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REGULATORY GUIDE

REGULATORY GUIDE 1.68

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INITIAL TEST PROGRAMS FOR WATER-COOLED NUCLEAR POWER PLANTS

A. INTRODUCTION

Purpose

This revision of Regulatory Guide (RG) 1.68 describes the general scope and depth that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for demonstrating compliance with the NRC regulations identified below as they pertain to Initial Test Programs (ITPs) for light water cooled nuclear power plants. Appendix A addresses the specific tests recommended or required for the ITP. Appendix B to this guide provides information about ITP-related inspections that the NRC staff will perform, including the appropriate regional offices. Finally, Appendix C contains guidance on the preparation and content of procedures for preoperational, fuel loading, initial criticality, low power, and power ascension tests.

Applicable Rules & Regulations

This revision to RG 1.68 may not be suitable for ITPs required as part of new regulations issued after the date shown on the front page of this guide. For example, RG 1.68 does not address new equipment added to the current fleet of plants to mitigate long term station blackout and beyond design basis events. New NRC regulations and RGs may be used to implement requirements for mitigating equipment (i.e., containment venting systems, hardened vents, hardened filter vents, etc.) in light water reactors (LWRs) with mitigating strategies to address long term station blackout and beyond design basis events. Licensees or applicants that may be impacted by newer regulations should contact the NRC staff for guidance prior to developing ITPs for any equipment mandated by the new regulations.

The application requirements for an operating license (OL), standard design approval (SDA), design certification (DC), combined license (COL), and manufacturing license (ML) respectively, are presented in Title 10 of the *Code of Federal Regulations*, (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (10 CFR Part 50) (Ref. 1), 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 2), 10 CFR 50.34, "Contents of Applications; Technical Information," 10 CFR 52.47, "Contents of Applications, Technical Information," 10 CFR 52.79, "Contents of Application; Technical Information in Final Safety Analysis Report,"

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Electronic copies of this regulatory guide, previous version of this guide, and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/. The regulatory guide is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML13051A027. The regulatory analysis for DG-1259 may be found in ADAMS at Accession No. ML13051A035 and the staff responses to the public comments on DG-1259 may be found in ADAMS at Accession No. ML13051A034.

10 CFR 52.137, “Contents of Applications; Technical Information,” and 10 CFR 52.157, “Contents of Application; Technical Information in Final Safety Analysis Report,” (FSAR). These provisions require, in part, that an OL, SDA, DC, COL and ML applicant provide the principal design criteria for the proposed facility. The introduction to Appendix A, “General Design Criteria (GDC) for Nuclear Power Plants,” to 10 CFR Part 50 states that these principal design criteria are to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety (i.e., SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public).

Criterion XI, “Test Control,” of Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50 states, in part, that a test program shall be established to assure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures, which incorporate the requirements and acceptance limits contained in applicable design documents. In accordance with 10 CFR Part 50, Appendix A, all SSCs important to safety that are required to perform these functions need to be tested to ensure that they will perform properly. These functions, as noted throughout the GDCs, are those necessary to ensure that specified design conditions of the facility are not exceeded during any condition of normal operation, including anticipated operational occurrences, or as a result of postulated accident conditions.

The regulations in §50.34(b)(6)(iii) and §52.79(a)(28) require, in part, that the OL or COL applicant include “Plans for preoperational testing and initial operations” in the FSAR. Subsection 52.157(f)(27) of 10 CFR requires an ML applicant to submit “Necessary parameters to be used in developing plans for preoperational testing and initial operation.” Chapter 14, “Initial Test Program,” of RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition” (Ref. 3), and regulatory position C.I.14, “Verification Programs,” of RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition)” (Ref. 4), provide guidance on information related to the ITP to be included in the FSAR to enable the NRC staff to perform its safety evaluations for OLs and COLs.

For DC and SDA applicants, 10 CFR 52.47 and 10 CFR 52.137 do not require plans for an ITP. However, DC and SDA applicants have previously submitted plans for an ITP in their applications to assist the COL applicant that references those SDAs or DCs to meet the requirement in §52.79(a)(28) to include plans for an ITP in its application. For COL applicants referencing a DC or SDA, 10 CFR 52.73 “Relationship to Other Subparts,” Section (a), states that a COL applicant may reference a DC under Subpart B, “Standard Design Certifications” or SDA under Subpart E, “Standard Design Approval.” In accordance with §52.79(a)(28), the COL applicant is required to provide plans for preoperational testing and initial operations in the application.

The regulations in §52.47(b)(1) and §52.80(a) also require, in part, that applications for a DC or COL include the inspections, tests, analyses and acceptance criteria (ITAAC) necessary to demonstrate that the facility has been constructed and will be operated in conformity with the COL and NRC regulations. ITAAC often include testing requirements that are also considered preoperational tests and are completed as part of the ITP under §52.79(a)(28). For additional details on guidance for ITAAC, see RG 1.215, “Guidance for ITAAC Closure under 10 CFR Part 52," (Ref. 5).

As part of the regulations in §50.34(b)(6)(iii) and §52.79(a)(28) for the ITP, the OL or COL applicant should describe in the FSAR the major phases of the ITP and the specific objectives to be achieved for each major phase. The descriptions and objectives of these test phases should be demonstrated to be consistent with the general guidelines and applicable regulatory positions contained in this regulatory guide, or justifications should be provided for any exceptions.

Some safety-significant design requirements cannot be verified by ITAAC because the testing can only be done after fuel load. For example, testing of the main steam isolation valves at high flow conditions, testing involving 100 percent load rejection from the turbine, or verification of fuel and control rod performance cannot be verified by ITAAC. These requirements and supporting analyses for these tests should be identified in the applicable sections of the design certification documents or Section 14.2 of the COL FSAR.

While regulations require all SSCs important to safety be tested, all of them need not be tested to the same stringent requirements. Specifically, GDC 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50 requires, in part, that SSCs important to safety shall be tested to quality standards commensurate with the importance of the safety functions to be performed. Criterion XI of Appendix B to 10 CFR Part 50 also includes a graded approach for important to safety SSCs in the quality assurance (QA) program. Accordingly, the administrative requirements that govern the conduct of the test program (e.g., test program objectives, organizational elements, personnel qualifications, evaluation and approval of test results, and test records retention) contain provisions for the application of such administrative controls in a manner commensurate with the safety significance of the SSCs within its scope. This provides a systematic approach to the "defense-in-depth" concept. This concept dictates that the plant must be designed, constructed, and tested to (1) provide for safe normal operation, (2) ensure that, in the event of errors, malfunctions, and off-normal conditions, the reactor protection systems and other design features will mitigate the event or limit its consequences to defined and acceptable levels, and (3) ensure that adequate safety margin exists for events of extremely low probability or arbitrarily postulated hypothetical events without substantial reduction in the safety margin for the protection of public health and safety.

Related Guidance

- RG 1.9, "Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants" describes methods that can be used to test and demonstrate emergency load capacities.

- RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," contains guidance on vibration monitoring recommendations during initial reactor startup.

- RG 1.29, "Seismic Design Classification," identifies that SSCs designated as Seismic Category I are considered to be more important to safety than other SSCs.

- RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants," identifies processes for cleaning, flushing, and layup of components.

- RG 1.41, "Preoperational Testing of Redundant Onsite Electric Power Systems To Verify Proper Load Group Assignments," describes testing to verify redundancy and electrical independence.

- RG 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," contains test leakage detection system sensitivity and capabilities methods.

- RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety Feature Atmosphere Cleanup Systems in Light Water Cooled Nuclear Power Plants," contains guidance on testing of ventilation systems.

- RG 1.68.1, “Initial Test Program of Condensate and Feedwater Systems for Light Water Reactors,” contains guidance for testing the condensate and feedwater systems.

- RG 1.68.2, “Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water Cooled Nuclear Power Plants,” contains guidance on testing the remote shutdown capability.

- RG 1.68.3, “Preoperational Testing of Instrument and Control Air Systems,” provides detailed guidance on testing of instrument and control air systems. Compressed gas systems may also be relied upon for breathing air; therefore, this system should be tested for air load transients that can affect both plant safety and personnel safety.

- RG 1.69, “Concrete Radiation Shields And Generic Shield Testing For Nuclear Power Plants,” describes neutron and gamma radiation surveys to be conducted as part of the ITP.

- RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition,” provides guidance on information on the ITP to be included in the FSAR to enable the NRC staff to perform its safety evaluations for OLs and COLs.

- RG 1.79, “Preoperational Testing of Emergency Core Cooling Systems for Pressurized-Water Reactors,” identifies specific Emergency Core Cooling System (ECCS) preoperational and startup tests in PWRs.

- RG 1.79.1, “Initial Test Program of Emergency Core Cooling Systems for New Boiling Water Reactors,” identifies specific Emergency Core Cooling System (ECCS) preoperational and startup tests in BWRs.

- RG 1.118, “Periodic Testing of Electric Power and Protection Systems,” provides a test criterion that is also acceptable for preoperational testing of protection channels, including sensors.

- RG 1.140, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” identifies specific testing programs for heating, cooling, and ventilation systems.

- RG 1.155, “Station Blackout,” provides guidance for testing of emergency ac power supply reliability.

- RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” contains information on operations, maintenance, testing, and monitoring of SSCs after fuel load.

- RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” provides guidance on information on the ITP to be included in the FSAR to enable the NRC staff to perform its safety evaluations for OLs and COLs.

- RG 1.215, “Guidance for ITAAC Closure under 10 CFR Part 52," contains additional information on ITAAC necessary to demonstrate that the facility has been constructed and will be operated in conformity with the COL and NRC regulations.

- RG 8.38, “Control of Access to High and Very High Radiation Areas in Nuclear Power Plants,” discusses neutron and gamma radiation surveys and shielding requirements to be verified during the ITP.

- Inspection Manual Chapter (IMC) 2503 “Construction Inspection Program: Inspections of Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) Related Work,” specifies the policy used for the NRC inspections of the ITAAC of a combined license or Limited Work Authorization (LWA) and provides guidance for inspections intended to support 10 CFR 52.103(g).

- IMC 2504, “Construction Inspection Program – Inspection of Construction and Operational Programs,” specifies the inspection policies for reviewing the programs not directly related to ITAAC that support construction of a plant licensed in accordance with 10 CFR Part 52. Also specifies the inspection policies to assess whether a licensee conforms to and correctly implements the preoperational testing portion of the ITP contained in the FSAR.

- Inspection Manual Chapter (IMC) 2513, Appendix A, “Light Water Reactor - Preoperational Testing Phase,” describes the inspection program to verify systems and components important to safety are fully tested to demonstrate that they satisfy their design requirements.

- IMC 2513, Appendix B, “Light Water Reactor - Operational Preparedness Phase,” describes the inspection program to verify that management controls and procedures, including quality assurance programs, necessary for operation of the facility have been documented and implemented.

- IMC 2514, “Light Water Reactor Inspection Program – Startup Testing Phase,” describes the inspection activities used to verify that the licensee is meeting the requirements and conditions of the facility license for precritical tests, initial fuel loading, initial criticality, low-power testing, and power ascension tests.

- NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants,” contains guidance for tests on single failure proof overhead crane handling systems.

- NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A 36,” contains guidance for tests on single failure proof overhead crane handling systems.

- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 19.3, “Regulatory Treatment of Non-Safety Systems.”

- NUREG-1793, “Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design,” Supplements 1 and 2.”

Purpose of Regulatory Guides

The NRC issues regulatory guides to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency’s regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in regulatory guides will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This regulatory guide contains information collection requirements covered by 10 CFR Part 50 and 10 CFR Part 52 that the Office of Management and Budget (OMB) approved under OMB control numbers 3150-0011 and 3150-0151, respectively. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or requirement unless the requesting document displays a currently valid OMB control number.

B. DISCUSSION

Reason for Revision

Regulatory guide 1.68 is being revised to address design qualification tests and add additional preoperational, low-power, and power ascension tests for applications for OLs and COLs of light water reactors (LWRs) under 10 CFR Parts 50 and 52.

