Summary:

This regulatory guide 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.

Conformance with the guidelines in IEEE Std 387-1984, "IEEE Standard Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations," is acceptable to the NRC staff for satisfying the Commission's regulations with respect to design, qualification, and periodic testing of diesel generator units used as onsite electric power systems for nuclear power plants subject to the following:

1. DESIGN CONSIDERATIONS

The guidelines of IEEE Std 387-1984 should be supplemented as follows:...

1.6 Design provisions should include the capability to test each emergency diesel generator unit independently of the redundant units. Test equipment should not cause a loss of independence between redundant diesel generator units or between diesel generator load groups. Testability should be considered in the selection and location of instrumentation sensors and critical components (e.g., governor, starting system components). Instrumentation sensors should be readily accessible and designed so that their inspection and calibration can be verified in place. The overall design should include status indication and alarm features.

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1.7 Section 5.5.3.1 , " Surveillance Systems," of IEEE Std 387-1984 pertains to status indication of diesel generator unit conditions. The guidance in this section should be supplemented as follows:

1.7.1 A surveillance system should be provided with remote indication in the control room for displaying emergency diesel generator unit status, i.e., under test, ready-standby, lockout. A means of communication should also be provided between diesel generator unit testing locations and the main control room to ensure that the operators are cognizant of the status of the unit under test.

1.7.2 In order to facilitate trouble diagnosis, the surveillance system should indicate which of the emergency diesel generator protective trips has been activated first.

1.8 Section 5.5.4, "Protection," of IEEE Std 387-1984, which pertains to bypassing emergency diesel generator protective trips during emergency conditions, should be supplemented as follows:

The emergency diesel generator unit should be automatically tripped on an engine overspeed and generator-differential over-current. All other diesel generator protective trips should be handled in one of two ways: (1) a trip should be implemented with two or more measurements for each trip parameter with coincident logic provisions for trip actuation, or (2) a trip may be bypassed under accident conditions provided the operator has sufficient time to react appropriately to an abnormal diesel generator unit condition. The design of the coincident logic trip circuitry should include the capability for indication of individual sensor trips. The design of the bypass circuitry should include the capability for (1) testing the status and operability of the bypass circuits, (2) alarming in the control room for abnormal values of all bypass parameters (common trouble alarms may be used), and (3) manually resetting the trip bypass function. Capability for automatic reset is not acceptable.

Section 5.5.4(2) of IEEE Std 387-1984. on retaining all protective devices during emergency diesel generator testing, does not apply to a periodic test that demonstrates diesel generator system response under simulated accident conditions per Regulatory Positions 2.2.5, 2.2.6, and 2.2.12.

Potential Impact Number: 25967 **SRP Section Number:** 8-A

Document: RG 1.47 **Revision:** 0

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Title: BYPASSED AND INOPERABLE STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS

Summary:

This guide 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 (as defined in IEEE Std 279-1971), 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.

Potential Impact Number: 25968 **SRP Section Number:** 14.3

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Locations: 52.47(b), 52.47(b), 52.97(b)(1)

Summary:

10 CFR 52.47(b) establishes the requirements for design certification applications for plants that are significantly different from currently licensed designs. For the purposes of the review conducted in accordance with SRP Section 14.3 and its associated subsections, this paragraph of the rule establishes the scope of the design certification to which the ITAAC review applies.

10 CFR 52.97(b)(1) sets forth the Commission's comprehensive requirements for ITAAC, which provide reasonable assurance that the facility has been constructed, and will be operated in conformance with the license, the provisions of the Atomic Energy Act and the Commission's rules and regulations.

Potential Impact Number: 25969 **SRP Section Number:** 14.3.1

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS

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Locations: 52.47(a)(1)(iii), 52.47(a)(1)(iii), 52.97(b)(1), 52.47(b)

Summary:

10 CFR 52.47 (a)(1)(iii) establish requirements for the content of design certification applications with regard to site parameters.

10 CFR 52.47(b) establishes the requirements for design certification applications for plants that are significantly different from currently licensed designs. For the purposes of the review conducted in accordance with SRP Section 14.3 and its associated subsections, this paragraph of the rule establishes the scope of the design certification to which the ITAAC review applies.

10 CFR 52.97(b)(1) sets forth the Commission's comprehensive requirements for ITAAC, which provide reasonable assurance that the facility has been constructed, and will be operated in conformance with the license, the provisions of the Atomic Energy Act and the Commission's rules and regulations.

