U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

5.4.13 ISOLATION CONDENSER SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - Organization responsible for boiling-water reactor systems

Secondary - None

I. AREAS OF REVIEW

The isolation condenser system (ICS) in a boiling-water reactor (BWR) is a safety system that removes decay heat after reactor isolation during power operations. By removing decay heat, the system also limits reactor pressure increase to below the safety relief valve (SRV) setpoint and prevents SRV actuation following a scram. Upon actuation, the ICS provides water inventory that was held in the system piping to the reactor vessel. The ICS also provides decay heat removal (DHR) from the reactor core via the reactor coolant system (RCS) by natural circulation for an extended period without needing operator action. Abnormal events include an inadvertent isolation of all main steamlines, loss of condenser vacuum, pressure regulator failure, loss of feedwater, loss of offsite power (LOOP) or station blackout (SBO), anticipated transient without scram (ATWS), or a loss-of-coolant accident (LOCA). For each of these events, the high pressure part of the emergency core cooling system (ECCS) provides a backup function to the ICS. This review of the ICS is performed to assure conformance with the requirements of General Design Criteria (GDC) 2, 4, 5, 17, 33, 34, 35, 36, 37, and 54. In some plant designs, the ICS, in conjunction with the gravity-driven cooling system (GDCS) or the automatic depressurization system (ADS), may be part of the ECCS. In such cases, the ECCS function of the ICS is reviewed under Standard Review Plan (SRP) Section 6.3. The

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ability of the ICS to perform these ECCS functions is reviewed to assure conformance to the requirements of 10 CFR 50.46. In addition, the ICS may provide necessary DHR to cope with a SBO. The capability of the ICS to perform this DHR function is reviewed to assure conformance with the requirements of 10 CFR 50.63.

The ICS consists of high-pressure, independent loops operating by natural circulation. Each loop contains a heat exchanger that condenses steam on its tube side and transfers heat to water that is contained in a large isolation condenser/passive containment cooling pool (IC/PCC) vented to the atmosphere. The condensed steam returns by gravity to the reactor vessel. The IC/PCC pool is located above and outside the containment. The fuel and auxiliary pools cooling system (FAPCS) replenishes the water in the IC/PCC pool during normal operation. A safety-related, independent FAPCS line provides emergency makeup from the fire protection system or from piping connections located in the yard external to the reactor building.

The staff will review the applicant's safety analysis report (SAR) or design control document (DCD) relating to the ICS. The specific areas of review are as follows:

1. System Design. The review of the ICS includes the system design bases, design criteria, components, support systems, and instrumentation and controls employed in the system. The review also includes process and instrumentation diagrams that describe the automatic and manual use of the ICS to limit RCS pressurization during power operation transients and DHR following reactor isolations during power operations.

2. Testing and Inspections. The reviewer examines the adequacy of the proposed preoperational and initial startup test program.

3. Technical Specifications. The reviewer evaluates the proposed technical specifications to assure that they are adequate with regard to limiting conditions of operation and periodic surveillance testing.

4. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this SRP section in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this SRP section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

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

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

Review Interfaces

Other SRP sections interface with this section as follows:

1. Review the design of systems that interface with the RCS with regard to the capability of the interfacing system to withstand full RCS pressure as part of the primary review responsibility for SRP Section 3.12.

2. Review the ECCS functions of the ICS as part of the primary review responsibility for SRP Section 6.3.

3. Review the ECCS functions of the ICS that relate to ATWS as part of the primary review responsibility for SRP Section 15.8.

In addition, the primary reviewer will coordinate, as necessary and by request, other reviewer evaluations that interface with the overall review of this SRP section as follows:

1. Review of the degree of separation of the ICS from the GDCS and ADS for protection against common mode failure of redundant systems, and review of protection against common mode failures from missiles as part of the review of SRP Sections 3.5.1.1, 3.5.1.2, 3.5.1.4, 3.5.1.5, 3.5.1.6, and 3.5.2.

2. Review of protection against flooding of the ICS and redundant equipment as part of SRP Section 3.4.1.

3. Review of protection against damage from pipe whip and jet impingement as part of SRP Sections 3.6.1 and 3.6.2, respectively.