Background

The applicant for an OL under 10 CFR Part 50, or a COL under 10 CFR Part 52, is responsible for ensuring that a suitable initial (preoperational and startup) test program will be conducted for the facility. The primary objectives of a suitable program are:

1. Provide additional assurance that the facility has been adequately designed.

2. Validate, to the extent practical, the analytical models.

3. Verify the correctness or conservatism of assumptions used to predict plant responses to anticipated transients and postulated accidents.

4. Provide assurance that construction and installation of equipment in the facility have been accomplished in accordance with design.

5. Familiarize the plant's operating and technical staff with the operation of the facility.

6. Verify by trial use, to the extent practical, that the facility operating procedures and emergency procedures are adequate.

The ITP may include system and component tests, monitoring of SSC performance, and inspection and surveillance test activities for plant SSCs. An ITP satisfying these objectives should provide the necessary assurance that the facility can be operated in accordance with design requirements and in a manner that will not endanger the health and safety of the public.

As previously noted, the ITP consists of preoperational and initial startup tests. "Preoperational testing," as used in this regulatory guide, consists of those tests conducted following completion of construction inspections and tests, but before fuel loading, to demonstrate, to the extent practical, the capability of SSCs to meet the performance requirements to satisfy the design criteria.

Initial startup testing, as used in this regulatory guide, consists of equipment performance tests completed during and after fuel loading. These performance tests are normally completed during fuel loading, pre-critical, initial criticality, low power and power ascension phases to confirm the design bases and demonstrate, to the extent practical, that the plant will operate in accordance with design and that it is capable of responding to anticipated transients and postulated accidents as specified in the FSAR. Section C.1.13.4 "Operational Program Implementation," of RG 1.206 provides guidance on completion of ITP license conditions after the COL is issued. The OL and COL FSAR include this information to enable the NRC staff to perform inspections of ITP license conditions.

After fuel load, licensees should implement operations, testing, maintenance, and monitoring of SSCs consistent with technical specifications (TS) and the requirements and guidance in 10 CFR 50.65, "Requirements for Monitoring the effectiveness of Maintenance at Nuclear Power Plants," RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (Ref. 6), and American Society of Mechanical Engineers (ASME) "*Code for Operation and Maintenance of Nuclear Power Plants,* (OM Code), Division 1, Section IST – Light Water Reactor Nuclear Power Plants," (Ref. 7).

The power ascension test phase of the ITP should be completed in an orderly and expeditious manner. Failure to complete the power ascension test phase within a reasonable time period may indicate inadequacies in the licensee's operating and maintenance capabilities or may result from basic design problems. Also, design- or construction-related problems disclosed during power ascension testing can be more readily rectified if the reactor power production and, consequently, the accumulation of radioactive fission products, have been kept to a minimum during this testing phase. Baseline data on the performance of plant systems obtained and documented early in the plant's life will permit early identification of degradation or undesirable trends.

The ITP should be designed to demonstrate the performance of SSCs and design features that will be used during normal facility operations, as well as the performance of standby systems and features that must function to maintain the plant in a safe condition in the event of malfunctions or accidents. The startup tests should be sequenced so that plant safety is never entirely dependent on the performance of untested SSCs.

As mentioned in the introduction to this regulatory guide, the ITP is required to include suitable testing of all SSCs important to safety. Both Appendices A and B to 10 CFR Part 50 recognize that some SSCs are more important to safety than others. For example, SSCs designated as Seismic Category I by RG 1.29, "Seismic Design Classification," (Ref. 8) are considered to be more important to safety than other SSCs identified as important to safety in the functional design criteria of Appendix A to 10 CFR Part 50. Thus, the NRC does not intend that the same test requirements be established for all SSCs important to safety. Rather, applicants should implement a graded approach to testing in order to provide reasonable assurance, considering the importance to safety of the item, that the item will perform satisfactorily while, at the same time, accomplishing the testing in a cost-effective manner. Documentation (such as procedures and records) associated with testing also should be commensurate with the importance to safety of the item being tested.

To provide for the development and safe execution of the ITP, the applicant should formulate advance plans for the entire testing program before the NRC staff completes its review of the COL application. Because of the complexity of these tests and the significant amount of resources needed to develop and execute the complete program, it is important for the applicant to give early consideration to the following:

1. Define the responsibilities of the organization that will carry out the program. This should include the degree of participation of the principal design organizations in formulating test objectives and acceptance criteria.

2. Develop realistic schedules for preparing detailed testing, plant operating, and emergency procedures. Schedules should be established for conducting the major phases of the test program relative to the expected fuel loading date.

3. Establish methods or plans for providing the necessary resources at the times needed to maintain the schedules. If service contracts are to be used, it is necessary to have sufficient trained staff for good contract management. Hiring and training schedules for the plant's operating and technical staff should be established so that experienced and qualified personnel will be available for the development of testing, operating, and emergency procedures. In addition, it is important to consider the staffing effects that could result from overlapping ITPs at multiunit sites.

4. Formulate administrative controls to govern the development and conduct of the ITP, including controls that will (a) provide for orderly turnover of plant systems and components from construction personnel or other preliminary checkout groups to the preoperational testing group and (b) ensure that general prerequisites (such as completion of construction, construction or preliminary tests, and inspections) will be satisfied before preoperational and/or startup tests of individual systems or components.

5. Establish early plans for using available information about operating experience, including reportable occurrences from other operating power reactors. This is important in developing and conducting the test program to help minimize recurrence of significant problems that could have been avoided by more comprehensive testing.

If the facility is using first-of-a-kind (FOAK) SSCs that are new, unique, or special design feature in the facility, then the in-plant functional testing requirements needed to verify their performance should be identified at an early date to permit the test requirements to be appropriately accounted for in the final test design. For example, some new plant designs licensed under 10 CFR Part 52 have new passive plant design features and FOAK tests for systems that are safety-related or important to safety. Consequently, each new DC, SDA, ML, COL, or OL applicant for an advanced plant should identify new FOAK tests in the given plant. For DC and COL applicants, the NRC will verify that applicable FOAK tests proposed by the applicant are included in the ITP.

A "Prototype Plant," as used in this regulatory guide, is a nuclear reactor used to test design features, such as the testing required under §50.43(e). The prototype plant is similar to a FOAK or standard plant design in all features and size but may include additional safety features to protect the public and the plant staff from the possible consequences of accidents during the testing period. The purpose of the prototype plant is to perform testing of new or innovative design features in the new plant while also using the plant as a commercial nuclear power facility.

Design qualification testing requirements may be met with separate effects or integral system tests, prototype tests, or a combination of tests, analyses, and operating experience. These requirements implement the Commission's policy on proof of performance testing for all advanced reactors with the goal of resolving all safety issues before authorizing construction. Some prototype plant tests or FOAK tests may not be resolved until after fuel load and plant startup. The SDAs, DCs, MLs, COLs, and OLs shall identify certain design qualification tests for the new advanced reactor designs that are used to mitigate accidents during the ITP testing period. For additional details, see Sections 6 and 7 of Appendix A of this guide for regulatory guidance on FOAK tests and design qualification tests.

The ITP should also include testing the performance of non-safety related risk significant systems in passive plant designs that have special regulatory treatment. For additional details on the regulatory treatment of non-safety systems (RTNSS), refer to the RTNSS testing requirements in §50.36(c)(2)(ii)(D), GDCs in Appendix A of 10 CFR Part 50, vendor test recommendations, and the guidance in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.3, "Regulatory Treatment of Non-Safety Systems (Passive Advanced Light Water Reactors)" (Ref. 9). The preoperational tests for RTNSS may also satisfy some ITAAC testing requirements.

The NRC staff's safety evaluations of ITPs are based on FSAR information in the OL application submitted under 10 CFR Part 50. For a COL issued under 10 CFR Part 52, the staff's safety evaluation of the ITP is based on information provided in the FSAR portion of the COL application. The staff uses this information to support decisions to issue a COL. In addition, the NRC uses the information provided in the FSAR as a basis for the inspection activities associated with ITPs. The satisfactory performance of approved test programs helps confirm that adequate safety margins exist, such that there is no undue risk to the health and safety of the public as a result of facility operation.

Harmonization with International Standards

The International Atomic Energy Agency (IAEA) has established a series of safety guides and standards constituting a high level of safety for protecting people and the environment. IAEA safety guides are international standards to help users striving to achieve high levels of safety. Pertinent to this regulatory guide, IAEA Safety Guide NS-G-2.9, "Commissioning of Nuclear Power Plants," issued June 2003, (Ref. 10) addresses tests to be performed during commissioning of a nuclear power plant. This regulatory guide incorporates similar testing program recommendations and is consistent with the basic safety principles provided in IAEA Safety Guide NS-G-2.9.

C. STAFF REGULATORY GUIDANCE

1. Criteria for Selection of Defense in Depth Plant Functions to be Tested

Pursuant to requirements of 10 CFR Parts 50 and 52, each applicant or licensee should prepare and conduct an ITP to demonstrate that the plant can be operated in accordance with design requirements important to safety as defined by Appendix A to 10 CFR Part 50. Suitable tests should be conducted to verify the performance capabilities, as delineated in Appendix A to 10 CFR Part 50, of SSCs that meet one or more of the following criteria:

1. They will be used for shutdown and cool down of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period.

2. They will be used for shutdown and cool down of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions.

3. They will be used to establish conformance with safety limits or limiting conditions for operation that will be included in the facility's technical specifications.

4. They are classified as engineered safety features or will be relied on to support or ensure the operation of engineered safety features within design limits.

5. They will function during a design basis event with credit taken in the accident analysis of the facility, as described in the FSAR Chapter 15.

6. They will be used to process, store, control, or limit the release of radioactive materials.

7. They are relied upon to maintain their structural integrity during normal operation, anticipated transients, simulated test parameters, and design-basis event conditions to avoid damage to safety-related SSCs.

Appendix A to this regulatory guide provides a representative list of systems to be tested, as well as performance capabilities important to safety, as defined by Appendix A to 10 CFR Part 50, which should be demonstrated for light-water cooled nuclear power plants. However, licensees should also conduct in-plant testing to verify the adequacy of the construction, installation, and design for other systems and design features not listed in Appendix A if those systems or design features meet any of the above criteria. The ITP may be developed and implemented using a graded approach, which should ensure that the greatest attention is given to the most safety significant SSCs, such as those considered to be engineered safety features. This graded approach should include testing for non safety related risk significant SSCs.

2. Prerequisites for Testing

The construction or installation of SSCs should be essentially complete (to the degree that outstanding construction items could not be expected to affect the validity of test results). The designated construction-related inspections and tests also should be completed before preoperational tests begin.

The overall test program should also include TS surveillance tests necessary to demonstrate proper operation of interlocks, set points, and other protective features, systems, and equipment.

In addition, administrative controls should be established to ensure adequate retesting of systems or design features that are returned to construction custody, maintained, or modified during or after preoperational testing.

3. Scope, Conditions, and Length of Testing

The testing of SSCs should include, to the extent practical, simulation of the effects of control system and equipment failures or malfunctions that could reasonably be expected to occur during the plant's lifetime. The test program also should include testing to determine that the system and component interactions are in accordance with design. To the extent practical, the plant conditions during the tests should simulate the actual operating, abnormal operating occurrences and emergency conditions to which the SSCs may be subjected.

To the extent practical, the duration of the tests should be sufficient to permit equipment to reach its normal equilibrium conditions (e.g., temperatures and pressures) and, thus, decrease the probability that failures, including "run-in" type failures, will occur during plant operation. "Run-in" type failures are early "burn-in" failures where SSCs exhibit high failure rates when first introduced or operated due to defects, design errors, and other early sources of potential failures such as handling and installation errors. The ITP can greatly reduce the possibility of SSCs failing early in plant operation by identifying and correcting these early sources of failures.

4. Procedures

The licensee should develop detailed test procedures to implement the ITP, including preoperational, initial criticality, low power and power ascension tests. The SSCs should be tested using procedures that include appropriate checklists and signature blocks to control the sequence and performance of testing. The test procedures should be developed and reviewed by personnel with appropriate technical backgrounds and experience and should receive final approval by persons in designated management positions within the applicant's or licensee's organization. In addition, each test procedure should include acceptance criteria that account for the uncertainties used in transient and accident analyses. Principal design organizations should participate in establishing those test acceptance criteria and related performance requirements.