Potential Impact Number: 25970 **SRP Section Number:** 14.3.2

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Locations: 52.47(b), 52.47(b), 52.97(b)(1)

Summary:

10 CFR 52.47(b) establishes the requirements for design certification applications for plants that are significantly different from currently licensed designs. For the purposes of the review conducted in accordance with SRP Section 14.3 and its associated subsections, this paragraph of the rule establishes the scope of the design certification to which the ITAAC review applies.

10 CFR 52.97(b)(1) sets forth the Commission's comprehensive requirements for ITAAC, which provide reasonable assurance that the facility has been constructed, and will be operated in conformance with the license, the provisions of the Atomic Energy Act and the Commission's rules and regulations.

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Potential Impact Number: 25971 **SRP Section Number:** 14.3.3

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Locations: 52.47(b), 52.47(b), 52.97(b)(1)

Summary:

10 CFR 52.47(b) establishes the requirements for design certification applications for plants that are significantly different from currently licensed designs. For the purposes of the review conducted in accordance with SRP Section 14.3 and its associated subsections, this paragraph of the rule establishes the scope of the design certification to which the ITAAC review applies.

10 CFR 52.97(b)(1) sets forth the Commission's comprehensive requirements for ITAAC, which provide reasonable assurance that the facility has been constructed, and will be operated in conformance with the license, the provisions of the Atomic Energy Act and the Commission's rules and regulations.

Potential Impact Number: 25972 **SRP Section Number:** 14.3.4

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Locations: 52.47(b), 52.47(b), 52.97(b)(1)

Summary:

10 CFR 52.47(b) establishes the requirements for design certification applications for plants that are significantly different from currently licensed designs. For the purposes of the review conducted in accordance with SRP Section 14.3 and its associated subsections, this paragraph of the rule establishes the scope of the design certification to which the ITAAC review applies.

10 CFR 52.97(b)(1) sets forth the Commission's comprehensive requirements for ITAAC, which provide reasonable assurance that the facility has been constructed, and will be operated in conformance with the license, the provisions of the Atomic Energy Act and the Commission's rules and regulations.

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Potential Impact Number: 25973 **SRP Section Number:** 14.3.5

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Locations: 52.47(b), 52.47(b), 52.97(b)(1)

Summary:

10 CFR 52.47(b) establishes the requirements for design certification applications for plants that are significantly different from currently licensed designs. For the purposes of the review conducted in accordance with SRP Section 14.3 and its associated subsections, this paragraph of the rule establishes the scope of the design certification to which the ITAAC review applies.

10 CFR 52.97(b)(1) sets forth the Commission's comprehensive requirements for ITAAC, which provide reasonable assurance that the facility has been constructed, and will be operated in conformance with the license, the provisions of the Atomic Energy Act and the Commission's rules and regulations.

Potential Impact Number: 25974 **SRP Section Number:** 14.3.6

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Locations: 52.47(b), 52.47(b), 52.97(b)(1)

Summary:

10 CFR 52.47(b) establishes the requirements for design certification applications for plants that are significantly different from currently licensed designs. For the purposes of the review conducted in accordance with SRP Section 14.3 and its associated subsections, this paragraph of the rule establishes the scope of the design certification to which the ITAAC review applies.

10 CFR 52.97(b)(1) sets forth the Commission's comprehensive requirements for ITAAC, which provide reasonable assurance that the facility has been constructed, and

will be operated in conformance with the license, the provisions of the Atomic Energy Act and the Commission's rules and regulations.

Potential Impact Number: 25975 **SRP Section Number:** 14.3.7

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Locations: 52.47(b), 52.47(b), 52.97(b)(1)

Summary:

10 CFR 52.47(b) establishes the requirements for design certification applications for plants that are significantly different from currently licensed designs. For the purposes of the review conducted in accordance with SRP Section 14.3 and its associated subsections, this paragraph of the rule establishes the scope of the design certification to which the ITAAC review applies.

10 CFR 52.97(b)(1) sets forth the Commission's comprehensive requirements for ITAAC, which provide reasonable assurance that the facility has been constructed, and will be operated in conformance with the license, the provisions of the Atomic Energy Act and the Commission's rules and regulations.

Potential Impact Number: 25976 **SRP Section Number:** 14.3.8

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Locations: 52.47(b), 52.47(b), 52.97(b)(1)

Summary:

10 CFR 52.47(b) establishes the requirements for design certification applications for plants that are significantly different from currently licensed designs. For the purposes of the review conducted in accordance with SRP Section 14.3 and its associated subsections, this paragraph of the rule establishes the scope of the design certification to which the ITAAC review applies.