4. Review of the proposed technical specifications as part of SRP Section 16.0.

5. Review of preoperational and critical startup test programs as part of SRP Section 14.2.

6. Review of proper seismic and quality group classification as part of SRP Sections 3.2.1 and 3.2.2.

7. Review of design adequacy of seismic Category I structure or building containing the ICS and equipment as part of SRP Sections 3.3.1, 3.3.2, 3.4.2, 3.5.3, 3.7.1, 3.7.2, 3.7.3, 3.8.1, 3.8.3, 3.8.4, 3.8.5, and 3.10.

8. Review of whether the ICS design is compatible with the containment system and containment heat removal capability and can be isolated from the containment as part of SRP Sections 6.2.2, 6.2.4, and 6.2.6.

9. Review of adequacy of controls and instrumentation of the ICS with regard to the required features of automatic actuation, remote sensing and indication, and remote control as part of the review of SRP Sections 7.3 and 7.4.

10. Review of adequacy and reliability of offsite and emergency onsite power, the sufficiency of battery capacity, the use of direct current power only to support operation of specified systems/subsystems, and the plant's capabilities to cope with a SBO as part of SRP Chapter 8.

11. Review of design and installation of the ICS to ascertain that they meet applicable codes and are adequate for the proper functioning as part of the primary review responsibility of SRP Section 3.9.3.

12. Review of in-service testing of pumps and valves for the ICS as part of SRP Section 3.9.6.

13. Review of environment qualification of ICS equipment as part of SRP Section 3.11.

14. Review of the proposed ITAAC regarding SSCs associated with the ICS as part of the of SRP Section 14.3.

15. Review of quality assurance requirements as part of SRP Chapter 17.

16. Review of instrumentation and control systems, as part of the primary review responsibility for SRP Sections 7.3 and 7.4, including evaluation of the adequacy of controls and instrumentation of the ICS with regard to the required features of automatic actuation, remote sensing and indication, and remote control.

17. Review of ICS materials to ensure a low probability of abnormal leakage, rapidly propagating failure, and gross rupture as part of SRP Section 6.1.1.

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

II. ACCEPTANCE CRITERIA

Requirements

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

1. GDC 2, as it relates to SSCs important to safety being designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform their safety functions.

2. GDC 4, as it relates to SSCs important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and being appropriately protected against dynamic effects that may result from equipment failures and from events and conditions outside the nuclear power unit.

3. GDC 5, as it relates to SSCs important to safety not being shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.

4. GDC 17, as it relates to onsite and offsite electric power systems being provided to permit functioning of SSCs important to safety.

5. GDC 33, as it relates to the system capability to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary so that fuel design limits are not exceeded.

6. GDC 34, as it relates to the system design being capable of transferring fission product decay heat and other residual heat from the reactor core such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded.

7. GDC 35, as it relates to the system design being capable of providing abundant emergency core cooling following any loss of coolant such that (1) fuel and clad damage that could interface with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts..

8. GDC 36, as it relates to the ECCS being designed to permit appropriate periodic inspection of important components.

9. GDC 37, as it relates to the ECCS being designed to permit appropriate periodic pressure and functional testing to assure (1) structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

10. GDC 54, as it relates to piping systems penetrating primary containment being provided with leak detection and isolation capabilities.

11. 10 CFR 50.46, as it relates to the calculated cooling performance being calculated in accordance with an acceptable evaluation model and being sufficient to provide assurance that the most severe postulated LOCAs are calculated.

12. 10 CFR 50.63, as it relates to design provisions to support the plant’s ability to withstand for a specified duration and recover from an SBO.

13. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;

14. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

1. GDC 2 provides the design bases for plant structures, systems and components (SSCs) for protection from natural phenomena, i.e., earthquakes, tornados, hurricanes, floods, tsunamis, and seiche without loss of capability or loss of safety function. In the application, consideration should be given to the historical data of the phenomena, potential combination of normal and accident conditions, and the importance of the safety functions performed by the SSCs. With respect to GDC 2, the application should demonstrate that SSCs are designed to withstand the effects of the above listed phenomena without loss of integrity or capability to perform their safety function. The application should demonstrate that all quality assurance requirements of 10 CFR 50, Appendix B have been applied to the activities affecting safety related functions of these SSCs.

