U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

6.3 EMERGENCY CORE COOLING SYSTEM

REVIEW RESPONSIBILITIES

Primary - **Organizations responsible for the review of the BWR and PWR emergency core cooling system.**

Secondary - **Organizations responsible for the review of other systems and technical areas related to the BWR and PWR emergency core cooling system.**

I. <u>AREAS OF REVIEW</u>

The reviewer reviews the information presented in the applicant's safety analysis report (SAR) regarding the emergency core cooling system (ECCS). The major elements of the review are:

The specific areas of review are as follows:

1. The design bases for the ECCS are reviewed to assure that they satisfy applicable regulations, including the general design criteria and the requirements of 10 CFR 50.46 regarding ECCS acceptance criteria.

2. The design basis for the automatic depressurization systems (ADS) are also reviewed for compliance with TMI Action Plan Items and associated guidance. This applies to BWRs and the advanced passive reactors (both PWRs and BWRs).

3. For advanced passive reactors which rely on gravitational head to provide ECCS injection to the RCS, the RCS must be designed with an ADS such that the available gravitational head is sufficient to provide adequate core cooling when depressurized.

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4. For advanced reactors which rely on passive safety-related systems and equipment to automatically establish and maintain safe-shutdown conditions for the plant, these passive safety systems must be designed with sufficient capability to maintain safe-shutdown conditions for 72 hours, without operator actions and without non-safety-related onsite or offsite power.

5. The design of the ECCS is reviewed to determine that it is capable of performing all of the functions required by the design bases.

6. The preoperational and initial startup test programs for the ECCS are reviewed to determine if they are sufficient to confirm the performance capability of the ECCS. The need for special design features to permit the performance of adequate test programs should also be reviewed.

7. The proposed technical specifications (TSs) are reviewed to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

8. Inspection, Test, Analysis, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the applicant's proposed information on the ITAAC associated with the systems, structures, and components (SSCs) related to this SRP section is reviewed in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria - Design Certification." The staff recognizes that the review of ITAAC is performed after review of the rest of this portion of the application against acceptance criteria contained in this SRP section. Furthermore, the ITAAC are reviewed to assure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

9. COL Action Items and Certification Requirements and Restrictions. COL action items may be identified in the NRC staff's final safety evaluation report (FSER) for each certified design to identify information that COL applicants must address in the application. Additionally, DCs contain requirements and restrictions (e.g., interface requirements) that COL applicants must address in the application. For COL applications referencing a DC, the review performed under this SRP section includes information provided in response to COL action items and certification requirements and restrictions pertaining to this SRP section, as identified in the FSER for the referenced certified design.

Review Interfaces

The listed SRP sections interface with this section as follows:

1. Evaluation of the capability of the low-pressure portions of the ECCS that interface with the RCS to withstand full RCS pressure is performed under SRP Section 3.12.

2. Evaluation of the ability of the ECCS to mitigate the consequences of a spectrum of loss-of-coolant accidents is performed under SRP Section 15.6.5.

3. Evaluation of the capability of the new applicant's ADS systems and the PWR core melt stabilization system to support mitigation of severe accidents is performed under SRP Section 19.2.

4. Review of the effects of pipe breaks outside containment on ECCS is performed under SRP Section 3.6.1. This review includes the evaluation of the effect of pipe whip, jet impingement forces, and environmental conditions.

5. Review of the acceptability of, and environmental qualification test program for, ECCS equipment is performed under SRP Section 3.11. This review includes consideration of the post-accident environmental design and source term considerations described in TMI action plan item II.B.2 and II.K.3.24 of NUREG-0737.

6. Review of the capability of those auxiliary systems essential for ECCS operation (service water system, component cooling system, ultimate heat sink, and condensate storage facility) is performed under SRP Sections 9.2.1, 9.2.2, 9.2.5, and 9.2.6. The evaluations include portions of the power conversion systems (e.g., steam supply lines, steam generators, feedwater systems) which interface with the reactor coolant system in such a way as to influence the course of a loss-of-coolant accident (LOCA) for a particular plant.

7. Review of the capability and design of the pneumatic supply system for those applicants which use a pneumatic supply for the ADS function is performed under SRP Section 9.3.1.

8. Review of the adequacy of ECCS-associated controls and instrumentation with regard to the features of automatic actuation, remote sensing and indication, and remote control is performed under SRP Section 7.3. This review includes consideration of the instrumentation described in TMI action plan items II.D.3 (positive indication in the control room of flow in the discharge pipe of RCS relief and safety valves) and II.F.2 (instrumentation which provides an unambiguous, easy-to-interpret indication of inadequate core cooling) of NUREG-0737.

9. Review of the inlet design of the containment sump ECCS suction screen is performed under SRP Section 6.2.2. The review is to assure that containment sumps provide a reliable, long-term recirculation cooling capability and that ECCS pump performance will not be adversely affected by post-LOCA conditions impacting the sumps and that the system operation and performance is consistent with the intent of Regulatory Guide 1.82.

10. Review of the adequacy of the containment isolation is performed under SRP Section 6.2.4. This review is, in part, to assure that portions of the ECCS penetrating the containment barrier are designed with acceptable isolation features to maintain containment integrity for all operating conditions, including accidents.

11. Review of the adequacy of the power supply for the ECCS is performed under SRP Sections 8.1, 8.2, 8.3.1, and 8.3.2. In addition, reviews of the plant's overall capabilities to withstand or cope with, and recover from a Station Blackout (SBO) is performed under SRP Section 8.4. The review should coordinate with the review of the ECCS if the system is required to ensure adequate core cooling as required by 10 CFR 50.63 and the guidance of Regulatory Guide 1.155.

12. Review of the seismic and quality group classifications for the ECCS is performed under SRP Section 3.2.1 and 3.2.2.

13. Review of the criteria used for postulating the effects of pipe breaks both inside and outside containment on ECCS is performed under SRP Section 3.6.2. This review includes criteria used for postulating the effects of pipe whip, jet impingement forces, and any related environmental conditions.

14. Review of the loading combinations (operational, LOCA, seismic, and thermal stratification loads) and the associated stress limits is performed under SRP Section 3.9.3.

15. Review of the adequacy of the inservice testing program for pumps and valves is performed under SRP Section 3.9.6. This review is to assure that the ECCS piping and component configurations allow for full flow testing of safety related pumps and check valves and provisions are made to allow for the use of advanced techniques to detect degradation and to monitor system performance.

16. Review of the structures housing the ECCS for the proper seismic classification is performed under SRP Sections 3.8.1, 3.8.2, and 3.8.3.