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10 CFR 52.97(b)(1) sets forth the Commission's comprehensive requirements for ITAAC, which provide reasonable assurance that the facility has been constructed, and will be operated in conformance with the license, the provisions of the Atomic Energy Act and the Commission's rules and regulations.

Potential Impact Number: 25977 **SRP Section Number:** 14.3.9

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Locations: 52.47(b), 52.47(b), 52.97(b)(1)

Summary:

10 CFR 52.47(b) establishes the requirements for design certification applications for plants that are significantly different from currently licensed designs. For the purposes of the review conducted in accordance with SRP Section 14.3 and its associated subsections, this paragraph of the rule establishes the scope of the design certification to which the ITAAC review applies.

10 CFR 52.97(b)(1) sets forth the Commission's comprehensive requirements for ITAAC, which provide reasonable assurance that the facility has been constructed, and will be operated in conformance with the license, the provisions of the Atomic Energy Act and the Commission's rules and regulations.

Potential Impact Number: 25978 **SRP Section Number:** 14.3.10

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Locations: 52.47(b), 52.47(b), 52.97(b)(1)

Summary:

10 CFR 52.47(b) establishes the requirements for design certification applications for plants that are significantly different from currently licensed designs. For the purposes of the review conducted in accordance with SRP Section 14.3 and its associated

subsections, this paragraph of the rule establishes the scope of the design certification to which the ITAAC review applies.

10 CFR 52.97(b)(1) sets forth the Commission's comprehensive requirements for ITAAC, which provide reasonable assurance that the facility has been constructed, and will be operated in conformance with the license, the provisions of the Atomic Energy Act and the Commission's rules and regulations.

Potential Impact Number: 25979 **SRP Section Number:** 14.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.55a

Summary:

10 CFR 50.55a requires that structures, systems and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance to the safety function performed. In this regard, 10 CFR 50.55a identifies the required portions of the ASME Boiler and Pressure Vessel Code applicable to structures, systems and components important to safety.

Potential Impact Number: 25980 **SRP Section Number:** 14.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 1

Summary:

GDC 1 requires, in part, that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

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Potential Impact Number: 25981 **SRP Section Number:** 14.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 2

Summary:

GDC 2 requires, in part, requires structures, systems and components important to safety to be designed to withstand the effects of natural phenomena, including in combination with appropriate normal or accident conditions, without loss of the capability to perform their safety function.

Potential Impact Number: 25982 **SRP Section Number:** 14.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 4

Summary:

GDC 4 requires, in part, that structures, systems, and components important to safety be protected from the dynamic effects of equipment failures and from events and conditions outside the nuclear power unit.

Potential Impact Number: 25983 **SRP Section Number:** 14.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 14

Summary:

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GDC 14 requires, in part, that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture.

Potential Impact Number: 25984 **SRP Section Number:** 14.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 16

Summary:

GDC 16, requires, in part, that reactor containment provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment.

Potential Impact Number: 25985 **SRP Section Number:** 14.3.2

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 50

Summary:

GDC 50 requires, in part, that the reactor containment structure, including access openings and penetrations be designed such that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

Potential Impact Number: 25986 **SRP Section Number:** 14.3.4

Document: SECY 90-016

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Title: EVOLUTIONARY LIGHT-WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Summary:

SECY 90-016 describes several issues the staff considered fundemental to agency decisions on the acceptability of evolutionary ALWR designs and includes areas where the staff proposes to depart from current Commission regulations.

Potential Impact Number: 25987 **SRP Section Number:** 14.3.4

Document: SECY 93-087
Title: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

Summary:

SECY 93-087 and related staff requirements memorandum established the Commissions position with regard to certain policy, technical, and licensing issues pertaining to evolutionary and advanced light water designs.

Potential Impact Number: 25989 **SRP Section Number:** 14.3.1

Document: RG 1.60 Revision: 1
Title: DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS

Summary:

Regulatory Guide 1.60 provides guidance acceptable to the staff for defining seismic response spectra for the seismic design of nuclear power plants.

Potential Impact Number: 25990 **SRP Section Number:** 14.3.5

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Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.55a(h)

Summary:

10 CFR 50.55a(h) and referenced standard IEEE Std. 279-1971 provide the requirements for safety related protection systems for nuclear power plants.