2. With respect to GDC 4, the application should demonstrate compatibility of components with environmental conditions that are acceptable by compliance with the applicable provisions of the ASME Code and by compliance with the positions of Regulatory Guide 1.44, “Control of the Use of Sensitized Stainless Steel.”

   Regulatory Guide 1.44 contains staff positions related to unstabilized austenitic stainless steel of the AISI Type 3XX series used for components of the RCPB. Positions related to BWR piping materials, including verification of nonsensitization of the material by an approved test, are described in Attachment A to Generic Letter 88-01. The technical bases for the positions provided in Generic Letter 88-01 and similar recommendations related to minimizing stress corrosion cracking in susceptible piping of BWRs are detailed in NUREG-0313, Revision 2.

   Upon resolution of GSI-191, the review should include consideration of the resolution of this issue.

3. Pursuant to GDC 5, SSCs that are important to safety should 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, the event of an accident in one

unit, an orderly shutdown and cooldown of the remaining units. With respect to GDC 5, the application should demonstrate that the ICS design's ability to accomplish these safety-related functions is not compromised for each unit regardless of equipment failures or other events that may occur in another unit.

4. With respect to GDC 17, the application should demonstrate conformance with the guidelines in RG 1.93 with respect to providing onsite and offsite electric power systems to permit functioning of SSCs important to safety to ensure their safety function assuming either power system is not functioning. The application should demonstrate 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 and possible loss of power generated by the nuclear power plant.

5. With respect to GDC 33, the application should demonstrate that the system can supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary and in the event that either onsite or offsite ac power is unavailable.

6. With respect to GDC 34, the application should demonstrate that the system can transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design pressure of the reactor coolant pressure boundary are not exceeded. The application should demonstrate that the ICS provides the capability for decay heat removal, and that component redundancy, leak detection and isolation capabilities are provided. ICS operation should be assured with a single active failure including electric power.

7. With respect to GDC 35, the application should demonstrate that the system design is capable of providing abundant emergency core cooling following any loss of coolant such that: (1) fuel and clad damage that could interface with continued effective core cooling is prevented and, (2) clad metal-water reaction is limited to negligible amounts. The application should demonstrate that the ICS provides water inventory and DHR capability following reactor shutdown, and that component redundancy, leak detection and isolation capabilities are provided. ICS operation should be assured with a single active failure including electric power.

8. With respect to GDC 36, the application should demonstrate that the ECCS is designed in a manner permitting appropriate inspection of important components, to assure the readiness, integrity, and capability of the system.

9. With respect to GDC 37, the application should demonstrate that the ICS is designed to permit appropriate periodic pressure and functional testing to assure: (1) structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

10. With respect to GDC 54, the application should demonstrate that piping systems penetrating primary containment are provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect the importance to safety of isolating these piping systems. An acceptable design of such piping systems must have the capability to test periodically the operability of the isolation valves and associated apparatus and to determine whether valve leakage is within acceptable limits.

11. With respect to 10 CFR 50.46, the application should demonstrate that an acceptable evaluation model is used so that the cooling performance is sufficient to assure that the most severe postulated LOCA is calculated. The application should demonstrate that: 1) the maximum fuel element cladding temperature shall not exceed 2200°F, 2) the maximum total cladding oxidation shall nowhere exceed 0.17 times the total cladding thickness before oxidation, 3) the total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react, 4) changes in core geometry shall be such that the core remains amenable to cooling, and 5) after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

12. For 10 CFR 50.63, the application should demonstrate compliance with RG 1.155, which identifies acceptable methods to support the plant's ability to withstand for a specified duration and recover from an SBO.

Technical Rationale

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

1. Compliance with GDC 2 requires that nuclear power plant SSCs important to safety be designed to withstand the effects of seismic events and other natural phenomena without losing their capability to perform their safety functions. The subject SSCs 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. Because the ICS performs ECCS functions, the ICS, and structures housing the ICS, must be capable of withstanding the effects of natural phenomena.

   A seismic design classification system has been developed for identifying those plant features that should be designed to withstand the effects of a safe-shutdown earthquake (SSE). Regulatory Guide 1.29, Position C.1, states that systems required for safe shutdown, including their foundations and supports, are designated as Seismic Category I and should be designed to withstand the effects of the SSE and remain functional. Compliance with Regulatory Guide 1.29 provides assurance that the ICS will perform its intended safety function in the event of an earthquake.