17. Review of the applicable inservice inspection requirements is performed under SRP Section 6.6.

18. Review of the thermal shock effect of water injected into the primary coolant system from the ECCS is performed, on a generic basis, under SRP Sections 5.3.2 and 5.3.3.

19. Review of compliance of the NUREG-0737 item II.K.2.15 requirements is performed under SRP Section 5.4.2.1. This review assures that B&W once-through steam generator tubes are designed with sufficient margin to assure that if the tubes are stressed under slug flow conditions, mechanical integrity will be maintained.

20. Review of the proposed preoperational and initial startup test programs is performed under SRP Section 14.2 to assure that they are consistent with the intent of Regulatory Guides 1.68.

21. Review of compliance of Task Action Plan items I.C.2 and I.C.6 of the NUREG-0737 requirements regarding procedures to assure that system operability status is known is performed under SRP Section 13.5.1.

22. Review of quality assurance is performed under SRP Sections 17.1, 17.2, and 17.3.

23. Review of compliance with Task Action Plan items II.B.2 of NUREG-0737 and NUREG-0718 requirements regarding radiation and shielding design review is performed under SRP Sections 12.1 through 12.5 to assure adequate access to vital areas and protection of safety equipment.

24. Review of technical specifications is performed under SRP Section 16.0.

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

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criterion (GDC) 2 as it relates to the seismic design of structures, systems, and components (SSCs) whose failure could cause an unacceptable reduction in the capability of the ECCS to perform its safety function.

2. GDC 4 as it relates to dynamic effects associated with flow instabilities and loads (e.g., water hammer).

3. GDC 5 as it relates to SSCs important to safety shall not be shared among nuclear power units unless it can be demonstrated that sharing will not impair their ability to perform their safety function.

4. GDC 17 as it relates to the design of the ECCS having sufficient capacity and capability to assure that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded during anticipated operational occurrences and that the core is cooled during accident conditions.

5. GDC 27 as it relates to the system design having the capability to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

6. GDCs 35, 36, and 37 as they relate to the ECCS being designed to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling, to permit appropriate periodic inspection of important components, and to permit appropriate periodic pressure and functional testing.

7. 10 CFR 50.46, in regard to the ECCS being designed so that its cooling performance is in accordance with acceptable evaluation models, which identifies and accounts for uncertainties in the analysis method and inputs; alternatively, an ECCS evaluation model may be developed in conformance with Appendix K to 10 CFR Part 50.

8. TMI Action Plan item II.K.3.18 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(vii) for applicants subject to 10 CFR 50.34(f), with respect to eliminating the need for manual actuation of the BWR ADS to assure adequate core cooling.

9. TMI Action Plan item II.K.3.21 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(viii) for applicants subject to 10 CFR 50.34(f), with respect to studying the design of BWR 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.

10. TMI Action Plan item II.K.3.28 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(x) for applicants subject to 10 CFR 50.34(f), with respect to BWR ADS-associated equipment and instrumentation being capable of performing their intended functions during and following an accident, while taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves.

11. TMI Action Plan item II.K.3.45 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(xi) for applicants subject to 10 CFR 50.34(f), with regard to providing an evaluation of depressurization methods, other than full actuation of the ADS, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown for BWRs.

12. TMI Action Plan item III.D.1.1 of NUREG-0737, equivalent to 10 CFR 50.34(f)(2)(xxvi) for applicants subject to 10 CFR 50.34(f), with respect to the provisions for a leakage detection and control program to minimize the leakage from those portions of the ECCS outside of the containment that contain or may contain radioactive material following an accident.

13. TMI Action Plan item II.K.3.16 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(vi) for applicants subject to 10 CFR 50.34(f), with regard to providing an evaluation of methods to reduce challenges and failures of reactor coolant system relief valves for BWRs.

14. TMI Action Plan item II.K.3.24 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(ix) for applicants subject to 10 CFR 50.34(f), with respect to the adequacy of space cooling for long-term operation of HPCI and RCIC systems for BWRs to maintain the operating environment within allowable limits.

15. TMI Action Plan item II.D.3 of NUREG-0737, equivalent to 10 CFR 50.34(f)(2)(xi) for applicants subject to 10 CFR 50.34(f), with respect to the requirements that reactor coolant system relief and safety valves be provided with a positive indication in the control room of flow in the discharge pipe.

16. TMI Action Plan item II.F.2 of NUREG-0737, equivalent to 10 CFR 50.34(f)(2)(xviii) for applicants subject to 10 CFR 50.34(f), with respect to the requirement that instrumentation or controls provide an unambiguous, easy-to-interpret indication of inadequate core cooling.

17. 10 CFR 52.47(a)(1)(vi), which requires ITAAC (for design certification) sufficient to assure that the SSCs in this area of review will operate in accordance with the certification.

18. 10 CFR 52.97(b)(1), which requires ITAAC (for combined licenses) sufficient to assure that the SSCs in this area of review have been constructed and will be operated in conformity with the license and the Commission'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 review described in Subsection I of 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. In regard to the ECCS acceptance criteria of 10 CFR 50.46, the five major performance criteria deal with:

   A. Peak cladding temperature.

   B. Maximum calculated cladding oxidation.

   C. Maximum hydrogen generation.

   D. Coolable core geometry

   E. Long-term cooling.

   Guidance, procedures and methods that are acceptable for meeting the requirements for a realistic or best-estimate evaluation model for ECCS performance can be found in Regulatory Guide 1.157. This method must identify and account for uncertainties in the analysis method and inputs such that there is a high level of probability that the acceptance criteria is not exceeded (addresses Generic Issue C-4). Alternatively, Appendix K to 10 CFR Part 50 contains guidance for conservative ECCS evaluation models. These areas are reviewed as a part of the effort associated with the LOCA analysis (SRP Section 15.6.5). However, the impact of various postulated single failures on the operability of the ECCS, ECCS response times, break locations (including ECCS break locations), and break sizes impacting ECCS capabilities are evaluated under this SRP section.

2. The ECCS must meet the requirements of GDC 35. The system must have alternate sources of electric power, as required by GDC 17, and must be able to withstand a single failure. The ECCS should retain its capability to cool the core in the event of a failure of any single active component during the short term immediately following an accident, or a single active or passive failure during the long-term recirculation cooling phase following an accident.

   A passive failure in a fluid system is a breach in the fluid pressure boundary or mechanical failure that adversely affects a flowpath. SECY-94-084 states the approved position that passive advanced light-water reactor designs need not assume passive component failures in addition to the initiating failure in the application of single-failure criterion to assure safety of the plant. In addition, the staff considers, on a long-term basis, passive component failures in fluid as potential accident initiators, in addition to initiating events. Check valves in the passive safety systems (except those for which proper function can be demonstrated and documented) are considered components subject to single-failure consideration.