Potential Impact Number: 25991 **SRP Section Number:** 14.3.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 1

Summary:

GDC 1 establishes the requirements for quality standards and records for design, fabrication, erection, and testing of structures, systems and components important to safety.

Potential Impact Number: 25992 **SRP Section Number:** 14.3.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 2

Summary:

GDC 2 establishes the requirements to design structures, systems and components important to safety to withstand the effects of natural phenomena, including earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches, without loss of capability to perform their safety functions.

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Potential Impact Number: 25993 **SRP Section Number:** 14.3.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 4

Summary:

GDC 4 establishes the requirements for structures, systems and components important to safety to be designed to accommodate the environmental conditions associated with normal operations, maintenance, testing and postulated accidents. These structures, systems and components shall also be protected against the dynamic effects of equipment failures.

Potential Impact Number: 25994 **SRP Section Number:** 14.3.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 13

Summary:

GDC 13 establishes the requirements for instrumentation and controls to monitor variables and systems over their anticipated ranges for normal operation, anticipated operational occurrences, and accident conditions, and control these variables and systems within their prescribed operating ranges.

Potential Impact Number: 25995 **SRP Section Number:** 14.3.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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Location: Appendix A, GDC 19

Summary:

GDC 19 establishes the requirements to provide a control room to provide the capability to operate the nuclear power unit safely under normal operations and to maintain safe shutdown under accident conditions.

Potential Impact Number: 25996 **SRP Section Number:** 14.3.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 20

Summary:

GDC 20 establishes the requirements for the design of reactor protection systems to automatically initiate operation of appropriate systems, including those for reactivity control, to ensure that specified acceptable fuel design limits are not exceeded for anticipated operational occurrences; and to sense accident conditions and initiate operation of systems important to safety.

Potential Impact Number: 25997 **SRP Section Number:** 14.3.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 21

Summary:

GDC 21 establishes the requirements for reliability and testability of the protection system, including design requirements for independence and redundancy.

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Potential Impact Number: 25998 **SRP Section Number:** 14.3.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 22

Summary:

GDC 22 establishes the requirements for independence in the design of the reactor protection system. Appropriate independence must be provided to ensure that the effects of natural phenomena, normal operations, maintenance, testing, and accident conditions do not result in a loss of the protection function.

Potential Impact Number: 25999 **SRP Section Number:** 14.3.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 23

Summary:

GDC 23 establishes the requirements for acceptable failure states for the design of protection systems.

Potential Impact Number: 26000 **SRP Section Number:** 14.3.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 25

Summary:

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GDC 25 establishes the requirements for the protection system to ensure that reactivity control malfunctions do not result in conditions exceeding the specified acceptable fuel design limits.

Potential Impact Number: 26001 **SRP Section Number:** 14.3.5

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 29

Summary:

GDC 29 establishes the requirements for reactor protection and reactivity control systems to be designed to provide an extremely high probability that they will perform their safety functions for anticipated operational occurrences.

Potential Impact Number: 26003 **SRP Section Number:** 14.3.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 17

Summary:

GDC 17 establishes the requirements for the design of onsite and offsite electric power systems 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 anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

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Potential Impact Number: 26004 **SRP Section Number:** 14.3.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.49

Summary:

10 CFR 50.49 establishes the requirements for certain electrical equipment important to safety to be environmentally qualified to perform its safety function in environments associated with normal, and accident conditions.

Potential Impact Number: 26005 **SRP Section Number:** 14.3.6

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.63

Summary:

10 CFR 50.63 establishes the requirements that a nuclear power plant be able to withstand for a specified duration and recover from a station blackout.

Potential Impact Number: 26006 **SRP Section Number:** 14.3.6

Document: C&S: IEEE 308
Title: IEEE STANDARD CRITERIA FOR CLASS 1E POWER SYSTEMS FOR NUCLEAR POWER GENERATING STATIONS

Summary:

IEEE Std 308 provides acceptable guidance and criteria for the design of Class IE power system for nuclear power plants.

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Potential Impact Number: 26007 **SRP Section Number:** 14.3.7

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.49

Summary:

10 CFR 50.49 establishes the requirements for certain electrical equipment important to safety to be environmentally qualified to perform its safety function in environments associated with normal, and accident conditions.

Potential Impact Number: 26008 **SRP Section Number:** 14.3.7

Document: RG 1.97 Revision: 3
Title: INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT

Summary:

Regulatory Guide 1.97 provides guidance for providing instrumentation to monitor plant variables and systems during and following an accident.