2. GDC 4 requires that SSCs important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. SSCs should also 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. The ICS provides DHR capacity, and the dynamic effects of water hammer could degrade system effectiveness. Compliance with GDC 4 assures that the ICS will remain functional and provide essential cooling necessary for DHR.

3. GDC 5 prohibits the sharing of SSCs 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. The ICS provides DHR capability after the reactor is shut down and isolated. The reactor water cleanup/shutdown cooling (RWCU/SDC) system and other engineered safety features (ADS, passive containment cooling system (PCCS), and GDCS) back up the ICS. The ICS should be designed so that the ability to accomplish these safety-related functions is not compromised for each unit regardless of equipment failures or other events that may occur in another unit. Meeting the requirements of GDC 5 provides assurance that the unacceptable effects of equipment failures or other events occurring in one unit of a multiunit site will not propagate to the unaffected unit(s).

4. GDC 17 establishes requirements for providing onsite and offsite electric power systems to permit functioning of SSCs important to safety to ensure their safety function assuming either power system is not functioning. The safety function for each electric power system is 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 ICS is a safety system and must therefore be able to meet its safety function in the event of postulated accidents and with a single failure.

5. GDC 33 specifies requirements for a system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary and be designed to assure safety function performance with either onsite or offsite alternating current (ac) power not available. The ICS is designed as a limited high-pressure reactor coolant makeup system with flow rates sufficient to meet the criteria for small breaks in the reactor coolant pressure boundary without ac power. The RWCU/SDC system and other engineered safety features (ADS, PCCS, and GDCS) back up the ICS. Compliance with GDC 33 assures that specified acceptable fuel design limits are not exceeded as a result of reactor coolant pressure boundary leakage, rupture of small piping, or other small components that are part of the boundary.

6. GDC 34 establishes requirements for a system to transfer fission product decay heat and other residual heat from the reactor core. The ICS provides the capability for DHR. Compliance with GDC 34 precludes fuel damage or reactor coolant pressure boundary overpressurization in the event of anticipated operational occurrences that would adversely affect functions of systems that provide normal heat removal from the reactor core.

7. GDC 35 establishes requirements to provide abundant emergency core cooling following any loss of coolant to prevent fuel damage that could interfere with continued effective core cooling and to limit clad metal-water reactions to negligible amounts. The ICS provides limited additional water inventory and DHR capability following reactor shutdown and isolation. Compliance with GDC 34 precludes fuel damage in the event of anticipated operational occurrences that would adversely affect functions of systems that provide normal heat removal from the reactor core.

8. GDC 36 requires the ECCS to be designed to permit appropriate inspection of important components. The ICS is an ECCS and therefore must be designed in a manner to permit inspection of components to assure the integrity and capability of the system.

9. GDC 37 establishes requirements for the ECCS to be designed to permit appropriate periodic pressure and functional testing. The ICS is an ECCS and therefore must be designed to permit pressure and functional testing of components to assure the structural and leaktight integrity of its components, operability and performance of the active components of the system, and operability of the system as a whole under conditions as close to design as practical, as well as the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, and the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

10. GDC 54 requires that piping systems that penetrate primary reactor containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems must be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits. Piping in the ICS passes through the containment boundary and is provided with isolation valves and integrity verification capabilities. Compliance with GDC 54's containment isolation and leak detection requirements provides a high level of assurance that the containment will perform its safety function in the event of a postulated accident and will maintain the capability to prevent a significant uncontrolled release of radioactivity.

11. 10 CFR 50.46 establishes acceptance criteria for ECCS for light-water nuclear power reactors as it relates to the calculated cooling performance being calculated in accordance with an acceptable evaluation model and sufficient to provide assurance that the most severe postulated LOCAs are calculated. The ICS is an ECCS. Therefore, its emergency core cooling capability must be calculated, in accordance with an acceptable evaluation model, for a number of postulated LOCAs to provide assurance that the more severe postulated accidents are examined.