3. The ECCS must be designed to permit periodic inservice inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, piping, pumps, and valves in accordance with the requirements of GDC 36. The ECCS must be designed to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation, as required by GDC 37.

4. The combined reactivity control system capability associated with ECCS must meet the requirements of GDC 27 and should conform to the recommendation of Regulatory Guide 1.47. The primary mode of actuation for the ECCS must be automatic, and actuation must be initiated by signals of suitable diversity and redundance. Provisions should also be made for manual actuation, monitoring, and control of the ECCS from the reactor control room.

5. The design of the ECCS should conform to the recommendations of Regulatory Guide 1.1.

6. Design features and operating procedures, designed to prevent damaging water hammer due to such mechanisms as voided discharge lines and water entrainment in steam lines shall be provided, in order to meet the requirements of GDC 4.

7. The design of those portions of the system which are not safety related, whose failures could have an adverse effect on the ECCS system, must be in accordance with GDC 2, and acceptance is based on meeting Position C2 of Regulatory Guide 1.29. Also see SECY-94-084 for policy and technical issues associated with the regulatory treatment of non-safety systems in passive plant designs.

8. Interfaces between the ECCS and component or service water systems must be such that operation of one does not interfere with, and provides proper support (where required) for, the other. In relation to these and other shared systems, e.g., residual heat removal (RHR) and containment heat removal systems, the ECCS must conform to GDC 5.

9. The requirements of Task Action Plan Item II.K.3(15) of NUREG-0737 and NUREG-0718, which involves isolation of HPCI and RCIC for BWR plants, must also be satisfied.

10. The requirements and guidance regarding ECCS outage times and reports on ECCS unavailability, contained in Task Action Plan Item II.K.3.17, and Generic issue B-61, must also be satisfied.

11. 10 CFR 52.47(a)(1)(vi) specifies that the application of a design certification should contain proposed ITAAC necessary and sufficient to assure the plant is built and will operate in accordance with the design certification. 10 CFR 52.97(b)(1) specifies that the COL identifies the ITAAC necessary and sufficient to assure that the facility has been constructed and will be operated in conformity with the license. SRP 14.3 provides guidance for reviewing the ITAAC. The requirements of 10 CFR 52.47(a)(1)(vi) and 10 CFR 52.97(b)(1) will be met, in part, by identifying inspections, tests, analyses, an acceptance criteria of the top-level design feature of the ECCS in the design certification application and combined license, respectively.

In addition to the above criteria, the acceptability of the ECCS may be based on the degree of design similarity with previously approved plants.

Technical Rationale

The technical rationale for application of these requirements and/or SRP 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 SSCs important to safety be designed to withstand the effects of natural phenomena without the loss of capability to perform their safety functions.

   GDC 2 is applicable because the ECCS is relied upon to provide sufficient emergency core cooling flow to protect the integrity of the reactor core during postulated accidents, including the loss-of-coolant accident. Regulatory Guide 1.29 provides guidance for determining which SSCs should be designed to withstand the safe shutdown earthquake (SSE). Position C.2 recommends that SSCs whose continued function is not required but whose failure could reduce the functioning of the ECCS to an unacceptable safety level should be designed and constructed to withstand the SSE.

   Meeting the requirements of GDC 2, and positions of Regulatory Guide 1.29, provides assurance that plant safety is enhanced by ensuring the integrity of Seismic Category I portions of the ECCS and thus the capability to provide core cooling following a seismic event.

2. Compliance with GDC 4 requires that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accident conditions. These conditions include consideration of the dynamic effects of flow instabilities and the loadings caused by water hammer events.

   GDC 4 is applicable because the ECCS provides emergency core cooling in the event that normal cooling methods are not available or are insufficient.

   Meeting the requirements of GDC 4 provides assurance that dynamic effects of events such as flow instabilities and water hammer will not adversely affect the fundamental integrity and capability of the ECCS systems to provide core cooling in the event of accidents.

3. Compliance with GDC 5 prohibits the sharing of SSCs among nuclear power units unless it can be shown that such sharing will not significantly impair the ability of the SSCs to perform their safety functions, including, in the event of an accident in one unit, and orderly shutdown and cooldown of the remaining units.

   GDC 5 is applicable because the ECCS provides an important safety function in its ability to provide emergency core cooling and shutdown capability following postulated accidents. The ECCS system must be designed such that the ability to perform this and other designated safety-related functions are 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 unacceptable effects of equipment failures or other events occurring in one unit of a multi-unit site will not propagate to the unaffected unit(s).

4. Compliance with GDC 17 requires that an on-site and off-site electric power supply system be provided to permit functioning of SSCs important to safety.

   GDC 17 is applicable, as it relates to the ECCS systems, because it requires that each power supply system have sufficient capacity and capability to ensure that the core can be cooled in the event of an accident, and that the fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded during anticipated operational occurrences (AOOs). The ECCS is dependent upon the availability of electrical power supplied from the Class IE emergency electrical busses. The power supplies for the ECCS must maintain voltages at electrical equipment within the design limits. With voltages below design limits, electric equipment may not have sufficient capacity or capability to reliably perform their intended safety function during a design basis event.

Meeting the requirements of GDC 17 enhances plant safety by ensuring that the ECCS capacity and capabilities will be sufficient to ensure that the fuel design limits and reactor coolant pressure boundary integrity are maintained during AOOs and that the core is cooled during accidents.

5. Compliance with GDC 27 establishes requirements regarding the combined reactivity control system capability.

   GDC 27 is applicable because upon actuation the ECCS in PWRs provides rapid injection of borated water to ensure reactor shutdown and adequate core cooling with appropriate margins for stuck control rods. Injection of borated water provides negative reactivity to reduce reactor power to residual levels and ensures sufficient cooling flow to the core.

   Meeting the requirements of GDC 27 for the ECCS augments the protection for the primary fission product barrier by providing a means to ensure that the core, under postulated accident conditions, can be safely shutdown and will be maintained in a coolable geometry.

6. Compliance with GDC 35 requires that an emergency core cooling system be provided that is capable of transferring heat from the reactor core, following a loss of reactor coolant, at a rate sufficient to ensure that the core remains in a coolable geometry and that the clad metal-water reaction is limited to negligible amounts.