Potential Impact Number: 26009 **SRP Section Number:** 14.3.8

Document: CODE OF FED. REGS 40CFR190
Title: ENVIRONMENTAL RADIATION PROTECTION STANDARDS FOR NUCLEAR PLANT OPERATIONS

Summary:

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40 CFR 190 provides the Environmental Protection Agencies requirements with regard to radiation doses received by members of the public in the general environment and to radioactive materials introduced into the general environment as the result of operations which are part of a nuclear fuel cycle.

Potential Impact Number: 26010 **SRP Section Number:** 14.3.8

Document: SECY 95-090
Title: EMERGENCY PLANNING UNDER 10 CFR PART 52

Summary:

SECY 95-090 provided the Commission with the staff's views on how emergency planning requirements will be addressed at each phase of nuclear power plant licensing under 10 CFR 52, including applications for early site permits, design certification and combined license. The positions and views of the staff as provided in the SECY paper were developed, in part, from information provided in the draft Supplement 2 to NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," that is attached to the SECY paper.

Potential Impact Number: 26011 **SRP Section Number:** 14.3.9

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.34(f)(2)(iii)

Summary:

10 CFR 50.34(f)(2)(iii) establishes the NRC's requirements with regard to human factors design of the control room for nuclear power plants.

Potential Impact Number: 26012 **SRP Section Number:** 14.3.9

APPENDIX II
Potential Impacts

Document: RG 1.97 Revision: 3
Title: INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT

Summary:

Regulatory Guide 1.97 provides the guidance and methods acceptable to the staff for providing instrumentation for monitoring plant variables and systems during and following an accident.

Potential Impact Number: 26013 **SRP Section Number:** 14.3.10

Document: RG 1.68 Revision: 2
Title: INITIAL TEST PROGRAMS FOR WATER-COOLED NUCLEAR POWER PLANTS

Summary:

Regulatory Guide 1.68 describes the general scope and depth of testing that is acceptable to the staff for conduct of pre-operational and startup testing.

Potential Impact Number: 26014 **SRP Section Number:** 18.0

Document: NUREG 0700
Title: GUIDELINES FOR CONTROL ROOM DESIGN REVIEWS

Summary:

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 of NUREG-0700, Revision 0, was provided as one source of guidance regarding these analyses.

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The guidance of NUREG-0700, Revision 0, was updated in NUREG-0700, Revision 1, to reflect changes in HFE review concerns and HSI technologies.

Potential Impact Number: 26015 **SRP Section Number:** 18.0

Document: NUREG 0711
Title: HUMAN FACTORS ENGINEERING PROGRAM REVIEW MODEL

Summary:

The staff of the Nuclear Regulatory Commission is performing nuclear power plant design certification reviews based on a design process plan that describes the human factors engineering (HFE) program elements that are necessary and sufficient to develop an acceptable detailed design specification and an acceptable implemented design. Current regulations and guidance documents do not address the criteria for design process review. Therefore, the HFE Program Review Model (HFE PRM) was developed as a basis for performing design certification reviews that include design process evaluations as well as review of the final design. A central tenet of the HFE PRM is that the HFE aspects of the plant should be developed, designed, and evaluated on the basis of a structured top-down system analysis using accepted HFE principles. NUREG-0711 describes the development of the HFE PRM and gives a detailed description of each HFE review element.

Potential Impact Number: 26016 **SRP Section Number:** 8.2

Document: FINAL SER ABWR CH 8
Title: ELECTRIC POWER SYSTEMS

Summary:

In SECY-91-078, the staff recommended that the Commission approve its position that an evolutionary plant design should include an alternate power source to the non-safety loads unless it can be demonstrated that the design margins will result in transients for a loss of non-safety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs. In its August 15, 1991, SRM, the Commission approved the staff's position. The staff in NUREG-1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility

Requirements Document," Volume 2 (August 1992), clarified the intent of this position. Further, the staff has evaluated the electrical design for the ABWR relative to the provision of an alternate source of power for non-safety loads against the following proposed applicable regulation, which is based on the position included in NUREG-1242: "The electric power system of the standard design must include an alternate power source that is provided to a sufficient string of non-safety loads so that forced circulation could be maintained, and the operator has available the complement of non-safety equipment that would most facilitate the ability to bring the plant to a stable shutdown condition, following a loss of the normal power supply and plant trip."