Test procedures should ensure that temporary instrument cables and test leads used during the startup test phase are routed in a manner that will not compromise electrical separation criteria. Available information on operating experience, including reportable occurrences at operating power reactors, also should be used appropriately in developing and executing the test procedures.

Approved test procedures for satisfying FSAR testing commitments should be made available to the NRC approximately 60 days before their intended use.

Before fuel loading, the results of completed preoperational tests should be evaluated by personnel or groups that the licensee has designated to determine if any tests and acceptance criteria have not been met. Appropriate remedial actions, including retesting, should be taken if the acceptance criteria associated with preoperational tests are not satisfied.

5. Schedule

Sufficient time should be scheduled to perform orderly and comprehensive testing. Previous applicants' or licensees' schedules for conducting the preoperational and initial startup phases have typically allowed a minimum of approximately 9 months for preoperational testing and 3 months for initial start-up testing. Significantly shorter time periods should be justified.

6. Participation of Plant Operating and Technical Staff

The licensee's plant operating and plant technical staff should participate, to the extent practical, in developing and conducting the ITP and evaluating the test results.

7. Trial Testing of Plant Emergency, Operating, and Surveillance Test Procedures

Plant emergency, operating, and surveillance test procedures should, to the extent practical, be developed, trial tested, and corrected during the ITP, before fuel load, to establish their adequacy. Trial testing may include low power tests that simulate real and off-normal plant conditions after fuel load, when necessary. This can be accomplished by having plant operators trained in and using the incorporated plant emergency, operating, and surveillance test procedures to the maximum extent possible during the ITP. Additionally, trial testing of emergency, operating, and surveillance test procedures should be incorporated into a plant referenced simulator that meets the requirements of 10 CFR 55.46(c) and is used in the operator training program.

In evaluating plant conditions after each stage of power ascension, the test team should review the results and confirm whether the testing can proceed to the next power level. As part of this review, the test team should consider plant conditions, such as transients, core anomalies, or plant stability issues, and plant conditions that would impact radiation safety under 10 CFR Part 20 (doses to workers and pubic), effluent concentration limits under Appendix B to Part 20, design objectives and as low as (is) reasonably achievable (ALARA) provisions in 10 CFR Part 50.36a and 10 CFR Part 50, Appendix I, in controlling and monitoring liquid and gaseous effluents.

8. Milestones and Power Hold Points

Licensees should establish appropriate hold points at selected milestones throughout the power ascension test phase to ensure that designated personnel or groups evaluate and approve relevant test results before proceeding with the power-ascension test phase. At a minimum, licensees should establish hold points at approximately 25-percent, 50-percent, and 75-percent power-level test conditions for pressurized-water reactors, and at appropriate power to flow test conditions for boiling-water reactors.

9. Test Reports

The preoperational and startup testing procedures and test results should be retained as part of the plant's historical record in accordance with 10 CFR 50.36, "Technical Specification," 10 CFR 50.71, "Maintenance of Records, Making of Reports," and 10 CFR 50, Appendix B, Criterion XVII, "Test Records." The test reports should also include test results associated with license conditions in the plant specific ITP. In addition, a summary of the startup testing should be included in a startup report. This summary should include the following information:

1. a description of the method and objectives for each test;

2. a comparison of applicable test data with the related acceptance criteria, including the systems' responses to major plant transients (such as reactor scram and turbine trip);

3. design- and construction-related deficiencies discovered during testing, system modifications and corrective actions required to correct those deficiencies, and the schedule for implementing these modifications and corrective actions unless previously reported to the NRC;

4. justification for acceptance of systems or components that are not in conformance with design predictions or performance requirements; and

5. conclusions about system or component adequacy.

D. IMPLEMENTATION

The purpose of this section is to provide information on how applicants and licensees[1] may use this guide and information regarding the NRC's plans for using this regulatory guide. In addition, it describes how the NRC staff complies with 10 CFR 50.109, "Backfitting" and any applicable finality provisions in 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

Use by Applicants and Licensees

Applicants and licensees may voluntarily[2] use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this regulatory guide may be deemed acceptable if they provide sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations. Current licensees may continue to use guidance the NRC found acceptable for complying with the identified regulations as long as their current licensing basis remains unchanged.

Licensees may use the information in this regulatory guide for actions which do not require NRC review and approval such as changes to a facility design under 10 CFR 50.59, "Changes, Tests, and Experiments." Licensees may use the information in this regulatory guide or applicable parts to resolve regulatory or inspection issues.

Use by NRC Staff

The NRC staff does not intend or approve any imposition or backfitting of the guidance in this regulatory guide. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this regulatory guide, unless the licensee makes a change to its licensing basis. The NRC staff does not expect or plan to request licensees to voluntarily adopt this regulatory guide to resolve a generic regulatory issue. The NRC staff does not expect or plan to initiate NRC regulatory action which would require the use of this regulatory guide. Examples of such unplanned NRC regulatory actions include issuance of an order requiring the use of the regulatory guide, requests for information under 10 CFR 50.54(f) as to whether a licensee intends to commit to use of this regulatory guide, generic communication, or promulgation of a rule requiring the use of this regulatory guide without further backfit consideration.

During regulatory discussions on plant specific operational issues, the staff may discuss with licensees various actions consistent with staff positions in this regulatory guide, as one acceptable means of meeting the underlying NRC regulatory requirement. Such discussions would not ordinarily be considered backfitting even if prior versions of this regulatory guide are part of the licensing basis of the facility. However, unless this regulatory guide is part of the licensing basis for a facility, the staff may not represent to the licensee that the licensee's failure to comply with the positions in this regulatory guide constitutes a violation.

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1 In this section, "licensees" refers to licensees of nuclear power plants under 10 CFR Parts 50 and 52; and the term "applicants," refers to applicants for licenses and permits for (or relating to) nuclear power plants under 10 CFR Parts 50 and 52, and applicants for standard design approvals and standard design certifications under 10 CFR Part 52.

2 In this section, "voluntary" and "voluntarily" means that the licensee is seeking the action of its own accord, without the force of a legally binding requirement or an NRC representation of further licensing or enforcement action.

If an existing licensee voluntarily seeks a license amendment or change and (1) the NRC staff's consideration of the request involves a regulatory issue directly relevant to this new or revised regulatory guide and (2) the specific subject matter of this regulatory guide is an essential consideration in the staff's determination of the acceptability of the licensee's request, then the staff may request that the licensee either follow the guidance in this regulatory guide or provide an equivalent alternative process that demonstrates compliance with the underlying NRC regulatory requirements. This is not considered backfitting as defined in 10 CFR 50.109(a)(1) or a violation of any of the issue finality provisions in 10 CFR Part 52.

Additionally, an existing applicant may be required to comply to new rules, orders, or guidance if 10 CFR 50.109(a)(3) applies.

If a licensee believes that the NRC is either using this regulatory guide or requesting or requiring the licensee to implement the methods or processes in this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfit appeal with the NRC in accordance with the guidance in NUREG-1409, "Backfitting Guidelines," (Ref. 11) and the NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 12).

REFERENCES[3]

1. Code of Federal Regulations (CFR), *Title 10, Energy*, Part 50, "Domestic Licensing of Production and Utilization Facilities."

2. CFR, *Title 10, Energy*, Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

3. U.S. Nuclear Regulatory Commission (NRC), Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," Washington, DC.

4. NRC, RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," NRC, Washington, DC.

5. NRC, RG 1.215, "Guidance for ITAAC Closure under 10 CFR Part 52," Washington, DC.

6. NRC, RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Washington, DC.

7. American Society of Mechanical Engineers (ASME), "*Operation and Maintenance of Nuclear Power Plants* (OM Code), Division 1, "Section IST – Light Water Reactor Nuclear Power Plants," ASME, New York, NY.[4]

8. NRC, RG 1.29, "Seismic Design Classification,", Washington, DC.

9. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.3, "Regulatory Treatment of Non-Safety Systems," Washington DC.

10. International Atomic Energy Agency (IAEA), Safety Guide NS-G-2.9, "Commissioning of Nuclear Power Plants," issued June 2003, IAEA, Vienna, Austria, 2003.[4]

11. NRC, NUREG-1409, "Backfitting Guidelines," July 1990, NRC, Washington, DC. (ADAMS Accession No. ML032230247)

12. NRC, Management Directive (MD) 8.4, "Management of Facility-Specific Backfitting and Information Collection," Washington, DC.

13. ASME, Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," 2004, ASME, New York, NY.[5]

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3 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at: http://www.nrc.gov/reading-rm/doc-collections/. The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD; the mailing address is USNRC PDR, Washington, DC 20555; telephone 301-415-4737 or 800-397-4209; fax 301-415-3548; and e-mail pdr.resource@nrc.gov.

4 Copies of International Atomic Energy Agency (IAEA) documents may be obtained through their Web site: WWW.IAEA.Org/ or by writing the International Atomic Energy Agency P.O. Box 100 Wagramer Strasse 5, A-1400 Vienna, Austria. Telephone (+431) 2600-0, Fax (+431) 2600-7, or E-Mail at Official.Mail@IAEA.Org

14. NRC, RG 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing,” Washington, DC.

15. NRC, RG 1.118, “Periodic Testing of Electric Power and Protection Systems,” Washington, DC.

16. NRC, RG 1.41, “Preoperational Testing of Redundant Onsite Electric Power Systems To Verify Proper Load Group Assignments,” Washington, DC.

17. NRC, RG 1.68.1, “Initial Test Program of Condensate and Feedwater Systems for Light Water Reactors,” Washington, DC.

18. NRC, RG 1.155, “Station Blackout,” Washington, DC.

19. NRC, RG 1.9. “Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants,” Washington, DC.

20. NRC, RG 1.79, “Preoperational Testing of Emergency Core Cooling Systems for Pressurized-Water Reactors,” Washington, DC.

21. NRC, RG 1.79.1, “Initial Test Program of Emergency Core Cooling Systems for New Boiling Water Reactors,” Washington, DC.

22. NRC, RG 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Washington, DC.

23. NRC, RG 1.140, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Washington D.C.

24. NRC, NUREG-1793, “Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design,” Supplements 1 and 2,” Washington DC. (Multiple ADAMS Accession numbers, see http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1793/)

25. NRC, RG 1.45, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage,” Washington, DC.

26. Nuclear Energy Institute (NEI) 97-06, “Steam Generator Program Guidelines,” Rev. 2, dated September 9, 2005.[6] (ADAMS Accession No. ML052710007) (Rev. 3 is ML111310708)

27. Electric Power Research Institute (EPRI) TR-1008219 “PWR Primary-to-Secondary Leak Guidelines,” Revision 3, December 2004.[7] (ADAMS Accession No. ML050840534)

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5 Copies of American Society of Mechanical Engineers (ASME) standards may be purchased from ASME, Three Park Avenue, New York, New York 10016-5990; telephone 800-843-2763. Purchase information is available through the ASME Web-based store at http://www.asme.org/Codes/Publications/.

6 Publications from the Nuclear Energy Institute (NEI) are available at their Web site: http://www.nei.org/ or by contacting the headquarters at Nuclear Energy Institute, 1776 I Street NW, Washington DC 20006-3708, Phone: 202-739-800, Fax 202-785-4019.

7 Copies of Electric Power Research Institute (EPRI) documents may be obtained by contacting the Electric Power Research Institute, 3420 Hillview Avenue, Palo Alto, CA 94304, Telephone: 650-855-2000 or on-line at

28. NRC, NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," Washington, DC, May 1979. (ADAMS Accession No. ML110450636)

29. NRC, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," Washington, DC, July 1980. (ADAMS Accession No. ML070250180)

30. NRC, RG 1.68.3, "Preoperational Testing of Instrument and Control Air Systems," Washington, DC.

31. CFR, *Title 10, Energy*, Part 20, "Standards for Protection against Radiation."

32. NRC, RG 1.69, "Concrete Radiation Shields And Generic Shield Testing For Nuclear Power Plants," Washington, DC.

33. NRC, RG 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," Washington, DC.

34. NRC, RG 1.68.2, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants," Washington, DC.

35. Federal Register Notice (FRN), "Licenses, Certifications, and Approvals for Nuclear Power Plants, Final Rule," (Statements of Consideration) Volume 72, Number 166, (72 FR 49352), August 28, 2007.[8]

36. NRC, Inspection Manual Chapter (IMC) 2513, Appendix A, "Light Water Reactor Preoperational Testing Phase," NRC, Washington DC.