   GDC 35 is applicable because following a breach in the reactor coolant pressure boundary, reactor coolant is lost at a rate determined by several factors, including break size and RCS pressure. The emergency core cooling systems are relied upon to inject adequate cooling water into the RCS during a LOCA and to circulate the water through the core to provide for core cooling. The ECCS systems must inject cooling water at a rate sufficient to ensure that the calculated changes in core geometry will be such that the core remains amenable to cooling, and that the calculated cladding oxidation and hydrogen generation meet the specified performance criteria.

   Meeting the requirements of GDC 35 ensures that the ECCS, assuming a single failure, can provide core cooling under accident conditions sufficient to maintain the core in a coolable geometry and to minimize the production of hydrogen due to reaction of water with the fuel cladding.

7. Compliance with GDC 36 requires that the emergency core cooling systems be designed to allow for periodic inspections of important components to ensure the integrity and capability of the system.

   GDC 36 is applicable because the ECCS system arrangements must be designed such that adequate clearances are available to conduct periodic inspections of important components. Conduct of periodic inspections is necessary to show that important components of the ECCS systems are being maintained within their design basis specifications and that no significant deterioration is occurring in the systems. Meeting the requirements of GDC 36 enhances plant safety by ensuring that important ECCS components can be inspected and will be capable of operating as designed to cool the core under accident conditions.

8. Compliance with GDC 37 requires that the emergency core cooling systems be designed to allow for comprehensive periodic pressure and functional testing.

   GDC 37 is applicable because the ECCS is required to undergo periodic pressure testing to verify the structural and leak-tight integrity of important components. Periodic functional testing of the ECCS verifies that the systems will operate as designed including the full operational sequence necessary to initiate ECCS operation. Periodic functional test programs, such as the ECCS pump and valve testing, are premised upon the establishment of a reference set of parameters (based upon design specifications) and a consistent test method to allow for the detection of significant system degradation.

   Meeting the requirements of GDC 37 enhances plant safety by ensuring that important ECCS components can be tested and will remain capable of operating as designed to provide core cooling under postulated accident conditions.

9. Compliance with 10 CFR 50.46, requires that the ECCS be designed so that the calculated cooling performance is in accordance with an acceptable evaluation model or alternately a model in conformance with the features of Appendix K.

   10 CFR 50.46 is applicable because the primary function of the ECCS is to provide emergency core cooling and negative reactivity addition in the event of a LOCA resulting from a break in the primary reactor coolant system. The primary ECCS safety functions are comprehensively modeled and evaluated for breaks up to and including the double-ended severance of a reactor coolant pipe to show that the ECCS will limit the peak clad temperature to below 1204°C (2200°F) and ensure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved.

   Meeting the requirements of 10 CFR 50.46, enhances plant safety by ensuring that the ECCS is designed and evaluated in such a way that the calculated core cooling performance after a LOCA conforms to critical criteria necessary to show that the core geometry will remain amenable to cooling and that long-term decay heat removal will be provided.

III. REVIEW PROCEDURES

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

For each area of review specified in subsection I of this SRP section, the review procedure is identified below. These review procedures are based on the identified SRP acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the relevant NRC requirements identified in subsection II.

For combined license (COL) reviews, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report or design certification document (DCD). The COL review also includes the proposed technical specifications to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

Much of the review described below is generic in nature and is not performed for each plant. That is, the reviewer compares the ECCS design and parameters to those of previously reviewed plants and then devotes the major portion of the review effort to those areas where the application is not identical to previously reviewed plants. The following steps are taken by the reviewer to determine that the acceptance criteria of subsection II have been met. These steps should be adapted to COL or design certification reviews as appropriate.

1. The relationship of the system under review to other previously approved plants is established. Systems or design features claimed to be identical or equivalent to those of previously approved plants are confirmed to be identical or equivalent.

2. Piping diagrams are reviewed to evaluate the functional reliability of the system in the event of single failures. That is, by referring to piping and instrumentation diagrams, the existence of the redundancy required by the criteria is confirmed.

3. The significant design parameters (e.g., pump net positive suction head, pump head vs. flow, accumulator volume and pressure, water storage volume, system flow rate and pressure, etc.) are examined for each component to confirm that these parameters satisfy operating requirements and the recommendations of Regulatory Guide 1.1.

4. The piping and instrumentation diagrams are checked to see that essential ECCS components are designated seismic Category I and Safety Class II (the cooling water side of heat exchangers can be Safety Class III).

5. The ECCS design is reviewed to confirm that the system can function in post-accident environments, considering possible mechanical effects, missiles, and the pressure, temperature, moisture, radioactivity, and chemical conditions resulting from LOCA. Protection against valve motor flooding should be confirmed by the reviewer. Regarding the effects of pressure, temperature, etc., the reviewer should confirm that accident conditions are specified which provide the basis for proof tests for environmental qualification of ECCS components.

6. The criteria, supporting analyses, plant design provisions, and operator actions that will be taken are reviewed to ensure that there will not be unacceptably high concentrations of boric acid in the core region (resulting in precipitation of a solid phase) during the long-term cooling phase following a postulated LOCA.

7. The ECCS design is reviewed to confirm that there are provisions for maintenance of the long-term coolant recirculation and decay heat removal systems, e.g., pump or valve overhaul, in the post-LOCA environment (including consideration of radioactivity). The ECCS design is reviewed to confirm that switching suction to the containment sump does not occur prematurely (not enough inventory to provide pump suction) and adversely affect the accident (addresses GI-24).

8. The availability of an adequate source of water for the ECCS is confirmed, and the source volume, location, and susceptibility to failure (e.g., freezing) are evaluated. In PWRs, the piping from the water source to the ECCS safety injection pumps is evaluated for conformance with Branch Technical Position RSB 6-1.

9. The ECCS flow paths are reviewed to determine the extent to which flow from the ECCS pumps is diverted as a backup feature to other safeguards equipment (e.g., RHR, containment spray). The reviewer should confirm that the remaining portion of the flow provides abundant core cooling, despite the most severe single failure that affects ECCS flow.

10. For a BWR or the advanced passive safety plants (both BWR and PWR), the reactor coolant ADS systems are reviewed utilizing the following additional procedures to verify compliance with the Acceptance Criteria:

    A. The ADS systems, including electrical power supplies, are reviewed to verify they have sufficient independence, redundancy, and capability to allow the ADS to function properly assuming a single failure.

    B. The ADS design is reviewed to verify that actuation of the system can be completed automatically and that manual actuation is not required to assure adequate core cooling.