In SECY-91-078, the staff recommended that the Commission approve this posi- tion for evolutionary plant designs. The evolutionary plant design should include at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus. In its August 15, 1991, SRM, the Commission approved the staff's position.

Therefore, the staff's proposed applicable regulation for offsite power sources is as follows:

The electric power system of standard plant design must include at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus.

Potential Impact Number: 26017 **SRP Section Number:** 8.2

Document: FINAL SER CE80 CH 8
Title: ELECTRIC POWER SYSTEMS

Summary:

In SECY-91-078, the staff recommended that the Commission approve the staff position that an evolutionary plant design should include at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus. In its August 15, 1991, SRM, the Commission approved the staff's position.

APPENDIX II
Potential Impacts

In SECY-91-078, the staff recommended that the Commission approve the staff's position that an evolutionary plant design should include an alternate power source to the non-safety-related loads, unless it can be demonstrated that the design margins will result in transients for a loss of non-safety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs. In its August 15, 1991 SRM, the Commission approved the staff's position. The staff, in its safety evaluation report (SER) for the EPRI Evolutionary Utility Requirements Document (URD) clarified the intent of this position: "...an alternate power source be provided to a sufficient string of non-safety loads so that forced circulation could be maintained, and the operator would have available to him the complement of non-safety equipment that would most facilitate his ability to bring the plant to a stable shutdown condition, following a loss of the normal power supply and plant trip."

The staff proposed applicable regulations for electrical distribution, as follows:

(1) at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non- safety buses in such manner that the offsite source can power the safety buses upon failure of any non-safety bus.

(2) an alternate power source that is provided to a sufficient string of non-safety loads so that forced circulation could be maintained, and the operator has available a complement of non-safety equipment that would most facilitate the ability to bring the plant to a stable shutdown condition, following a loss of the normal power supply and plant trip.

The staff has determined that conformance to the applicable GDC and guidelines cited above is a sufficient basis for accepting the electric power systems for the System 80+ design.

Potential Impact Number: 26018 **SRP Section Number:** 8.2

Document: FINAL SER CE80 CH 8
Title: ELECTRIC POWER SYSTEMS

Summary:

The NRC developed a policy issue in SECY-90-016, dated January 12, 1990, that was approved by the Commission on June 26, 1990, which requires that the evolutionary ALWRs meet the SBO rule by including an AAC power source (e.g., CTG) of diverse

design capable of powering at least one complete set of normal shutdown loads and to back up the EDGs. EPRI has also included a requirement that a large-capacity, diverse AAC power source (e.g., CTG) with the capacity to power one complete set of normal safe-shutdown loads and to back up the EDG be part of the evolutionary ALWR design. Therefore, the staff's proposed applicable regulation as the standard design must provide an alternate ac power source for the purpose of dealing with station blackout.

Potential Impact Number: 26019 **SRP Section Number:** 8.2

Document: FINAL SER ABWR CH 8
Title: ELECTRIC POWER SYSTEMS

Summary:

In SECY-90-016, dated January 12, 1990, the staff recommended that the Commission approve its position that a diverse, AAC source, be provided for SBO in evolutionary LWRs. In its June 26, 1990, SRM, the Commission approved the staff's position.

Potential Impact Number: 26020 **SRP Section Number:** 8.3.1

Document: RG 1.160 **Revision:** 1 , 1/1/95
Title: MONITORING THE EFFECTIVENESS OF MAINTENANCE AT NUCLEAR POWER PLANTS

Summary:

The scope of monitoring efforts under the maintenance rule, as defined in 10 CFR 50.65(b), encompasses those SSCs that directly and significantly affect plant operations, regardless of what organization actually performs the maintenance activities. Maintenance activities that occur in the switchyard can directly affect plant operations; and as a result, electrical distribution equipment out to the first inter-tie with the off-site distribution system (i.e., equipment in the switchyard) should be included as defined in 10 CFR 50.65(b).

NUMARC 93-01 provides methods acceptable to the NRC staff for complying with the provisions of 10 CFR 50.65. NUMARC 93-01 references other documents, but NRC's

endorsement of NUMARC 93-01 should not be considered as endorsement of the referenced documents.

Section D, Implementation indicates that the guide (Revision 1) will be used to evaluate the effectiveness of emergency diesel generator maintenance activities associated with compliance with 10 CFR 50.63.