37. NRC, IMC 2513, Appendix B, "Light Water Reactor Operational Preparedness Phase," Washington DC.

38. NRC, IMC 2514, "Light Water Reactor Inspection Program – Startup Testing Phase," Washington DC. (http://www.nrc.gov/reading-rm/doc-collections/insp-manual/manual-chapter/mc2514.pdf)

39. NRC, IMC 2514, Appendix A, "Startup Test Program Inspection Procedures," Washington DC. (http://www.nrc.gov/reading-rm/doc-collections/insp-manual/manual-chapter/mc2514a.pdf)

40. NRC, IMC 2503 "Construction Inspection Program: Inspections of Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) Related Work," Washington DC.

41. NRC, IMC 2504, "Construction Inspection Program – Inspection of Construction and Operational Programs," Washington DC.

42. NRC, RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," Washington, DC.

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http://my.epri.com/portal/server.pt.

8 Printed copies of Federal Register notices are available for a fee from the U.S. Government Printing Office, 732 N Capitol Street, NW, Washington, DC 20401, telephone (866) 521-1800. Electronic versions may be downloaded for free from the Government Printing Office Web site: http://www.gpo.gov/fdsys/.

APPENDIX A

INITIAL TEST PROGRAM

This appendix incorporates information relevant to applications for operating licensees (OLs) submitted under 10 CFR Part 50 and to applications for design certifications (DC), standard design approvals (SDA), manufacturing licenses (ML), and combined license (COL) applications submitted under the applicable appendix to 10 CFR Part 52. For OL applications under 10 CFR Part 50, the applicant must describe the initial test program (ITP) in the FSAR in accordance with §50.34(b)(6)(iii) and §50.43(e). For COL applications under 10 CFR Part 52, the applicant must describe the ITP, in accordance with §52.79(a)(28). The requirements in 10 CFR 52.47 and 10 CFR 52.137 do not require an SDA or DC applicant to submit an ITP. However, SDA and DC applications should include proposed testing activities for the ITP to support the COL applications. The COL applicant is responsible for identifying the remaining portions of the site-specific ITP tests and all of the applicable tests for each ITP test phase.

Sections A-1 through A-5 of this appendix address the specific tests required for each of the five phases of the ITP, including (1) Preoperational Testing, (2) Initial Fuel Loading and Pre-Criticality Testing, (3) Initial Criticality Testing, (4) Low-Power Testing, and (5) Power-Ascension Testing. In addition, Sections A-6 and A-7 contain guidance for implementing first-of-a-kind tests and design qualification tests for new advanced reactors. No particular significance should be attached to the order in which the tests are listed. In general, those listed in Section 1, "Preoperational Testing" should precede those listed in Section 2, "Initial fuel loading and Pre-Critical Tests" and so on. The regulations in 10 CFR 50.34 and 10 CFR 52.79 require that DC, SDA, COL, ML, and OL applications identify certain design qualification tests for new reactor designs that are used to mitigate accidents or severe transients during the ITP testing period.

A-1. Preoperational Testing

For new plants licensed under10 CFR Part 52, the ITAAC incorporated into the COL comprises a set of preapproved verifications that licensees must meet before fuel load. In many cases, the ITAAC testing requirements are the same or similar to the preoperational tests required to be completed as part of the ITP. Preoperational testing not associated with ITAAC should be performed in accordance with this regulatory guide and the guidance in the ASME OM Code, Division 1 (Ref. 7). In cases in which preoperational tests are performed that demonstrate the acceptance criteria of both the ITP and the ITAAC the test results should be recorded under both programs.

Following plant construction, testing should be accomplished to demonstrate the proper performance of SSCs and design features in the assembled plant. To ensure valid test results the preoperational tests should not proceed until construction of the system has been essentially completed and the designated construction inspections and tests have been satisfactorily completed.

Construction inspections and tests typically consist of activities such as initial instrument calibration, flushing, cleaning, wiring, continuity and separation checks, hydrostatic pressure tests, and functional tests of components. In some cases, licensees may also perform final calibration of digital interfaces after the construction phase and during the preoperational test phase. For additional details, see Section J, "Instrumentation and Control Systems" in this appendix.

Preoperational tests should demonstrate that SSCs will operate in accordance with their design during the preoperational test phase. Testing should include, as appropriate and practical, manual and automatic operation, operation of systems and their components, automatic operation, operation in all alternate or secondary modes of control, and operation and verification tests to demonstrate expected operation following a loss of power sources and in degraded modes for which the systems are designed to remain operational. Tests also should include, as appropriate, verifications of the proper functioning of instrumentation and controls, permissive and prohibit interlocks, and equipment protective devices of which malfunction or premature actuation may shut down or defeat the operation of systems or equipment.

The ASME OM Code (Ref. 7) as incorporated by reference in 10 CFR 50.55a specifies pre-service testing and in-service testing requirements for pumps, valves, and dynamic restraints. The operational readiness of pumps, valves, and dynamic restraints needs to be demonstrated before relying on those components to perform their safety functions. Pre-service testing of pumps, valves, and dynamic restraints might be accomplished as part of the preoperational testing activities.

Preoperational tests might be required by conditions in the license that the NRC has issued for nuclear power plants under construction. For example, the NRC specified license conditions for some new nuclear power plants licensed in accordance with 10 CFR Part 52 that include preoperational test requirements for pyrotechnic-actuated valves. Tests performed to satisfy license conditions are also accomplished as part of the preoperational testing activities.

System vibration, expansion (in discrete temperature step increments), and restraint tests also should be conducted. This testing should include verification (by observations and measurements), as appropriate, that piping and components have adequate clearances to accommodate potential water hammer induced movements, vibrations, and expansions acceptable for (1) Class 1, 2, and 3 systems, as defined by the ASME Boiler and Pressure Vessel Code (ASME B&PV Code) Section III (Ref. 13), (2) other high energy piping systems inside Seismic Category I structures, (3) high energy portions of systems whose failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable level, and (4) Seismic Category I portions of moderate energy piping systems located outside containment. In addition, inspections or testing may be conducted for flow-induced vibration loads on components that must maintain their structural integrity to avoid damage to safety-related SSCs.

The SSCs and tests listed in Sections A-1.a. through A-1.p. below are representative of the plant features that should undergo preoperational testing. This list is provided to indicate the extent of testing necessary to demonstrate that the facility can be operated in accordance with design requirements. In general, the tests listed below make no distinction between boiling water reactors (BWRs) and pressurized water reactors (PWRs). An applicant may combine tests of items listed in this appendix and should include preoperational tests of the listed SSCs as appropriate for the facility. Preoperational tests should not be limited to the list provided below because additional or different tests may be dictated by the particular plant design or the nomenclature applied to plant systems and features.

A-1.a. <u>Reactor Coolant System</u>

The reactor coolant system (RCS) includes all pressure containing components (such as pressure vessels, piping, pumps, and valves) within the reactor coolant pressure boundary, as defined in 10 CFR 50.2, "Definitions." For the reactor coolant system the following tests should be performed:

1. *Integrated Systems Tests*. Perform expansion and restraint tests to confirm the acceptability of clearances and displacements of vessels; piping; piping hangers; and seismic and other hold down, support, or restraining devices in the as built system during normal hot functional testing plant conditions. The system should be subjected to hot and/or cold testing with simultaneous operation of auxiliary systems.

2. *Component Tests*. The following RCS components should undergo appropriate tests and measurements:

   a. pressurizer, including pressurizer heaters and pressurizer spray and throttle valves (PWR);

   b. pumps, motors, and associated power sources;

   c. steam generators;

   d. pressure relief valves, block valves, and associated dump tanks, as well as supports and restraints for discharge piping;

   e. main steam isolation valves;

   f. other valves;

   g. instrumentation used to monitor system performance or perform permissive and prohibit interlock functions;

   h. reactor vessel and reactor internal vent valves;

   i. safety and relief valves including testing of acoustic monitors used to detect leakage downstream from safety and relief valves;

   j. jet pumps;

   k. heat exchangers; and

   l. pyrotechnic-actuated squib valves.

3. *Vibration Tests*. The reactor internals[1] and other components, such as piping systems, heat exchangers, and rotating machinery should undergo vibration testing.

4. *Pressure Boundary Integrity Tests*. All pressure boundaries should be subjected to hydrostatic tests to obtain baseline data for subsequent inservice inspection and testing.

A-1.b. <u>Reactivity Control Systems</u>

1. *Control Rod System Tests*:

   a. Demonstrate normal operation and reactor trip or scram capability of the control rods and control rod drive system.

   b. Demonstrate proper operation of functions such as control rod withdrawal inhibit features, runback features, rod withdrawal sequence control devices, and rod worth minimizers.

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1 Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," (Ref. 14) should be used as guidance for vibration monitoring of reactor internals and other components.

c. Demonstrate proper operation of rod position instrumentation and proper interaction of the control rod drive system with other systems and design features, such as automatic reactor power control systems and refueling equipment.

d. Demonstrate proper operation, including the correct failure mode on loss of power, for the control rod drive system and proper operation of system alarms.

2. *Chemical Control System Tests (PWR Only)*:

a. Verify proper blending of boron solution and water; uniform mixing; adequacy of sampling and analytical techniques; operation of heaters and heat tracing; and operation of instrumentation, controls, interlocks, and alarms.

b. Demonstrate by test the proper rate of emergency boration injection into the reactor coolant system and the rate of dilution from the primary system.

c. Verify the redundancy, electrical independence, and operability of system components. Demonstrate the correct failure mode on loss of power to system components.

3. *Standby Liquid Control System (SLCS) Tests*.

a. Demonstrate proper operation of the system with demineralized water in the system flow paths and discharge.

b. Verify the proper boron enrichment, flow rates, tank volume and concentration of neutron absorber solution before entry into the technical specification (TS) mode when SLCS operability is required.

c. Demonstrate the operability of instrumentation, controls, interlocks, alarms, and equipment protective devices in valve controls.

d. Verify the operability of electrical room heaters, air spargers, and/or heat tracing.

e. Verify proper operation of the SLCS nitrogen pressurization system, if provided.

f. Conduct test firings of squib-actuated valves and demonstrate design injection capability.

g. Conduct tests, as appropriate, to verify redundancy and electrical independence.

A-1.c. Reactor Protection System and Engineered Safety Features Actuation Systems

1. Verify (by testing) the response time of each of the protection channels, including sensors.[2] Acceptance criteria for the response time of the protection channels should account for the response time of the associated hardware between the measured variable and the input to the sensor (e.g., snubbers, sensing lines, and flow-limiting devices).

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2 Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems," (Ref. 15) provides a test criterion that is also acceptable for preoperational testing of protection channels, including sensors.

2. Verify proper operation in all combinations of logic; calibration and operability of primary sensors; proper trip and alarm settings; proper operation of permissive, prohibit, and bypass functions; and operability of bypass switches.

3. Demonstrate redundancy, electrical independence,[3] coincidence, and safe failure on loss of power.

4. If appropriate for the facility design, demonstrate the operability of backup scram solenoid valves and devices, including detectors, logic, and final control elements to protect the facility for anticipated transients without scram.

A-1.d. <u>Residual or Decay Heat Removal Systems</u>

1. Verify the operability of systems and design features provided or relied on to dissipate or channel thermal energy from the reactor to the atmosphere or to the main condenser or other systems following off-normal conditions or anticipated transients, including reactor scram.

2. Verify the operability of systems and design features provided for makeup of coolant, to dissipate residual heat, to cool the reactor down to a cold-shutdown condition, and to maintain long-term cooling.