C. Design features and system analysis to verify performance of the ADS under all accident conditions are reviewed. The reviewer should verify the ADS can satisfy performance requirements without taking credit for non-safety-related equipment or instrumentation, and accounting for normal air (or nitrogen) leakage through the valves. For those BWR applicants which use a pneumatic supply for the ADS function, the capability and design of the pneumatic supply system is reviewed under SRP Section 9.3.1.

D. The applicant's evaluation of the ADS with respect to reactor vessel integrity limits is reviewed. If integrity limits could be exceeded during rapid cooldown, the applicant should evaluate alternate depressurization methods, other than full actuation of the ADS system, such as early depressurization with one or two relief valves.

E. The ADS design is reviewed to verify that the depressurization/safety valves be provided with a positive indication in the control room of flow in the discharge pipe.

11. The design of ECCS injection lines is reviewed to confirm that the isolation provisions at the interface with the reactor coolant system are adequate. The number and type of valves used to form the interface between low pressure portions of the ECCS and the reactor coolant system must provide adequate assurance that the ECCS will not be subjected to a pressure greater than its design pressure. This may be accomplished by any of the following provisions:

    A. One or more check valves in series with a normally closed motor-operated valve. The motor-operated valve is to be opened upon receipt of a safety injection signal once the reactor coolant pressure has decreased below the ECCS design pressure.

    B. Three check valves in series.

    C. Two check valves in series, provided that there are design provisions to permit periodic testing of the check valves for leaktightness and the testing is performed at least annually.

12. The reviewer should identify those portions of nonsafety-related systems which could have an adverse effect on ECCS and should ensure that design provisions are in place to prevent these situations.

13. Motor-operated isolation valves in ECCS lines connecting the accumulators to the reactor coolant system in a pressurized water reactor (PWR) are reviewed to ensure that adequate provisions are made against inadvertent isolation. Safety-related MOVs are subject to qualification testing to demonstrate the capability of the valve to open, close, and seat against the maximum differential pressure and flow (addresses GI-87).

14. The capacity and settings of relief valves provided for the ECCS to satisfy system overpressure protection requirements are reviewed. In particular, for PWRs, the reviewer confirms that the accumulator relief valves have adequate capacity so that leakage from the reactor coolant system will not jeopardize the integrity of the accumulators.

15. The ECCS is reviewed to evaluate the adequacy of design features that have been provided to prevent damaging water (steam) hammer due to such mechanisms as voided discharge lines, water entrainment in steam lines and steam bubble collapse. For systems with a water supply above the discharge lines, voided lines are prevented by proper vent location and filling and venting procedures. However, for the core spray and low pressure coolant injection systems of BWRs, the low elevation of the

suppression pool will result in line voidage because of back leakage through pump discharge check valves and leaking valves in the full flow test line. Proper vent location and filling and venting procedure are still needed. In addition, a special keep-full system with appropriate alarms is needed to supply water to the discharge lines for any system which has a water source below the level of the highest pump discharge lines and at sufficiently high pressure to prevent voiding.

For the High Pressure Coolant Injection (HPCI) system of BWRs which use a steam-driven turbine, typical design features for the steam supply line include: (a) drain pots with testable drain pot level switches, (b) sloped lines, and (c) limitations on opening and closing sequences and seal-ins for manual operation of the isolation valves to prevent introducing water slugs into the line. The turbine exhaust line features include sloped lines and vacuum breakers. HPCI steam isolation valves should be subject to qualification testing to demonstrate the capability of the valve to open, close, and seat against maximum differential pressure and flow (addresses GI 87).

Guidance for water hammer prevention and mitigation is found in NUREG-0927.

16. The reviewer confirms that no component or feature of the ECCS in one reactor facility on a multiple plant site is shared with the ECCS in another facility, or that shared features clearly meet the requirements of GDC 5.

17. The reviewer confirms that within an individual reactor facility, any components shared between the ECCS and other systems (e.g., coolant makeup systems, residual heat removal systems, containment cooling systems) satisfy engineered safeguard feature design requirements and that the ECCS function of the shared component is not diminished by the sharing.

18. The reviewer confirms that ECCS components located exterior to the reactor containment are housed in a structure which, in the event of leakage from the ECCS, permits venting of releases through iodine filters designed in accordance with Regulatory Guide 1.52.

19. The complete sequence of ECCS operation from accident occurrence through long-term core cooling is examined to see that a minimum of manual action is required and, where manual action is used, a sufficient time (greater than 20 minutes) is available for the operator to respond.

20. The reviewer confirms that long-term cooling capacity is adequate in the event of failure of any single active or passive component of the ECCS. If an intermediate heat transport system, such as the component cooling water system, is used to provide long-term cooling capability, the system must be designed and constructed to an appropriate group classification, must be seismic Category I, and must be capable of sustaining a single active or passive failure without loss of function.

21. With respect to ECCS power requirements, instrumentation and controls, and valve controls, the reviewer:

    A. Confirms that the power requirements of the ECCS, including the timing of electrical loads, are compatible with the design of onsite emergency power systems, both a-c and d-c.

    B. Confirms that there are sufficient instrumentation and controls available to the reactor operator to provide adequate information in the control room to assist in assessing post-LOCA conditions, including the more significant parameters such as coolant flow, coolant temperature, and containment pressure. If ECCS flow is diverted as a backup to other safeguards systems, the reviewer confirms that instrumentation and controls are available to provide sufficient information in the control room to determine that adequate core cooling is being provided.

C. Confirms that automatic actuation and remote-manual valve controls are capable of performing the functions required, that suitable interlocks are provided, which do not impair separation of power trains or inhibit the required valve motions, and that instrumentation and controls have sufficient redundancy to satisfy the single failure criterion.

22. Analyses are provided by the applicant in Chapter 15 of the SAR or DCD to assess the capability of the ECCS to meet functional requirements. These analyses are reviewed, as described in SRP Section 15.6.5, to determine conformance of the ECCS to the acceptance criteria. However, the following portions of the review of ECCS response in loss-of-coolant accidents are performed by the reviewer under this SRP section:

    A. The lower limit of break size for which ECCS operation is required is established; i.e., the maximum break size for which normal reactor coolant makeup systems can maintain reactor pressure and coolant level is determined. The capability of the ECCS to actuate and perform at this lower limit of break size is confirmed.

    B. The reviewer confirms that the analyses take into account a variety of potential locations for postulated pipe breaks, including ECCS injection lines.

    C. The reviewer confirms that the analyses take into account a variety of single active failures. The reviewer should keep in mind that different single failures may be limiting, depending on the particular break location and break size postulated.

    D. The ECCS component response times (e.g., for valves, pumps, power supply) are reviewed to confirm that they are within the delay times used in the accident analyses.