12. 10 CFR 50.63 requires that a plant provide sufficient capacity and capability to ensure that the core is cooled in the event of an SBO for a determined duration. The ICS provides DHR from the reactor core. The design capability of the plant, which allows it to operate regardless of ac power source availability, enables continued performance of

the DHR function to support the plant in coping with an SBO. Regulatory Guide 1.155 identifies acceptable methods for complying with the requirements of 10 CFR 50.63. Compliance with 10 CFR 50.63 provides assurance that the ICS is capable of performing its intended function in the event of an SBO.

III. REVIEW PROCEDURES

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

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

The procedures below are used during a DC review to assure that the design criteria and bases and the preliminary design set forth in the preliminary SAR meet the acceptance criteria given in Subsection II of this SRP.

For the review of a COL application, the procedures are also used to verify that the initial design criteria and bases have been appropriately implemented in the final design set forth in the final SAR. The COL and DC review also include the proposed technical specifications to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

Upon request from the primary reviewer, other staff will provide input for the areas of review stated in Subsection I. The primary reviewer obtains and uses such input as necessary to assure that this review is complete.

For DC and COL applications submitted under Part 52, the level of information reviewed should be consistent with that of a FSAR submitted in an OL application. However, verification that the as-built facility conforms to the approved design is performed through the ITAAC process.

1. Using the ICS operating criteria specified in the applicant's SAR (or DCD), the reviewer confirms that the ICS can maintain coolant inventory in the reactor vessel and remove reactor decay heat following shutdown and isolation, thereby keeping the core covered and assuring cladding integrity. This determination is based on engineering judgment and independent calculations (where deemed necessary), using information as specified in items 2 and 3 below. The reviewer verifies that the decay heat loads used in the ICS analyses are applicable and consistent with SRP Section 4.2.

2. Using the description of the ICS given in the applicant's DCD, including component lists and performance specifications, the reviewer determines that the ICS piping and instrumentation will allow the system to operate as intended. This is accomplished by reviewing the process and piping and instrumentation diagrams (P&IDs) to confirm that piping arrangements permit the required flowpaths to be achieved and that sufficient process sensors are available to measure and transmit necessary information.

3. Using the comparison tables in the applicant's DCD, where applicable, the reviewer compares the ICS to the designs and capacities of such systems in similar plants to determine that there are no unexplained departures from previously reviewed plants. Where possible, comparisons should be made with actual performance data from similar systems in operating plants.

4. The reviewer examines the P&IDs and equipment layout drawings for the ICS and redundant or backup systems (RWCU/SDC, ADS, PCCS, and GDCS) to ensure that the systems are physically separated and can function independently.

5. The review determines whether adequate control and monitoring information is available to allow the operator to actuate the system manually within allowable time.

6. The reviewer shall confirm that automatic actuation and remote-manual valve controls are capable of performing the functions required and that sensor and monitoring provisions are adequate. The instrumentation and controls of the ICS and redundant or backup systems (RWCU/SDC, ADS, PCCS, and GDCS) have sufficient redundancy to satisfy the single failure criterion.

7. The reviewer ascertains that ICS operation is not dependent on ac power sources, and that sufficient battery capacity exists to permit operation of the ICS for the period needed without the availability of ac power.

8. The reviewer verifies that essential ICS components are designated seismic Category I.

9. The reviewer verifies that the applicant's proposed preoperational and initial startup test programs are in accordance with the guidance of Regulatory Guide 1.68. At the COL stage, the reviewer confirms that the applicant has provided sufficient information to identify the test objectives, methods of testing, and test acceptance criteria (see Positions C.2, C.3, and C.4 of Regulatory Guide 1.68). The reviewer verifies that the proposed test programs will provide reasonable assurance that the ICS will perform its safety function. As an alternative to this detailed evaluation, the reviewer may compare the ICS design to that of previously reviewed plants. If the design is essentially identical and if the proposed test programs are essentially the same, the reviewer may conclude that the proposed test programs are adequate for the ICS. If the ICS differs significantly from that of previously reviewed designs, the impact of the proposed changes on the required preoperational and initial startup testing programs are reviewed at the COL stage. This effort should particularly evaluate the need for any special design features required to perform acceptable test programs.

10. The reviewer evaluates, in conjunction with the reviewer primarily responsible for SRP Section 16.0, the applicant's proposed plant technical specifications of the DCD as follows:

    A. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when system equipment is inoperable as a result of repairs and maintenance.