3. Tests should be conducted, as appropriate, to verify redundancy and electrical independence (See Footnote 3). The following list illustrates the systems and components that should be tested:

   a. turbine bypass valves
   b. steam line atmospheric dump valves
   c. relief valves
   d. safety valves
   e. decay and heat removal system
   f. reactor core isolation cooling (RCIC) system
   g. main steam isolation valves, branch steam isolation valves, and non-return valves
   h. auxiliary feedwater systems
   i. condensate storage system
   j. emergency cooling towers ultimate heat sink
   k. cooling water systems
   l. isolation condenser system

Testing should include demonstrations that the systems will meet design performance requirements at approximately normal operating primary and secondary coolant system pressures and temperatures and over the range of expected reactor pressure vessel and steam generator levels. Operability of system pumps, valves, controls, and instrumentation should be

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3 Regulatory Guide 1.41, “Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments,” (Ref. 16) should be used as guidance for appropriate tests.

demonstrated and, to the extent practical, testing should provide reasonable assurance that flow instabilities (e.g., water hammer), including flow inducted vibration, will not occur in system components or piping, or inside the reactor pressure vessel, steam generators, and other heat exchangers during normal system startup and operation.

A-1.e. **Power Conversion System**

The power conversion system includes all components provided to channel reactor thermal energy during normal operation from the boundaries of the reactor coolant system to the main condenser, and those systems and components provided for return of condensate and feedwater[4] from the main condenser to complete the cycle. Appropriate system expansion, restraint, and operability tests should be conducted, to the extent practical, for the following systems and components:

1. steam generators
2. main steam system
3. main steam isolation valves
4. steam generator pressure relief and safety valves
5. steam extraction system
6. turbine stop, control, bypass, and intercept valves
7. main condenser hotwell level control system
8. condensate system
9. feedwater system
10. feedwater heater and drain systems
11. makeup water and chemical treatment systems
12. main condenser off gas system used to maintain condenser vacuum

A-1.f. **Waste Heat Rejection Systems**

The waste heat rejection systems include systems and components provided to remove unused or wasted thermal energy from systems (such as the power conversion and residual heat removal system), and channel or direct this energy to the environment. Tests should be conducted, as appropriate, to verify redundancy and electrical independence (See Footnote 3). Appropriate system operability tests should also be conducted to demonstrate, to the extent practical, that the following waste heat rejection systems and components, including associated instrumentation and controls, will perform as designed:

1. circulating water system
2. ultimate heat sink cooling towers and associated auxiliaries
3. raw water and service water cooling ultimate heat sink systems

4 Regulatory Guide 1.68.1, "Initial Test Program of Condensate and Feedwater Systems for Light Water Reactors," (Ref. 17) should be used as guidance for appropriate tests.

A-1.g. <u>Electrical Systems</u>

The plant electrical systems include the normal alternating current (ac) power distribution system, the emergency ac power distribution system including vital buses, the emergency ac power supplies or sources, and the direct current (dc) systems. Appropriate system and component tests should be conducted to verify, to the extent practical, that these systems will operate in accordance with design:

1. *Normal AC Power Distribution System.* Demonstrate proper operation of:

   a. protective devices

   b. initiating devices

   c. relaying and logic

   d. transfer schemes and trip devices

   e. permissive and prohibit interlocks

   f. instrumentation and alarms, and load shedding features

   Testing should also be conducted to demonstrate proper operation and load carrying capability of:

   a. breakers

   b. motor controllers

   c. switchgear

   d. transformers

   e. cables

   This testing should simulate, as closely as practical, actual service conditions (e.g., fully loading motor control centers and operation of supplied loads at rated conditions). Redundancy and electrical independence should be demonstrated where appropriate (see footnote 3).

   Tests should demonstrate that the integrated system will perform as designed in response to simulated partial and full losses of offsite power sources. Tests also should demonstrate degraded protection systems designed to transfer from offsite to onsite power sources during degraded voltage conditions.

2. *Emergency AC Power Distribution System.* Demonstrate proper operation of:

   a. protective devices

   b. relaying and logic

   c. transfer and trip devices

   d. permissive and prohibit interlocks

   e. instrumentation and alarms

   f. load shedding or stripping features

Testing should also be conducted to demonstrate proper operation and load carrying capability of:

a. breakers

b. motor controllers

c. switchgear

d. transformers

e. cables

This testing should simulate, as closely as practical, actual service conditions (e.g., fully loading motor control centers and operation of supplied loads at rated conditions). In addition, tests should demonstrate that emergency or vital loads will start in the proper sequence and operate under simulated accident conditions with both the normal (preferred) ac power source and the emergency (standby) power source.

Emergency loads should also be tested to demonstrate that they can start and operate with the maximum and minimum design voltage available. To the extent practical, the testing of emergency or vital loads should be conducted for a sufficient period of time to provide assurance that equilibrium conditions are attained. System redundancy and electrical independence should be verified by appropriate testing.

Loads supplied from the system, such as motor-generator sets with flywheels, which are designed to provide non-interruptible power to plant loads should be tested to demonstrate proper operation. If applicable for the facility design, testing should include under frequency and under voltage relays associated with such motor-generator sets. Full load tests for vital buses should be conducted using normal and emergency power supplies to the bus. Testing also should demonstrate the adequacy of the plant's emergency and essential lighting system. In addition, tests should be conducted to demonstrate proper operation of indicating and alarm devices used to monitor the availability of the emergency power system in the control room.

3. *Emergency or Standby AC Power Supplies.* Conduct appropriate tests for emergency ac power supplies to establish system reliability targets[5] to meet the requirements in 10 CFR 50.63. The testing includes verifying redundancy, electrical dependence, and proper voltage and frequency regulation under transient and steady state conditions. Auxiliary systems (such as those used for starting, cooling, heating, ventilating, lubricating, and fueling) should be appropriately tested to demonstrate that their performance is in accordance with design.

Testing should be conducted for a sufficient period of time to ensure that equilibrium conditions are attained. Testing also should demonstrate the proper loading logic, correct set points for trip devices, and proper operation of initiating devices and permissive and prohibit interlocks and should also demonstrate redundancy and electrical independence (see footnote 3). Emergency loads supplied should be confirmed to be in agreement with design sizing assumptions used for the power supplies.[6]

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5 Regulatory Guide 1.155, "Station Blackout," (Ref. 18) should be used as guidance for testing emergency ac power supply reliability targets.

6 Regulatory Guide 1.9, "Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants," (Ref. 19) should be used as guidance for applicable tests.

4. *DC System.*

a. Demonstrate proper calibration and trip settings of protective devices, including relaying, and proper operation of permissive and prohibit interlocks.

b. Demonstrate the design capability of batteries, battery chargers, transfer devices, converters, inverters, and emergency lighting systems. Testing should also be conducted to demonstrate the capability of batteries and battery charges, proper operation of breakers, transfer devices, converters, inverters, and cables. This testing should simulate, as closely as practical, actual service conditions.

c. Demonstrate operation of instrumentation, alarms, and ground detection instrumentation.

d. Demonstrate redundancy and electrical independence (See Footnote 3) and show that actual total system amperage loads are in agreement with design loads.

e. Demonstrate that the battery bank voltage minimum limit and individual cell limits are not exceeded. A discharge test of each battery bank should be conducted at full load and for design duration to demonstrate that the battery bank voltage minimum limit and individual cell limits are not exceeded.

A-1.h. <u>Engineered Safety Features</u>

Engineered safety features (ESFs) are those plant design features provided to prevent, limit, or mitigate the consequences of postulated accidents that are described in the safety analysis report. For the purpose of this guide, ESFs include features that prevent accidents from occurring or bound accident assumptions (such as cold water injection interlocks for PWRs, and rod worth minimizers for BWRs). Because ESFs vary for different plant designs, the list below only illustrates those that are commonly used to prevent, limit, or mitigate the consequences of postulated accidents. If the subject plant design provides other ESFs in addition to (or other than) those listed below, they also should be appropriately tested. Additionally, it should be noted that other categories of systems listed in Section A-1 of this appendix include plant features commonly designated as ESFs, which should be appropriately tested, such as the emergency ac power distribution system (Section A-1.g.(2)), emergency or standby ac power supplies (Section A-1.g.(3)), the dc system (Section A-1.g.(4)), and primary and secondary containments (Section A-1.i).

The testing of ESFs should demonstrate that such features will perform satisfactorily in all expected operating configurations or modes. Testing should include demonstrations of correct logic and set points, as well as proper operation of initiating devices, bypasses, permissive and prohibit interlocks, and equipment protective devices that could shut down or defeat the operation or functioning of such features. This may include testing safety related DC systems in passive plant ESF systems used to bring the plant to hot standby conditions. Concurrent testing of systems or features provided to ensure or support the operation of ESFs should also be conducted to demonstrate that they meet design requirements with the minimum number of operable components available with which these systems are designed to function. Examples of these types of systems are heating, ventilation, and air conditioning systems used to maintain the environment within design limits in the spaces housing ESFs; cooling water and seal injection systems; and protected compressed gas supplies. Appropriate tests should also be conducted to verify the functioning of protective devices such as leak tight covers, structures, or housings (low-pressure pneumatic or vacuum tests) provided to protect ESFs from flooding or keep full systems used to prevent water hammer and possible damage to fluid systems.

Tests should be conducted, as appropriate, to verify redundancy and electrical independence (see footnote 3). The following list illustrates the systems and components that should be tested:

1. *Emergency Core Cooling Systems (ECCS)*[7, 8]

   a. Perform expansion and restraint tests.

   b. Demonstrate operability using normal and emergency power supplies.

   c. Demonstrate operability in all modes of operation, including design pump/system runout conditions and injection at required flow rate and pressure.

   d. Demonstrate operability of interlocks and isolation valves provided for overpressure protection for low-pressure cooling systems connected to the reactor coolant system.

   e. Demonstrate operability, including proper flow rates, for systems used to dilute boron in the reactor vessel during post loss of coolant accident (LOCA) long term cooling.

   Specific ECCS preoperational tests in PWRs are provided in Regulatory Guide (RG) 1.79 (see footnote 7). Specific ECCS preoperational and startup tests in BWRs are provided in RG 1.79.1 (see footnote 8).

2. *Automatic Depressurization System (ADS)*. Testing should include such factors as accumulator capacity, relief valve capacity, and operability using all alternative power and pneumatic supplies. This preoperational test demonstrates proper operation of circuitry connections, ADS instrumentation and control logic functions and safety system logic functions, including integrated automatic decision making and trip logic functions associated with safety actions of ADS relief and pyrotechnic-actuated valves. The operational readiness of each pyrotechnic-actuated valve will be verified by initiating the actuator control circuitry to demonstrate acceptable electrical parameters with the charge removed from the valve. A sample of the pyrotechnic charges from the valve population will be fired in a test fixture to demonstrate their design-basis capability. The pyrotechnic-actuated valves will receive external and internal examinations for structural integrity and presence of foreign material and fluids. ADS initiation is provided to reduce pressure during LOCAs when high pressure injection systems are unable to restore reactor vessel water level. This allows makeup of cooling water from low pressure injection systems or passive core cooling systems.[8]

3. *Containment Post-accident Heat Removal Systems*. Testing of the containment spray system should include demonstrations that the spray nozzles, spray headers, and piping

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7 Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors," (Ref. 20) provides specific guidance for PWRs.

8 Regulatory Guide 1.79.1, "Initial Test Program of Emergency Core Cooling Systems for New Boiling Water Reactors," (Ref. 21) provides specific guidance for BWRs.

are free of debris; chemical addition systems operate properly; and proper transfer to the recirculation phase can be accomplished.

4. *Containment Combustible Gas Control System* (this includes the backup purge system). For containment combustible gas control systems located outside containment, testing should include demonstration that containment hydrogen monitoring is functional without the operation of the hydrogen recombiner. For hydrogen recombiners shared between plants or sites, tests should include demonstration that the shared recombiner can be transported and connected to the combustible gas control system within the time stated in the final safety analysis report (FSAR).

5. *Cold Water Interlocks* (including logic, circuitry, and final control devices used to prevent cold water injection into the reactor vessel).

6. *Air Return Fans* (used in ice condenser containments) *and Suppression Pool Makeup Systems* (used in BWR Mark III containments).

7. *Ventilation, Recirculation, and Filter Systems* (provided to minimize radioactive releases as a result of postulated accidents, including fuel handling accidents).[9,10]

8. *Tanks and Other Sources of Water Used for the ECCS*. Testing should include demonstrations of proper operation of associated alarms, indicators, controls, heating and chilling systems, and valves.