    E. The ECCS design adequacy for all modes of reactor operation (e.g., full power, low power, hot standby, cold shutdown, partial loop isolation) is confirmed.

23. The proposed plant technical specifications are reviewed to:

    A. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when ECCS equipment is inoperable due to repairs and maintenance. The means of indicating that safety systems have been bypassed or are inoperable should be in accordance with Regulatory Guide 1.47.

    B. Confirm that the limiting conditions for operation ensure that the specified operating parameters (minimum poison concentrations, minimum coolant reserve in storage, etc.) are within the bounds of the analyzed conditions.

    C. Verify that the frequency and scope of periodic surveillance testing is adequate.

24. The reviewer verifies that the emergency core cooling systems are designed to allow for comprehensive periodic inservice inspection, pressure and functional testing as indicated below:

    A. The reviewer confirms that the design provides the capability for periodically demonstrating that the system will operate properly when an accident signal is received. That is, it should be demonstrated by an applicant that pumps and valves operate on normal and emergency power and that water pressure and flow are as designed when the plant is operating (periodic system surveillance). When the plant is shut down for refueling, the system should be tested for delivery of coolant to the vessel. The ECCS design should have provisions to permit appropriate periodic inspection of important components and pressure testing.

B. For new applications, the reviewer verifies that the ECCS piping design incorporates provisions to allow for full flow testing (maximum design flow) of pumps and check valves. For those designs where it is not practical to conduct the inservice pump testing at design flow and pressure, full flow testing at maximum design flow with analysis to extrapolate to design pressure is sufficient.

C. For new applications, the reviewer verifies that the ECCS design incorporates provisions to allow for testing of ECCS system motor-operated valves under design-basis differential pressure.

D. For new applications, when it is not practicable to achieve design basis differential pressure during ECCS valve testing, a qualification test (under design-basis differential pressure) prior to installation and inservice valve tests conducted under the maximum practicable differential pressure is sufficient.

25. The reviewer determines any special test requirements and confirms that the proposed preoperational test program for the ECCS is in conformance with the intent of Regulatory Guide 1.68.

26. The reviewer evaluates the applicant responses to the following Task Action Plan items:

A. III.D.1.1 of NUREG-0737; the reviewer verifies that those portions of the ECCS located outside of containment that contain or may contain radioactive material following an accident are included in a leakage control program. The leakage control program should include periodic leak testing and measures to minimize leakage from the ECCS.

B. II.K.3(15) of NUREG-0660; the reviewer should verify that BWR applicants' designs for pipe-break detection circuitry will not cause inadvertent system isolation during pressure spikes resulting from HPCI and RCIC system initiation. For those plants utilizing a time delay relay, the minimum expected response time will be plant specific, the maximum response time shall be no greater than seven seconds unless the applicant provides proper justification for using a longer response time.

C. II.K.3.21 of NUREG-0737; the reviewer should verify that BWR applicants have studied the design of BWR 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.

D. II.K.3.16 of NUREG-0737, the reviewer should verify that applicants have evaluated methods to reduce challenges and failures of reactor coolant system relief valves.

27. The reviewer verifies that the applicant has administrative procedures in place that establish limitations on the ECCS cumulative outage times (GI B-61). The reviewer verifies that the applicant will prepare and submit annual reports on ECCS unavailability (II.K.3.17) that also include information on: outage dates, lengths, and causes; ECCS components involved; and any corrective action taken.

28. The reviewer verifies that the applicant has considered the following guidance regarding the design of the ECCS miniflow systems necessary to insure safety related ECCS pump protection:

A. Assure that the minimum cooling flow provided for the ECCS pumps is adequate under all conditions, including verification that the system configuration precludes pump-to-pump interaction during miniflow operation that could result in dead-heading one or more of the pumps. The miniflow must be sufficient to prevent damage to the pump(s) under all conditions.

B. The miniflow system shall be designed such that the miniflow function can be performed assuming a single failure. A single failure should not result in conditions causing no flow through the ECCS pumps.

C. In cases where only the miniflow return line is available for pump testing, flow instrumentation must be installed on the miniflow return line. This instrumentation is necessary to provide flow rate measurements during pump testing so that this data can be evaluated with the measured pump differential pressure to monitor for pump hydraulic degradation.

29. The reviewer evaluates the ECCS capability to provide reactor coolant system inventory additions during reduced inventory operations as follows:

A. PWR designs and related operating procedures should have a means of providing at least two available or operable means of adding inventory to the RCS that are in addition to pumps that are a part of the normal decay heat removal systems. These means should include at least one high pressure injection pump from the ECCS.

B. The water addition rate provided by each of the means should be at least sufficient to keep the core covered.

C. Procedures should be provided for use of these systems during loss of decay heat removal events. The path of water addition must be specified to assure the flow does not bypass the reactor vessel before exiting any opening in the RCS.

30. The reviewer verifies that the applicant has reviewed their ECCS design configurations to identify any unisolable piping connected to the RCS that could be subjected to temperature distributions which would result in unacceptable thermal stresses. This review should consider the potential for thermal stratification, thermal cycling and thermal fatigue given the ECCS piping configurations. The reviewer verifies that appropriate action has been taken, where such piping is identified, to ensure that the piping will not be subjected to unacceptable thermal stresses. The review focuses on ECCS configurations; reviewing the stress analysis and ensuring the stresses are in compliance with the ASME code is performed under SRP Section 3.9.3.

31. For reviews of DC and COL applications under 10 CFR Part 52, the reviewer should follow the above procedures to verify that the design set forth in the safety analysis report, and if applicable, site interface requirements meet the acceptance criteria. For DC applications, the reviewer should identify necessary COL action items. With respect to COL applications, the scope of the review is dependent on whether the COL applicant references a DC, an ESP or other NRC-approved material, applications, and/or reports.

After this review, SRP Section 14.3 should be followed for the review of Tier I information for the design, including the postulated site parameters, interface criteria, and ITAAC.

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. (For completeness, this evaluation finding includes the review effort described in SRP Section 15.6.5.)

The emergency core cooling system (ECCS) includes the piping, valves, pumps, heat exchangers, instrumentation, and controls used to transfer heat from the core following a loss-of-coolant accident. The scope of review of the ECCS for the __________ plant included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analyses, and design specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the ECCS and the manner in which the design conforms to these criteria and bases.