Potential Impact Number: 26021 **SRP Section Number:** 17.2

Document: RG 4.15 **Revision: 1**
Title: QUALITY ASSURANCE FOR RADIOLOGICAL MONITORING PROGRAMS (NORMAL OPERATIONS)--EFFLUENT STREAMS AND THE ENVIRONMENT

Summary:

The need of quality assurance is implicit in all requirements for effluent and environmental monitoring, and this need has been widely recognized. Regulatory Guide 1. 21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants;" Regulatory Guide 4.1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants;" Regulatory Guide 4. 8, "Environmental Technical Specifications for Nuclear Power Plants;" and Regulatory Guide 4.14, "Measuring, Evaluating, and Reporting Radioactivity in Releases of Radioactive Material in Liquid and Airborne Effluents From Uranium Mills," all give some guidance on means for assuring the quality of the measurements of radioactive materials in effluents and the environment outside of nuclear facilities. More complete and extensive guidance on this subject is provided in this document for nuclear power reactor facilities and for other facilities for which radiological monitoring is 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.

Potential Impact Number: 26022 **SRP Section Number:** 8.1

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

APPENDIX II
Potential Impacts

Location: GDC 17

Summary:

Criterion 17 -- Electric power systems. GDC 17 is referenced as acceptance criteria for SRP Sections 8.2, 8.3.1, and 8.3.2 in Table 8-1.

An onsite electric power system 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 anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Potential Impact Number: 26023 **SRP Section Number:** 8.1

APPENDIX II
Potential Impacts

Document: NRC MEMORANDUM 12/15/80, from F. Rosa to C. Michelson
Title: AEOD REPORT ON THE LOSS OF OFFSITE POWER EVENT AT ARKANSAS NUCLEAR ONE, APRIL 7, 1980 (REPORT DATED OCTOBER 15, 1980)

Summary:

Enclosure 1 to this memorandum provides the interpretation of the requirements of General Design Criterion 17 (GDC 17) which was used by the Power Systems Branch (PSB) in licensing reviews. This interpretation was provided in support of comments on an AEOD report on the Loss of Offsite Power event at Arkansas Nuclear One, April 7, 1980.

The enclosure breaks out the GDC by paragraph and provides a more suscinct description of the staff interpretation of the GDC requirements.

Potential Impact Number: 26030 **SRP Section Number:** 1.8

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix O, Para. 1 and 3

Summary:

Any person may submit a proposed preliminary or final standard design for a nuclear power reactor to the regulatory staff for its review.

The submittal shall also include a description, analysis and evaluation of the interfaces between the submitted design and the balance of the nuclear power plant.

NOTE: This PI is a duplicate of PI 20311.

Potential Impact Number: 26031 **SRP Section Number:** 1.8

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS

APPENDIX II
Potential Impacts

Location: 52.47 (a)(1)(vi)

Summary:

(vi) Proposed tests, inspections, analyses, and acceptance criteria 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 design certification.

vii) The interface requirements to be met by those portions of the plant for which the application does not seek certification. These requirements must be sufficiently detailed to allow completion of the final safety analysis and design-specific probabilistic risk assessment required by paragraph (a)(1)(v) of this section;

(viii) Justification that compliance with the interface requirements of paragraph (a)(1)(vii) of this section is verifiable through inspection, testing (either in the plant or elsewhere), or analysis. The method to be used for verification of interface requirements must be included as part of the proposed tests, inspections, analyses, and acceptance criteria required by paragraph (a)(1)(vi) of this section;

NOTE: This is a duplicate of PI 20313.

Potential Impact Number: 26032 **SRP Section Number:** 1.8

Document: DRAFT SER CE80 CH 1
Title: INTRODUCTION AND GENERAL DISCUSSION

Summary:

The staff's technical review of ABB-CE's application for certification of the System 80+ standard design was performed in accordance with Commission guidance and the requirements of Sections 52.47 and 52.48. This draft SER describes the results of the staff's technical review to date and a subsequent final SER will document the completion of the staff's review. The staff's evaluation of the technical information required by 52.47(a)(1)(i) was performed in accordance with the standard review plan (NUREG-0800) and is set forth throughout this report..

A guide to the staff's evaluation of the interface requirements that are required by 52.47(a)(1)(vii) is provided in Section 1.8 of this report.