9. *Containment Recirculation Fans* (if used as part of post-accident containment heat removal systems). Testing should include demonstrations that the fans can operate in accordance with design requirements at the containment design peak accident pressure.

10. *Ultimate Heat Sink:* Testing should verify pump flow rates meet design acceptance criteria for removing plant heat loads during normal plant conditions and emergency design-basis accident conditions.

11. *Passive Core Cooling System*[11] Preoperational testing of the passive core cooling system should be performed and verified using the guidance in RG 1.79. Some example tests include:

    a. FOAK testing of pyrotechnic squib valves
    b. FOAK testing of other new valve designs (e.g., nozzle check valves)

12. *Passive Containment Cooling System* [11] Preoperational testing of the passive containment cooling system should be performed to verify flow and heat removal rates are consistent with test acceptance criteria.

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9 These tests should be consistent with the provisions of Regulatory Guide 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants” (Ref. 22).

10 These tests should be consistent with the provisions of RG 1.140, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, Washington D.C. (Ref. 23).

11 NUREG-1793, “Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design,” Supplements 1 and 2 (Ref. 24)

A-1.i. Primary and Secondary Containments

Appropriate tests should be conducted to demonstrate that primary and secondary containments will function as designed. The testing methods and acceptance criteria should give due consideration to all systems and components that must operate for the containments to function as designed. In certain designs, normally or intermittently operating systems may be required to shut down and isolate to achieve containment isolation. For example, GDC 41, "Containment Atmosphere Cleanup," in Appendix A of 10 CFR Part 50 requires, in part, the secondary containment ventilation system in BWRs to shut down and isolate the normal ventilation paths to permit the standby gas treatment system to perform its design function. Therefore, appropriate testing should be conducted to demonstrate the operability of all components, features, and systems required to operate for the primary or secondary containment to function properly. Testing should include expected system configuration for routine testing to verify that bypass flow paths are not introduced and flow pressurization functions are not degraded.

Due consideration also should be given to plant features such as heating, ventilation, and air conditioning systems required to maintain environmental conditions within design limits for components or equipment provided to effect containment isolation. Testing should be sufficient to demonstrate redundancy, electrical independence requirements for isolation valves (see footnote 3), and proper operation of features (including proper operation of devices upon loss or failure of motive power) provided for isolation valves and other devices. To the extent practical, the testing should demonstrate that isolation devices perform as required under simulated accident conditions.

The following list illustrates the systems, features, and performance demonstrations that should be included in the test program:

1. containment design overpressure structural tests[12] and vacuum tests for sub-atmospheric containments:
2. containment isolation valve functional and closure timing tests;
3. containment isolation valve leak rate tests[13] and in-leakage tests for sub-atmospheric containments;
4. containment penetration leakage tests[13];
5. containment airlock leak rate tests[13];
6. integrated containment leakage tests[13;]
7. main steam-line leakage sealing and feedwater leakage sealing systems (BWR);
8. primary and secondary containment isolation initiation logic tests;
9. containment purge system tests;
10. containment and containment annulus vacuum breaker tests (BWR);
11. containment supplementary leak collection and exhaust system tests;

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12 Overpressure structural tests should be conducted in accordance with Section III of the ASME BPV Code.

13 The requirements for such tests are given in Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

12. containment air purification and cleanup system tests;
13. containment inerting system tests;
14. standby gas treatment system tests;
15. containment penetration pressurization system tests;
16. containment ventilation system tests;
17. secondary containment system ventilation tests;
18. containment annulus and cleanup system tests, including demonstrating the ability to maintain design pressure control in all modes of operation;
19. bypass leakage tests on pressure suppression containments;
20. ice condenser containments (sufficient measurements should be made to ensure that gross bypass leakage paths are not present); and
21. containment penetration cooling system tests.

In general, the test sequence should proceed from the low-pressure test to the accident-pressure test, or sufficient time should be allowed between tests to ensure that out-gassing from concrete or components within the containment will not affect the test results.

A-1.j. Instrumentation and Control Systems

The nomenclature applied to instrumentation and control systems varies widely with different plant designs; however, the primary functions are similar for light-water-cooled reactors. The principal functions of instrumentation and control systems are to:

1. control the normal operation of the facility within design limits;
2. provide information and alarms in the control room to monitor the operation and status of the facility and permit corrective actions to be taken for off normal plant conditions;
3. establish that the facility is operating within design and license limits;
4. permit or support the correct operation of engineered safety features; and
5. monitor and record important parameters during and following postulated accidents.

Tests also should verify that instruments for RCS leakage and secondary leakage are capable of detecting the required leakage rates using concentrations of radioactive material expected to be present in the RCS during routine operations.

In the design of nuclear power plants, postulated accident assumptions are often explicitly or implicitly bounded by the design of instrumentation and control systems (e.g., pressurizer level or feedwater flow control). In such cases, operation of the instrumentation and controls over the design operating range should be performed, and the effects of limiting malfunctions or failures should be simulated to demonstrate the adequacy of design and installation and the validity of accident analysis assumptions.

Tests should be conducted, as appropriate, to verify redundancy and electrical independence (see footnote 3).

The following list illustrates instrumentation and control systems that should be included in the test program (some of these tests can be conducted in conjunction with the appropriate system level tests):

1. pressurizer pressure and level control systems including transient response for pressurizer in-surge and out-surge;

2. main, auxiliary, and emergency feedwater control systems;

3. secondary system steam pressure control system;

4. recirculation flow control system;

5. reactor coolant system leak detection systems;

6. loose parts monitoring system;

7. leak detection systems used to detect failures in the ECCS and containment; recirculating spray systems located outside containment;

8. automatic reactor power control system, integrated control system, and T-average control system;

9. pressure control systems used to maintain design differential pressures to prevent leakage across boundaries provided to contain fission products (for example, those used to pressurize spaces between containment isolation valves);

10. seismic instrumentation;

11. traversing in-core probe system;

12. failed fuel detection system;

13. in-core and ex-core neutron instrumentation;

14. instrumentation and controls that effect transfers of water supplies to auxiliary feedwater pumps, ECCS pumps, and containment spray pumps;

15. automatic dispatcher control systems;

16. hotwell level control system;

17. feedwater heater temperature, level, and bypass control systems;

18. auxiliary startup instrument tests (neutron response checks);

19. instrumentation and controls used for shutdown from outside the control room;

20. instrumentation used to detect external and internal flooding conditions that could result from such sources as fluid system piping failures;

21. reactor mode switch and associated functions;

22. instrumentation that can be used to track the course of postulated accidents (such as containment wide range pressure indicators, reactor vessel water level monitors, containment sump or pressure suppression level monitors, high range radiation detection devices, and humidity monitors);

23. post-accident hydrogen monitors and analyzers used in the combustible gas control system;

24. annunciators for reactor control and engineered safety features;

25. final calibration of digital interfaces;

26. component cooling water, chilled water and ultimate heat sink;

27. refueling water storage tank and suppression pool level and temperature;

28. fire water system;

29. BWR and PWR interlocks and permissives;

30. primary to secondary leakage detection system through steam generators (PWR only); and

31. process computers.

A-1.k. <u>Radiation Protection Systems</u>

Appropriate tests should be conducted to demonstrate the proper operation of the following types of systems and components used to monitor or measure radiation levels, provide for personnel protection, or control or limit the release of radioactivity:

1. Test process, criticality, effluent, and area radiation monitors.

2. Test personnel monitors and radiation survey instruments.

3. Test laboratory equipment used to analyze or measure radiation levels and radioactivity concentrations.

4. Test high-efficiency particulate air filters and charcoal absorbers (see footnote 9).

5. Test leakage detection system sensitivity and capability meets TS leakage detection guidance in Regulatory Guide 1.45 (Ref. 25). This may include leakage detection system sensitivity and capability to detect RCS leakage from Steam Generators within the guidance in Nuclear Energy Institute (NEI) 97-06,

"Steam Generator Program Guidelines," (Ref. 26) and in Electric Power Research Institute (EPRI) TR-1008219, "PWR Primary-to-Secondary Leak Guidelines," (Ref. 27) (e.g., radiation monitor detection sensitivity is 30 gallons per day or 1.25 gallons per hour).

6. Test radiation monitor computer system.

7. Test radiation data transmission to the emergency response data system.

For radiation monitoring equipment that is used to perform automatic control functions, the tests should confirm, using established instrumentation set-points, that upon detecting elevated levels of radioactivity, the system initiates the proper automatic control features in ensuring the timely closures of isolation valves or dampers. Depending upon design features, the logic sequence and interdependence of the actuation of automatic features should be tested as well when linked to radiation levels process streams and radioactive effluents. Such features may include monitoring deviations of in-plant dilution and exhaust flow rates in terminating releases or isolating process flows when deviations exceed preset limits. Other design features include monitoring the temperature of steam generator blowdown to protect resin beds from excessive temperatures; thereby, preventing a sudden loss in decontamination factors and releases of radioactivity above established limits and contamination of otherwise clean portions of plant systems.

Tests should be conducted, as appropriate, to verify redundancy and electrical independence (see footnote 3).

A-1.l. <u>Integrity of Systems Outside of Containment that Contain Radioactive Material for BWRs and PWRs</u>

In accordance with the requirement in §50.34(f)(2)(xxvi), applicants shall provide for a leakage control and detection program in the design of systems outside containment that could contain radioactive material following an accident. As part of the initial test program, applicants shall submit a leakage control program, a schedule for retesting the systems, and actions taken to minimize leakage from these systems. The tests should include leak rate test results and a discussion of actions to reduce leakage from systems outside containment that could contain radioactive fluids or gases during or following a serious transient or accident. The goal is to minimize potential exposure to workers and the public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. The systems to be included in the leakage testing program include the following:

1. residual heat removal
2. containment spray
3. high-pressure injection and recirculation
4. containment spray recirculation
5. primary coolant sampling
6. reactor core isolation cooling (RCIC) (BWR only)
7. makeup and letdown (e.g., chemical volume control system) (PWR only)
8. waste gas - Testing of gaseous systems should include helium leak detection or equivalent testing methods

A-1.m. Radioactive Waste Handling and Storage Systems

Appropriate tests should be conducted to demonstrate the functional operability and design flow rates of systems and components that are used to process, store, and release (or control the release of) liquid, gaseous, and solid radioactive wastes. This testing should demonstrate, to the extent practical, that the pumps, tanks, controls, valves, and other equipment (including automatic isolation and protective features and instrumentation and alarms) will operate and function in accordance with design.

Solidification system tests should include verification that no free liquids are present in packaged wastes. Tests should include normal operation of resin storage and transfer systems with actual or simulated media (e.g., resin, charcoal, etc).

As a prerequisite to testing, steps should be taken to confirm that the proper types and amounts of filtration and adsorption media are present in processing equipment. This should be done to help ensure that system performance characteristics, expressed as removal efficiencies, decontamination factors, and hold up times, conform to the design basis. The testing should be designed to demonstrate compliance with liquid and gaseous effluent concentration limits of Appendix B , “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,” to 10 CFR Part 20, “Standards for Protection against Radiation,” (Ref. 31) and design objectives of Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ‘As Low as Is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents,” to 10 CFR Part 50.