B. Verify that the frequency and scope of periodic surveillance testing are adequate.

11. The reviewer confirms, in conjunction with the reviewer responsible for the civil and structural engineering portions of SRP Sections 3.3.1, 3.3.2, 3.4.2, 3.5.1.1, 3.5.1.2, 3.5.1.4, 3.5.1.5, 3.5.1.6, 3.5.2, 3.5.3, 3.7.1, 3.7.2, 3.7.3, 3.8.1, 3.8.3, 3.8.4, 3.8.5, and 3.10 that the ICS is housed in a structure that provides adequate protection against wind, tornadoes, floods, and missiles, as appropriate, based on a review of its design and design criteria.

12. The reviewer verifies the automatic and manual actions necessary for proper functioning of the ICS and redundant or backup systems (RWCU/SDC, ADS, PCCS, and GDCS) for completeness and practicality when used for residual heat removal per the criteria of item II.K.1.22 of NUREG-0737.

13. The reviewer examines the ICS break detection provisions to determine that the system is protected against spurious trip signals. For plants using a time delay for this protection, the reviewer verifies that the minimum and maximum expected response times are adequate.

    For plants that do not use a time delay for spurious isolation protection, the reviewer verifies that the applicant has provided proper justification for the design, and that the applicant's test program and test results are adequate.

14. The reviewer confirms that the ICS can withstand a LOOP to its support systems for their period of need per the criteria of item II.K.3.24 of NUREG-0737.

15. The reviewer confirms, per the criteria of item II.K.3.13 of NUREG-0737, that analyses have been provided or referenced to determine the need to separate the initiation parameters of the ICS and redundant or backup systems (RWCU/SDC, ADS, PCCS, and GDCS). On the basis of these study results, the reviewer evaluates the ICS design for appropriate provisions. In addition, the reviewer verifies that the ICS provides automatic restart capability.

16 The reviewer confirms that the plant can be brought to a safe-shutdown condition as defined in Section C of SECY-94-084.

17. The reviewer evaluates the adequacy of the ICS design features that have been provided to prevent damaging water (steam) hammer from such mechanisms as voided discharge lines, water entrainment, and steam bubble collapse. If the normal water supply is above the discharge lines, voided lines are prevented by proper vent location, filling, and venting procedures. The vents should be tested on a periodic basis and located for ease of operation to prevent the possibility of voiding the liquid-filled discharge lines. NUREG-0927 provides guidance for water hammer prevention and mitigation.

18. The reviewer confirms that the ICS capability is sufficient with respect to the plant's ability to cope with, and recover from, an SBO of a specified duration by determining compliance with Regulatory Guide 1.155, Positions C.3.2, C.3.3, and C.3.5, as they relate to the design of the ICS. This review is coordinated with the review of the SBO event under SRP Section 8.3.1.

19. The reviewer verifies that the test results for containment isolation valves indicate that the valves will isolate under expected conditions, as discussed in Generic Letter 89-10, Supplement 3.

20. The reviewer verifies that a leakage reduction program has been implemented and that the program and ICS design meet the criteria of NUREG-0737, item III.D.1.1.

For standard DC reviews under 10 CFR Part 52, the reviewer should follow the procedures above, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard DCD, including ITAAC, site interface requirements, and COL action items, meet the acceptance criteria given in Subsection II. SRP Section 14.3 contains procedures for the review of the DCD for the standard design, including the site parameters, interface criteria, and ITAAC.

For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

IV. EVALUATION FINDINGS

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

> The ICS includes the piping, valves, instrumentation, and controls used to remove decay heat from the reactor core whenever it is shut down and isolated. Certain engineered safety features and backup systems (RWCU/SDC, ADS, PCCS, and GDCS) provide redundancy for this system. The scope of review of the ICS for the plant included P&IDs, equipment layout drawings, and functional specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the ICS, analysis of the adequacy of the criteria and bases, and the conformance of the design to these criteria and bases.