APPENDIX II
Potential Impacts

Section 52.47(a)(2) describes the level of design information needed to certify a standard design. The acceptable level of design detail necessary for the staff to make its safety findings was one of the most challenging aspects of the staff's review. The SRM for SECY-90-377, 'Requirements for Design Certification Under 10 CFR Part 52,' sets forth the Commission's position on what level of design information is required for a certification application, and the staff has followed that guidance in preparing this document.

NOTE: This PI is a duplicate of PI 22895.

Potential Impact Number: 26033 **SRP Section Number:** 1.8

Document: DRAFT FINAL SER ABWR CH 1
Title: INTRODUCTION AND GENERAL DISCUSSION

Summary:

The staff's evaluation of the technical information required by 52.47(a)(1)(i) was performed in accor-dance with the standard review plan (NUREG-0800) and is set forth throughout this report.

A guide to the staff's evaluation of the interface requirements that are required by 52.47(a)(1)(vii) is provided in Section 1.8 of this report.

Section 52.47(a)(2) describes the level of design information needed to certify a standard design. The acceptable level of design detail necessary for the staff to make its safety findings was one of the most challenging aspects of the staff's review. The SRM for SECY-90-377, 'Requirements for Design Certification Under 10 CFR Part 52,' sets forth the Commission's position on what level of design information is required for a certification application, and the staff has followed that guidance in preparing this document.

1.8 Interface Requirements; Section 52.47(a)(1)(vii) requires interface requirements that must be met by the site-specific elements of the plant design, such as the ultimate heat sink. A discussion of the certified scope of the ABWR standard design is provided in Section 1.2.6 of this report. The staff has identified certain interface requirements as part of its technical review to date.

NOTE: This PI is a duplicate of PI 22896.

APPENDIX II
Potential Impacts

Potential Impact Number: 26034 **SRP Section Number:** 1.8

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Location: Appendix O

Summary:

The submittal for review of the standard design shall include a description, analysis and evaluation of the interfaces between the submitted design and the balance of the nuclear power plant.

Potential Impact Number: 26035 **SRP Section Number:** 14.3.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Summary:

10 CFR 50.55a requires that structures, systems and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance to the safety function performed. In this regard, 10 CFR 50.55a identifies the required portions of the ASME Boiler and Pressure Vessel Code applicable to structures, systems and components important to safety.

Potential Impact Number: 26036 **SRP Section Number:** 14.3.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 1

APPENDIX II
Potential Impacts

Summary:

GDC 1 requires, in part, that structures, systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

Potential Impact Number: 26037 **SRP Section Number:** 14.3.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 14

Summary:

GDC 14 requires, in part, that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture.

Potential Impact Number: 26038 **SRP Section Number:** 14.3.3

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: Appendix A, GDC 30

Summary:

GDC 30 requires, in part, that components which are part of the reactor coolant pressure boundary should be designed, fabricated, erected, and tested to the highest quality standards practical.

Potential Impact Number: 26039 **SRP Section Number:** 14.3.7

APPENDIX II
Potential Impacts

Document: CODE OF FED. REGS 10CFR50
Title: DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES
Location: 50.48

Summary:

10 CFR 50.48 provides the Commission's requirements for fire protection. For plants licensed to operate after January 1, 1979, the rule states that the licensee shall meet the requirements of GDC 3 according to the provisions of the license. In general, GDC 3 requires that structures, systems, and components important to safety be designed to minimize the probability and effect of fires and explosions; that noncombustible and heat resistant materials be used whenever possible; and that fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety.

Potential Impact Number: 26040 **SRP Section Number:** 14.3.11

Document: CODE OF FED. REGS 10CFR52
Title: EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS
Locations: 52.47(b), 52.47(b), 52.97(b)(1)

Summary:

10 CFR 52.47(b) establishes the requirements for design certification applications for plants that are significantly different from currently licensed designs. For the purposes of the review conducted in accordance with SRP Section 14.3 and its associated subsections, this paragraph of the rule establishes the scope of the design certification to which the ITAAC review applies.

10 CFR 52.97(b)(1) sets forth the Commission's comprehensive requirements for ITAAC, which provide reasonable assurance that the facility has been constructed, and will be operated in conformance with the license, the provisions of the Atomic Energy Act and the Commission's rules and regulations.

Potential Impact Number: 26041 **SRP Section Number:** 14.3.9

APPENDIX II
Potential Impacts

Document: NUREG 0711
Title: HUMAN FACTORS ENGINEERING PROGRAM REVIEW MODEL

Summary:

NUREG 0711 provides guidance relavant to the review of human factors engineering design for nuclear power plants.