Testing or calculations should include, as appropriate, verification of tank volumes, capacities, holdup times, and proper operation and calibration of associated instrumentation. Testing should also confirm the operation of local and remote alarm functions, including radioactivity or radiation levels above established set-points and tank content levels. Radioactive samples, appropriately spiked media with known quantities of radioactivity or appropriate types of radioactive sources, should be used for the given type of radiation detection method to verify operability and/or proper calibration of radiation detectors and monitors. The following list illustrates the systems, components, and features for which the test program should demonstrate operability:

1. liquid radioactive waste handling systems;

2. gaseous radioactive waste handling systems;

3. power cycle off-gas system monitoring and controlling the presence of explosive gas mixtures (H_2/O_2) in gaseous waste management subsystems;

4. solid waste handling systems and resulting waste products complying with waste classification and characteristic requirements;

5. isolation features for steam generator blow down, for both the presence of radioactivity and thermal protection of demineralizer beds;

6. isolation features for condenser off-gas systems and diversion of process flow to appropriate subsystems;

7. isolation features for ventilation systems and diversion of exhaust flows to HEPA/charcoal filtration subsystems;

8. isolation features for liquid radioactive waste effluent systems and diversion of effluent flows to appropriate subsystems;

9. isolation features (process interlocks, backflow preventers, differential pressures, etc.) of waste processing subsystems, as equipped, in preventing the cross-contamination of nonradioactive systems and avoiding unmonitored and uncontrolled radioactive releases;

10. operability of plant process and effluent sampling systems for expected types of media;

11. testing of liquid and wet waste solidification subsystems in verifying that residual amounts of free liquid present in process packaged wastes conform with regulatory requirements and waste acceptance criteria; and

12. for waste processing system supplemented with mobile skid-mounted processing equipment, as equipped, testing should include the hydraulic integrity of connections carrying radioactive fluids between mobile processing equipment and permanently installed plant subsystems.

A-1.n. <u>Fuel Storage and Handling Systems</u>

Appropriate tests should be conducted to demonstrate that equipment and components used to handle or cool irradiated and non-irradiated fuel will operate in accordance with design. Tests should be conducted, as appropriate, to verify redundancy and electrical independence (see footnote 3). The following list illustrates the equipment and component tests that the program should include:

1. spent fuel pool cooling system tests, including testing of anti-siphon devices, high radiation alarms, and low-water-level alarms and spent fuel pool water makeup;

2. refueling equipment tests, including hand tools, power equipment, bridge and overhead cranes, and grapples;

3. operability and leak tests of sectionalizing devices and drains and leak tests of gaskets or bellows in the refueling canal and fuel storage pool;

4. static and dynamic load testing[14] of cranes, hoists, and associated lifting and rigging equipment, including the fuel cask handling crane;

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14 NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants,” issued May 1979 (Ref. 28), and NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36,” issued July 1980 (Ref. 29), should be used as guidance for tests on single-failure-proof overhead crane handling systems.

5. fuel transfer devices;

6. irradiated fuel pool or building ventilation system tests;

7. computerized automated fuel handling systems, programming, indexing, data base controls for fuel storage information and interlocks should be tested for proper operation; and

8. equipment in the fuel cavity, including reactor vessel seals, vessel seal leakage detection, refueling cavity fuel storage handling equipment including temporary fuel storage racks and weir gates should be tested for proper operation.

Refueling equipment testing should demonstrate the operability of protective interlocks and devices. Static testing of cranes, hoists, and associated lifting and rigging equipment should be at 125 percent of rated load, and full operational testing should be at 100 percent of rated load.

A-1.o. <u>Auxiliary and Miscellaneous Systems</u>

The licensee should conduct appropriate tests to demonstrate the operability of auxiliary and miscellaneous systems. Tests should be conducted, as appropriate, to verify redundancy and electrical independence (see footnote 3). The following list illustrates the types of systems and features for which performance should be demonstrated by testing:

1. service and raw water cooling ultimate heat sink systems;

2. closed-loop cooling water systems;

3. component cooling water and chilled water systems;

4. reactor coolant makeup system;

5. reactor coolant and secondary sampling systems;

6. chemistry control systems for the reactor coolant and secondary coolant systems;

7. fire protection systems (including demonstrations of proper manual and automatic operation of fire detection, alarm, suppression, and smoke control systems);

8. seal water systems;

9. vent and drain systems (for contaminated or potentially contaminated systems and areas), and drain and pumping systems serving essential areas (e.g., spaces housing diesel generators, essential electrical equipment, and essential pumps);

10. purification and cleanup systems for the reactor coolant system;

11. compressed gas system[15] supplying pneumatic equipment, components, or instrumentation that are required to function to support the normal operation of the facility or are essential for the operation of standby safety equipment or engineered safety features;

12. boron recovery system;

13. communication systems;

14. heating, cooling, and ventilation systems serving the following areas should meet the guidance in RG 1.140:

    a. spaces housing engineered safety features
    b. primary containment
    c. battery rooms
    d. diesel generator buildings
    e. auxiliary, reactor, turbine, and radioactive waste handling buildings
    f. control room habitability systems
    g. ultimate heat sink pump house
    h. fuel storage and handling area ventilation system

15. shield cooling systems;

16. cooling and heating systems for the refueling water storage tank (PWR) and suppression pool cooling and heating for the condensate storage tank (BWR);

17. equipment and controls for establishing and maintaining subatmospheric pressures in sub-atmospheric containments;

18. heat tracing for freeze protection ;

19. emergency lighting for safe egress, post accident vital areas, and emergency facilities;

20. cathodic protection system for corrosion control of underground and submerged metallic surfaces (i.e., buried carbon steel pipes and tanks);

21. flood protection systems;

22. systems for protecting the foundation of structures from ground water (e.g., ground water level control and ground water chemistry);

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15 Regulatory Guide 1.68.3, "Preoperational Testing of Instrument and Control Air Systems," (Ref. 30) provides detailed guidance on testing of instrument and control air systems. Compressed gas systems may also be relied upon for breathing air; therefore, this system should be tested for air load transients that can affect both plant safety and personnel safety.

23. non-safety related redundant power suppliers or alternate power connections (e.g., cable/plugs that allow temporary interconnection of power systems);

24. Appendix R alternate power supplies to equipment, etc.;

25. leakage detection from the spent fuel pool, transfer canal, refueling cavity, etc.; and

26. regulatory treatment of nonsafety system (RTNSS) components (e.g., power systems, charging pumps and valves, etc.).

Communication system tests should include demonstrations of the proper operation of evacuation and other alarms, the public address system within the plant, systems that may be used if the plant is required to be shut down from outside the control room, and communication systems required by the facility's emergency plan. This testing may include a check for frequency interferences from emergency communication devices used at multi-unit sites or other communication devices.

Control room habitability system testing should include, as appropriate, demonstrations of the proper operation of smoke and toxic chemical detection systems and ventilation shutdown devices, including leak-tightness of ducts and flow rates, proper direction of airflows, and proper control of space temperatures.

A-1.p. <u>Reactor Component Handling Systems</u>

Include the following activities:

1. Conduct dynamic and static load tests (see footnote 14) of cranes, hoists, and associated lifting and rigging equipment (e.g., slings and strong backs used during refueling or the preparation for refueling).

2. Conduct full operational testing of cranes, hoists, and associated lifting and rigging equipment at 100 percent of rated load,

3. Conduct static testing of cranes, hoists, and associated lifting and rigging equipment at 125 percent of rated load.

4. Demonstrate operability of protective devices and interlocks.

5. Demonstrate operability of safety devices on equipment.

6. Demonstrate clearance for safe movement of heavy loads through designated paths.

7. Demonstrate operability of reactor component handling equipment in the refueling cavity, the refueling transfer canal and the spent fuel pool.

A-2. Initial Fuel Loading and Pre-Critical Tests

Licensees should conduct initial fuel loading cautiously to preclude inadvertent criticality. Licensees should establish and follow specific safety measures, such as (1) ensuring that all applicable TS requirements and other prerequisites have been satisfied, (2) establishing requirements for continuous monitoring of the neutron flux throughout core loading so that all changes in the multiplication factor are observed, (3) establishing requirements for periodic data taking, and (4) independently verifying that the fuel and control components have been properly installed.

Predictions of core reactivity should be prepared in advance to aid in evaluating the measured responses to specified loading increments. Comparative data on neutron detector responses from previous loadings of essentially identical core designs may be used in lieu of these predictions. Licensees should establish criteria and requirements for actions to be taken if the measured results deviate from expected values. Shutdown margin verifications should be performed at appropriate loading intervals (for BWRs), including full core shutdown margin tests. In addition, licensees should establish that the required shutdown margin exists, without achieving criticality.

To provide further assurance of safe loading, licensees should establish requirements for the operability of plant systems and components, including reactivity control systems and other systems and components necessary to ensure the safety of plant personnel and the public in the event of errors or malfunctions. The initial core loading should be directly supervised by a senior licensed operator having no other concurrent duties, and the loading operation should be conducted in strict accordance with detailed approved procedures. Appendix C to this regulatory guide describes typical prerequisites, precautions, and details that should be included in the initial fuel loading and pre-critical check procedures.

After the core is fully loaded, sufficient tests and checks should be performed to ensure that the facility is in a final state of readiness to achieve initial criticality and perform low power tests. The following list illustrates the types of tests and verifications that should be conducted during or following initial fuel loading:

a. shutdown margin verification for partially and fully loaded core;

b. testing of the control rod withdrawal and insert speeds and sequencers, control rod position indication, protective interlocks, control functions, alarms, and scram timing, and friction tests of control rods after the core is fully loaded;

c. final functional testing of the reactor protection system to demonstrate proper trip points, logic, and operability of scram breakers and valves, as well as demonstration of the operability of manual scram functions;

d. final test of the reactor coolant system to verify that system leak rates are within specified limits;

e. measurements of the water quality and boron concentration (PWR) of the reactor coolant system;

f. reactor coolant system flow tests to establish that:

    1. vibration levels are acceptable;

    2. differential pressures across the fully loaded core and major components in the reactor coolant system are in accordance with design values; and

    3. piping reactions to transient conditions (e.g., pump starting and stopping) and flows are as predicted for all allowable combinations of pump operation.

g. final calibration of source range neutron flux measuring instrumentation, including verification of proper operation of associated alarms and protective functions of source- and intermediate range monitors; and

h. mechanical and electrical tests of in-core monitors (including traversing in-core monitors, if installed).

Scram time tests should be sufficient to provide reasonable assurance that the control rods will scram within the required time under plant conditions that bound those under which the control rods might be required to function to achieve plant shutdown. To the extent practical, testing should demonstrate control rod scram times at both hot zero power and cold temperature conditions, with flow and no flow conditions in the reactor coolant system as required to bound conditions under which scram might be required. Additionally, the methods used to perform the test should be same as the test methods used after refueling evolutions.

For each test condition, those control rods for which the scram times fall outside the two sigma limit of the scram time data for all control rods should be retested a sufficient number of times (e.g., three times) to reasonably ensure proper performance during subsequent plant operations. For facilities using more than one type of control element or control rod drive design, scram times should be compared with identical designs (e.g., two control rods attached to a single drive mechanism). Additionally, the proper operation of decelerating devices used to prevent mechanical damage to the control rods should be demonstrated during this testing.

Reactor coolant system flow tests should include loss-of-flow tests to measure flow coast down. Differential pressure measurements across the fully loaded core and major components need not be repeated for plants using calculation models and designs identical to prototype plants.

A-3. Initial Criticality

Licensees should conduct the initial approach to criticality in a deliberate and orderly manner using the same rod withdrawal sequences and patterns that will be used during subsequent startups. Neutron flux levels should be continuously monitored and periodically evaluated. A neutron count rate of at least 1/2 counts per second should register on the startup channels before startup begins, and the signal to noise ratio should be known to be greater than 2. All systems required for startup or protection of the plant, including the reactor protection system and emergency shutdown system, should be operable, and in a state of readiness. The control rod or poison removal sequence should be accomplished using detailed procedures approved by personnel or groups designated by the licensee. For reactors that will achieve initial criticality by boron dilution, control rods should be withdrawn before dilution begins. The control rod insertion limits defined in the TS should be observed and followed.

Criticality predictions for boron concentration (PWRs) and control rod positions should be provided, and criteria and actions to be taken if actual plant conditions deviate from predicted values should be established. The reactivity addition sequence should be prescribed, and the procedure should require a cautious approach to achieving criticality to prevent passing through criticality in a period shorter than approximately 30 seconds (<1 decade per minute).

A-4. Low-Power Testing

Following initial criticality, licensees should conduct appropriate low power tests (normally at less than 5-percent power) to (1) confirm the design and, to the extent practical, validate the analytical models, and verify the correctness or conservatism of assumptions used in the safety analyses for the facility, and (2) confirm the operability of plant systems and design features that could not be completely tested during the preoperational test phase because of the lack of an adequate heat source for the reactor coolant and main steam systems.