The staff concludes that the design of the Emergency Core Cooling System is acceptable and meets the requirements of General Design Criteria 2, 4, 5, 17, 27, 35, 36, and 37; 10 CFR 50.34(f)(1)(vi), (vii), (viii), (ix), (x), and (xi); 10 CFR 50.34(f)(2)(xi), (xviii), and (xxvi) and 10 CFR 50.46. This conclusion is based on the following:

1. The applicant has met the requirements of GDC 2 with regard to the seismic design of nonsafety systems or portions thereof which could have an adverse effect on ECCS by meeting position C.2 of Regulatory Guide 1.29.

2. The applicant has met the requirements of GDC 4 as related to dynamic effects associated with flow instabilities and loads (e.g., water hammer).

3. The applicant has met the requirements of GDC 5 with respect to sharing of structures, systems, and components (SSCs) by demonstrating that such sharing does not significantly impair the ability of the ECCS to perform its safety function including, in the event of an accident to one unit, an orderly shutdown and cooldown of the remaining units.

4. The applicant has met the requirements of GDC 17 with regard to providing sufficient capacity and capability to assure that (a) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (b) the core is cooled and vital functions are maintained in the event of postulated accidents.

5. The applicant has met the requirements of GDC 27 with regard to providing combined reactivity control system capability to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained and the applicant's design meets the guidelines of Regulatory Guide 1.47.

6. The applicant has met the requirements of GDC 35 to provide abundant cooling for ECCS by providing redundant safety-grade systems that meet the recommendations of Regulatory Guide 1.1.

7. The applicant has met the requirements of GDC 36 with respect to the design of ECCS to permit appropriate periodic inspection of important components of the system.

8. The applicant has met the requirements of GDC 37 with respect to designing the ECCS to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation.

9. The applicant has provided an analysis of the proposed ECCS relative to the acceptance criteria of 10 CFR 50.46, and with regard to the evaluation models the guidance of Regulatory Guide 1.157 or alternatively Appendix K of 10 CFR 50. The

applicant has demonstrated that their ECCS designs satisfy the criteria for peak cladding temperature, maximum calculated cladding oxidation, maximum hydrogen generation, coolable core geometry, and long-term cooling in accordance with an acceptable evaluation model.

10. The applicant has met II.K.3.18 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(vii) for applicants subject to 10 CFR 50.34(f), with respect to eliminating the need for manual actuation of the BWR ADS to assure adequate core cooling.

11. The applicant has met II.K.3.21 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(viii) for applicants subject to 10 CFR 50.34(f), with respect to reviewing the design of BWR 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.

12. The applicant has met II.K.3.28 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(x) for applicants subject to 10 CFR 50.34(f), with respect to the BWR ADS-associated equipment and instrumentation being capable of performing their intended functions during and following an accident, while taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves.

13. The applicant has met II.K.3.45 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(xi) for applicants subject to 10 CFR 50.34(f), in regard to an evaluation of depressurization methods, other than full actuation of the ADS, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown for BWRs.

14. The applicant has met III.D.1.1 of NUREG-0737, equivalent to 10 CFR 50.34(f)(2)(xxvi) for applicants subject to 10 CFR 50.34(f), with respect leakage detection and control in the design of ECCS outside containment that contain (or may contain) radioactive material following an accident.

15. The applicant has met the requirements of Task Action Plan item II.K.3.15 of NUREG-0660 which involves isolation of HPCI and RCIC for BWR plants.

16. The applicant has met the requirements of Task Action Plan item II.K.3.16 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(vi) for applicants subject to 10 CFR 50.34(f), with regard to providing an evaluation of methods to reduce challenges and failures of reactor coolant system relief valves.

17. The applicant has met the requirements of Task Action Plan item II.K.3.24 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(ix) for applicants subject to 10 CFR 50.34(f), with respect to the adequacy of space cooling for long-term operation of HPCI and RCIC systems for BWRs to maintain the operating environment within allowable limits.

18. The applicant has met the requirements of Task Action Plan item II.D.3 of NUREG-0737, equivalent to 10 CFR 50.34(f)(2)(xi) for applicants subject to 10 CFR 50.34(f), with respect to the requirements that reactor coolant system relief and safety valves be provided with a positive indication in the control room of flow in the discharge pipe.

19. The applicant has met the requirements of Task Action Plan item II.F.2 of NUREG-0737, equivalent to 10 CFR 50.34(f)(2)(xviii) for applicants subject to 10 CFR 50.34(f), with respect to the requirement that instrumentation or controls provide an unambiguous, easy-to-interpret indication of inadequate core cooling.

For DC and COL reviews, the findings will also summarize (to the extent that the review is not discussed in other SER sections) the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable, and interface requirements and combined license action items relevant to this SRP section.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

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 docketed six months or more after the date of issuance of this SRP section, unless superceded by a later revision.

VI. REFERENCES

1. 10 CFR 50.34(f), "Additional TMI-Related Requirements."

2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

3. 10 CFR 50.63, "Loss of all Alternating Current Power."

4. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

5. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Dynamic Effects Design Basis."

6. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

7. 10 CFR Part 50, Appendix A, General Design Criterion 17, "Electric Power Systems."

8. 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control System Capability."

9. 10 CFR Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling."

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

11. 10 CFR Part 50, Appendix A, General Design Criterion 37, "Testing of Emergency Core Cooling System."

12. 10 CFR Part 50, Appendix K , "ECCS Evaluation Models."

13. Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps."

14. Regulatory Guide 1.29, "Seismic Design Classification."

15. Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

16. Regulatory Guide 1.52, “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.

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

18. Regulatory Guide 1.155, “Station Blackout.”

19. Regulatory Guide 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance.”

20. Branch Technical Position RSB 6-1, “Piping From the RWST (or BWST) and Containment Sump(s) to the Safety Injection Pumps,” attached to this SRP Section.

21. SECY 90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” dated January 12, 1990.

22. Staff Requirements Memorandum, “SECY 90-016 - Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements,” dated June 26, 1990.

23. SECY 93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” dated April 2, 1993.

24. Staff Requirements Memorandum, “SECY 93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” dated July 21, 1993.

25. NRC Letter to all Boiling Water Reactor Licensees, “NUREG-0737 Technical Specifications (Generic Letter No. 83-02),” January 10, 1983.

26. NRC Letter to all Holders of Operating Licenses or Construction Permits for Pressurized Water Reactors (PWRs), “Loss of Decay Heat Removal (Generic Letter No. 88-17),” October 17, 1988.

27. NRC Letter to All Holders of Light Water Reactor Operating Licenses and Construction Permits, “Guidance on Developing Acceptable Inservice Testing Programs (Generic Letter 89-04),” April 3, 1989.