The staff concludes that the ICS is acceptable and meets the requirements of GDC 2, 4, 5, 17, 33, 34, 35, 36, 37, 54, 10 CFR 50.46, 10 CFR 50.63, 10 CFR 52.47(a)(1)(vi), and 10 CFR 52.97(b)(1). This conclusion is based on the following:

1. The applicant has met the requirements of (cite Reg.) with respect to (state limits of review) in the following manner: (as an example, use one or more of the following as applicable)

    A. Meeting the regulatory position in Regulatory Guide________.

    B. Providing and meeting an alternative method to the regulatory position in Regulatory Guide________ that the staff has reviewed and found to be acceptable.

    C. Meeting the regulatory position in BTP ________.

    D. The calculational method used by the applicant for (state) has been previously reviewed by the staff and found acceptable; the staff has reviewed the key parameters in this case and found them to be acceptable.

    E. The applicant has met the requirements of (industry standard—number and title) that has been reviewed by the staff and determined to be appropriate for this application.

Repeat the above discussion for each requirement listed in Subsection II.1 - II.12.

2. The staff concludes that the ICS and redundant or backup systems (RWCU/SDC, ADS, PCCS, and GDCS) are found capable of removing core decay heat following reactor shutdown and isolation so that sufficient coolant inventory is maintained in the reactor vessel to keep the core covered and ensure cladding integrity. The staff concludes that this capability is available even with a LOOP and with a single active failure.

    The capability and capacity of the ICS are sufficient with respect to the plant's ability to cope with, and recover from, an SBO of a specified duration by determining adherence with Regulatory Guide 1.155, Positions C.3.2, C.3.3, and C.3.5, as they relate to the design of the ICS. (In addition, SRP Section 6.2 discusses conformance with GDC 55, 56, and 57 regarding containment isolation. SRP Sections 3.3 through 3.6 discuss conformance with GDC 2 and 4 for protection against natural phenomena, environmental hazards, and potential missiles. SRP Section 8.4 discusses conformance with the requirements of 10 CFR 50.63 for SBO.)

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this SRP section.

In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications submitted six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Dynamic Effects Design Bases."

3. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

4. 10 CFR Part 50, Appendix A, General Design Criterion 17, "Electric Power Systems."

5. 10 CFR Part 50, Appendix A, General Design Criterion 33, "Reactor Coolant Makeup."

6. 10 CFR Part 50, Appendix A, General Design Criterion 34, "Residual Heat Removal."

7. 10 CFR Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling."

8. 10 CFR Part 50, Appendix A, General Design Criterion 36, "Inspection of Emergency Core Cooling System."

9. 10 CFR Part 50, Appendix A, General Design Criterion 37, "Testing of Emergency Core Cooling System."

10. 10 CFR Part 50, Appendix A, General Design Criterion 54, "Systems Penetrating Containment."

11. 10 CFR 50, Appendix B, "Quality Assurance Criteria For Nuclear Power Plants And Fuel Reprocessing Plants."

12. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."

13. 10 CFR 50.63, "Loss of All Alternating Current Power."

14. 10 CFR 52.47, "Contents of Applications."

15. 10 CFR 52.97, “Issuance of Combined Licenses.”

16. 10 CFR 100, “Reactor Site Criteria.”

17. Regulatory Guide 1.29, “Seismic Design Classification.”

18. Regulatory Guide 1.44, “Control of the Use of Sensitized Stainless Steel.”

19. Regulatory Guide 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants.”

20. Regulatory Guide 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition,” Revision 3.

21. Regulatory Guide 1.93, “Availability of Electric Power Sources.”

22. Regulatory Guide 1.155, “Station Blackout.”

23. NUREG-0313, Revision 2, “Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping.”

24. NUREG-0718, “Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License,” Revision 2.

25. NUREG-0737, “Clarification of TMI Action Plan Requirements,” November 1980.

26. SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” March 28, 1994.

27. NUREG-0927, Revision 1, “Evaluation of Water Hammer Occurrences in Nuclear Power Plants,” March 1984.

28. NRC Letter to All Licensees of Operating Boiling Water Reactors (BWRs) and Holders of Construction Permits for BWRs, (Generic Letter 88-01), dated January 25, 1988, “NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping.”

29. NRC Letter to All Licensees and Applicants, (Generic Letter 89-10), dated June 28, 1989, and Supplements 1 through 6, “Safety Related Motor-Operated Valve Testing.”