28. NRC Bulletin 79-24, “Frozen Lines,” September 27, 1979.

29. NRC Bulletin 80-18, “Maintenance of Adequate Minimum Flow Thru Centrifugal Charging Pumps Following Secondary Side High Energy Line Rupture,” July 24, 1980.

30. NRC Bulletin 86-03, “Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line,” October 8, 1986.

31. NRC Bulletin 88-04, “Potential Safety-Related Pump Loss,” May 5, 1988.

32. NRC Bulletin 88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems,” June 22, 1988 and its Supplements 1 through 3.

33. NUREG-0718, “Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses.”

34. NUREG-0737, “Clarification of TMI Action Plan Requirements.”

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

36. American National Standard, “Single Failure Criteria for PWR Fluid Systems,” ANSI N658 (ANS 51.7).

37. Regulatory Guide 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident.”

38. NUREG-0933, “A Prioritization of Generic Safety Issues,” July 1991

39. NUREG-1462, “Final Safety Evaluation Report Related to Certification of the System 80+ Design.”

40. NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design.”

41. Regulatory Guide 1.79, “Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors.”

42. SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems in Passive Plant Designs.”

43. SECY-77-439, “Single Failure Criterion.”

44. 10 CFR Part 52, 52.47, “Contents of Applications.”

45. 10 CFR Part 52, 52.97, “Issuance of Combined Licenses.”

46. RC Inspection Manual Chapter IMC-2504, "Construction Inspection Program - Non-ITAAC Inspections," issued April 25, 2006.

BRANCH TECHNICAL POSITION RSB 6-1
CURRENTLY THE RESPONSIBILITY OF REACTOR SYSTEMS
PIPING FROM THE RWST (OR BWST) AND CONTAINMENT SUMP(S)
TO THE SAFETY INJECTION PUMPS

1. Background

Current PWRs utilize the refueling water storage tank (RWST) or the borated water storage tank (BWST) as the sole source of water for the safety injection pumps during the first 20 to 40 minutes of any accident that trips a safety injection signal. Since acceptable results of safety analyses of the accidents are based on the operation of a minimum number of these pumps interruption of this water supply for even a short period of time could result in unacceptably high fuel and cladding temperatures if the safety injection pumps fail because of cavitation or overheating.

General Design Criteria 35 requires that the emergency core cooling system have suitable redundancy in components and features and suitable interconnections to assure the system safety function can be accomplished assuming a single failure. The principal problem appears to be a definition of single failure. ANSI N658, "Single Failure Criteria for PWR Fluid Systems," defines an active failure as:

A. An active failure is a malfunction, excluding passive failures, of a component which relies on mechanical movement to complete its intended function upon demand."

B. "Spurious action of a powered component originating within its actuation or control system shall be regarded as an active failure unless specific design features or operating restrictions preclude such spurious action."

This branch position on the availability of the RWST is based on the above criteria and the recognition that water supplied from the RWST system to the ECCS system is absolutely essential in the event of a LOCA.

2. Branch Position

A. The single active failure criterion defined in (a) and (b) above will be applied in evaluating the design of the piping systems that connect the safety injection pumps to the RWST (BWST) and the containment sumps.

B. The piping systems, including valves, shall be designed to satisfy the requirements listed below without the need to disconnect the power to any valve.

C. The valves and piping between the RWST (or BWST) and the safety injection pumps must be arranged so that no single failure will prevent the minimum flow to the core required to satisfy 10 CFR 50.46.

D. The valves and piping between the RWST (or BWST) and safety injection pumps must be arranged so that no single active failure will result in damage to pumps such that the minimum flow requirements for long-term core and containment cooling after a LOCA are not satisfied.

E. The valves and piping that connect the RWST (or BWST) and the containment sump(s) to the safety injection pumps must be arranged so as not to preclude automatic switchover from the injection mode of ECCS operation to recirculation cooling from the sump. These piping systems must be arranged so that the differential pressure between the sump and the RWST (or BWST), even if there is a single active failure, will not result in a loss of core cooling or a path that permits release of radioactive material from the containment to the environment.

3. <u>Implementation</u>

A. Applicants for a construction permit for which an SER was published prior to April 16, 1975 will not be required to comply with the provisions of this item.

B. For plants with an operating license issued prior to July 1981 and operating license applications docketed prior to July 1981 the position will not be completely applied. Specifically, locking out power to valves will be permitted. For most plants it is expected that this will be sufficient to meet the single failure criteria. However, in other plants changes to the piping and valving arrangements may be required to satisfy the single failure criteria.

C. Applications docketed on or after July 1981 will be reviewed according to the provisions of this item.

SRP Section 6.3
Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in Draft Revision 2, dated April 2006, of this SRP section. See ADAMS accession number ML0616403370.

In addition, this SRP section was administratively updated in accordance with NRR Office Instruction LIC-200, Revision 1, "Standard Review Plan (SRP) Process." The revision also adds standard paragraphs to extend application of this updated SRP section to prospective applicant submissions pursuant to 10 CFR Part 52.

The technical changes are incorporated in Revision 2, dated [Month] 2007:

Review Responsibilities - Reflects changes in review branches resulting from reorganization and branch consolidation. Change is reflected throughout the SRP.

I. AREAS OF REVIEW

1. Reformatted the section with new numbering system. Incorporated reference to 10CFR52 from draft revision 1 - April 1996. Incorporated generic paragraphs relating to certified designs, ESPs, and COLs.

2. Added new design bases for passive cooled reactors (items 3 and 4).

3. Added new review guidance under “Review Interfaces” for TMI action Plan (item 5, 8) and Core melt stabilization system (item 3).

II. ACCEPTANCE CRITERIA

1. Reformatted the section with new numbering system. Incorporated reference to 10CFR52 from draft revision 1 - April 1996. Incorporated generic paragraphs relating to certified designs, ESPs, and COLs.

2. Added new acceptance criteria to include TMI action items (items 14 thru 16).

3. Under Specific Crieria added item to address Generic Issue C-4 (item 1) and B-61(item 10).

III. REVIEW PROCEDURES

1. Reformatted the section with new numbering system. Incorporated reference to 10CFR52 from draft revision 1 - April 1996. Incorporated generic paragraphs relating to certified designs, ESPs, and COLs.

2. Procedures revised to address GI-24, GI 87 and GI B-61

IV. EVALUATION FINDINGS

1. Added new items corresponding changes to acceptance criteria to include TMI action items (items 16 thru 19).

V. IMPLEMENTATION

None

VI. REFERENCES

1. Added reference to NRC Inspection Manual Chapter IMC-2504, “Construction Inspection Program - Non-ITAAC Inspections,” issued April 25, 2006.