The following list illustrates the tests that should be conducted if not previously completed during preoperational hot functional testing (tests that are specific to one type of light water reactor are noted by BWR and PWR as appropriate):

a. Verify boron and moderator temperature reactivity coefficients over the temperature and boron concentration ranges in which the reactor may initially be taken critical (PWR).

b. Measure control rod and control rod bank reactivity worth to (1) ensure that they are in accordance with design predictions and (2) confirm by analysis that the rod insertion limits will be adequate to ensure a shutdown margin consistent with accident analysis assumptions throughout core life, with the greatest worth control rod stuck out of the core.

c. Verify adequate overlap of source- and intermediate range neutron instrumentation.

d. Verify proper operation of associated protective functions and alarms provides for plant protection in the low power range (if not previously performed).

e. Verify flux distribution for comparison with distribution assumptions or predictions to check for potential errors in the loading or enrichment of fuel or lumped poison elements, as well as miss-positioned or uncoupled control rods. The measurements may be performed at a higher power level, depending on the sensitivity of in-core flux instrumentation using neutron and gamma radiation monitor surveys.

f. Verify proper response of process and effluent radiation monitors. To the extent practical, responses by installed process and effluent radiation monitors should be verified by laboratory analyses of samples from the process and/or effluent systems.

g. Verify chemical and radiochemistry tests and measurements to demonstrate the design capability of chemical control systems and installed analysis and alarm systems to maintain water quality within limits in the reactor coolant and secondary coolant systems.

h. Verify neutron and gamma radiation surveys are performed.

i. Demonstrate operability of control rod withdrawal and insertion sequencers and control rod withdrawal inhibit or block functions over the reactor power level range during which such features must be operable (BWR).

j. Demonstrate the capability of the primary containment ventilation system to maintain the containment environment and important components in the containment within design limits with the reactor coolant system at rated temperature and with the minimum availability of ventilation system components for which the system is designed to operate.

k. Demonstrate the operability of steam-driven engineered safety features, plant auxiliaries, and power conversion equipment.

l. Demonstrate the operability, including stroke times, of main and branch steam-line valves and bypass valves used for protective isolation functions at rated temperature and pressure conditions.

m. Demonstrate the operability of the main steam-line isolation valve leakage control system (BWR, during hot standby conditions).

n. Demonstrate the operability of the control room computer system.

o. Conduct control rod scram time testing at rated temperature in the reactor coolant system (if not previously conducted).

p. Demonstrate the operability of pressurizer and main steam system relief valves at rated temperature.

q. Demonstrate the operability of reactor coolant system purification and cleanup systems including tests for flow rates, heat transfer capability, purification and cleanup capability and pressure drops.

r. Demonstrate the operability of residual or decay heat removal systems, including auxiliary or emergency feedwater, atmospheric steam dump valves (PWR), and turbine bypass valves and the RCIC system or isolation condenser system (BWR).

s. Conduct vibration measurements of reactor vessel internals (see footnote 1) and reactor coolant system components (if not previously conducted).

t. Perform natural circulation tests of the reactor coolant system to confirm that the design heat removal, boron mixing plant cool down/depressurization, and stable natural circulation conditions are maintained throughout the test or by comparison of the plant's reactor coolant system hydraulic data to a reference prototype plant of similar design and configuration (PWR).

u. Demonstrate the operability of major or principal plant control systems, as appropriate.

A-5. Power-Ascension Tests

Licensees should complete low power tests, as described in the FSAR, and evaluate and approve the low power test results before beginning the power-ascension tests. Power ascension tests should demonstrate that the facility operates in accordance with design during normal steady state conditions and, to the extent practical, during and following anticipated transients. The tests should demonstrate that measured responses are in accordance with predicted responses to validate the analytical models. The predicted responses should be developed using real or expected values of attributes such as beginning of life core reactivity coefficients, flow rates, pressures, temperatures, pump coast down characteristics, and response times of equipment, as well as the actual status of the plant (not those values or plant conditions assumed for conservative evaluations of postulated accidents).

Tests and acceptance criteria should be prescribed to demonstrate the ability of major or principal plant control systems to automatically control process variables within design limits. Such tests are expected to provide assurance that the facility's integrated dynamic response is in accordance with design for plant events such as reactor scram, turbine trip, reactor coolant pump trip, and loss of feedwater heaters or pumps. Testing should be sufficiently comprehensive to establish that the facility can operate in all operating modes for which it has been designed; however, tests should not be conducted, or operating modes or plant configurations established, if they have not been analyzed or if they fall outside the range of assumptions used in analyzing postulated accidents in the facility's FSAR.

Appropriate consideration should be given to testing at the extremes of possible operating modes for facility systems. Testing under simulated conditions of maximum and minimum equipment availability within systems should be accomplished if the facility is intended to be operated in these modes (e.g., testing with different reactor coolant pump configurations, single-loop reactor coolant system operation, operation with the minimum allowable number of pumps, heat exchangers, or control valves in the feedwater, condensate, circulating, and other cooling water systems).

The following list illustrates the types of performance demonstrations, measurements, and tests that should be included in the power ascension test phase. Parenthetical numbers following the listed activities indicate the approximate power levels to be used in conducting the tests. If no number follows a listed activity, the demonstration, measurement, or test should be performed at the lowest practical power level. Tests that are specific to one type of light water reactor are noted by BWR or PWR, as appropriate. Power ascension test activities may need to be adjusted because of new plant design features for nuclear power plants licensed under 10 CFR Part 52.

a. Determine that power reactivity coefficients (PWR) or power versus flow characteristics (BWR) are in accordance with design values (25 percent, 50 percent, 75 percent, and 100 percent).

b. Determine that steady state core performance is in accordance with design. Sufficient measurements and evaluations should be conducted to establish that flux distributions, local surface heat flux, linear heat rate, departure from nucleate boiling ratio, radial and axial power peaking factors, maximum average planar linear heat generation rate, minimum critical power ratio, quadrant power tilt, and other important parameters are in accordance with design values throughout the permissible range of power to flow conditions (25 percent, 50 percent, 75 percent, 100 percent).

c. Demonstrate that core limits will not be exceeded during or following an exchange of control rod patterns that will be permitted during operation. (The demonstration test should be conducted at the highest power level at which control rod pattern exchanges will be allowed during plant operation).

d. Demonstrate the capabilities of plant features (such as part length control rods) and procedures for controlling core xenon transients. Acceptance criteria for the test should account for expected changes in core performance throughout core life (75 percent to 85 percent) (PWR). The results of xenon oscillation tests performed at plants of essentially identical design can be used to substitute or supplement this testing.

e. Demonstrate that core thermal and nuclear parameters are in accordance with predictions with a single high-worth rod fully inserted and during and following return of the rod to its bank position (50 percent).

f. Demonstrate that control rod sequencers, control rod worth minimizers, and rod withdrawal block functions operate in accordance with design, if not previously demonstrated (25 percent) (BWR).

g. Check rod scram times from data recorded during scrams that occur during the startup test phase to determine that the scram times remain within allowable limits.

h. Demonstrate the capability and/or sensitivity, as appropriate for the facility design of incore and ex-core neutron flux instrumentation, to detect a control rod misalignment equal to or less than the TS limits (50 percent, 100 percent) (PWR).

i. Verify that plant performance is as expected for rod runback and partial scram.

j. Demonstrate that ECCS high pressure coolant injection (HPCI) systems (e.g., HPCI for BWR/2-4, high pressure core spray (HPCS) for BWR/5-6, high pressure core flooder (HPCF) for ABWR) can start under simulated accident conditions and inject into the reactor coolant system as designed. This test should be conducted at a power level in the 25 percent to 50 percent range for BWRs with steam- or electric driven pumps, if not previously conducted. See RG 1.79 and RG 1.79.1 for additional details.

k. Demonstrate the design capability of all systems and components provided to remove residual or decay heat from the reactor coolant system, including the turbine bypass system, atmospheric steam dump valves, residual heat removal system in steam condensing mode, RCIC system (BWR/3-6, ABWR), isolation condenser system (BWR/2, ESBWR), reactor water cleanup shutdown cooling system non-regenerative heat exchangers (ESBWR), and auxiliary feedwater system. Testing of the auxiliary or emergency feedwater system should include provisions to provide reasonable assurance that excessive flow instabilities (e.g., water hammer) will not occur during subsequent normal system startup and operation (before exceeding 25-percent power). See RG 1.79 and RG 1.79.1 for additional details.

l. Demonstrate that the reactor coolant system operates in accordance with design. Sufficient measurements and evaluations should be conducted with the plant at steady state conditions to establish that flow rates, reverse flows through idle loops or jet pumps, core flow, differential pressures across the core and major components in the reactor coolant system, vibration levels of reactor coolant system components, and other important parameters are in agreement with design values, if not previously demonstrated.

m. Obtain baseline data for the reactor coolant system loose parts monitoring system, if not previously obtained.

n. Calibrate instrumentation and demonstrate the proper response of reactor coolant leak detection systems, if not previously demonstrated and check the operation of computer programs used to calculate reactor coolant system leakage rates. This should include proper operation of reactor coolant system leakage rate monitoring equipment used to detect leakage into the secondary system through the steam generators (PWR only).

o. Conduct vibration monitoring of reactor internals during steady state and transient operation to establish that design limits are not exceeded (see footnote 1). BWR plants may conduct extensive inspections and tests of the steam dryer for potential adverse flow effects from acoustic resonance and hydraulic loading during power ascension. Where evaluation of plant data indicates that structural limits might be reached during power ascension, procedural instructions will require reduction in power to an acceptable level with reevaluation of the steam dryer loads to demonstrate acceptable structural capability prior to continuing power ascension. See RG 1.20 for specific guidance.

p. Verify the proper operation of failed fuel detection systems (25 percent, 100 percent).

q. Verify by review and evaluation of printouts and video displays that the control room or process computer is receiving correct inputs from process variables, and validate that performance calculations performed by the computer are correct (25 percent, 50 percent, 75 percent, 100 percent).

r. Calibrate, as necessary, and verify the performance of major or principal plant control systems including:

1. T-average controller
2. automatic reactor control system
3. boron addition systems (PWR)
4. integrated control system
5. pressurizer control system
6. reactor coolant flow control system
7. main, auxiliary, and emergency feedwater control systems
8. hotwell level control systems
9. steam pressure control systems
10. reactor coolant makeup and letdown control systems (25 percent, 50 percent, 75 percent, 100 percent)

s. If not previously accomplished, verify, as appropriate, the operability, response times, relieving capacities, set points, and reset pressures for:

1. pressurizer relief valves
2. main steam-line relief valves
3. atmospheric steam dump valves
4. turbine bypass valves
5. turbine stop, intercept, and control valves (25 percent)

t. During transient tests, verify the operability, set points, and reset pressures of relief valves (25 percent).

u. Verify the operability and response times of main and branch steam-line isolation valves. For PWRs, licensees may submit justification for conducting this test at low power and/or a description of design qualification tests for valves of the same size and design (25 percent).

v. Verify that the main steam and feedwater systems operate in accordance with design performance requirements. In addition, verify that acoustic resonance criteria are not reached in the main steam system as power level is increased (25 percent, 50 percent, 75 percent, and 100 percent).

w. Demonstrate adequate beginning of life performance margins for shielding and penetration cooling systems to provide assurance that they will be capable of maintaining temperatures of cooled components within design limits with the minimum design capability of cooling system components available (100 percent).

x. Demonstrate adequate beginning of life performance margins for auxiliary systems required to support the operation of engineered safety features or maintain the environment in spaces that house engineered safety features to provide assurance that the engineered safety features will be capable of performing their design functions over the range of design capability of operable components in these auxiliary systems (50 percent, 100 percent).

y. Calibrate, as required, and verify the proper operation of important instrumentation systems, including:

1. reactor coolant system flow
2. core flow, level, and temperature
3. in-core and ex-core neutron flux
4. instruments and systems used to calculate the thermal power level (heat balance) of the reactor (25 percent, 50 percent, 75 percent, 100 percent)

z. Demonstrate that process and effluent radiation monitoring systems or airborne activity monitoring system, are responding correctly by performing independent laboratory testing or other analyses.

aa. Demonstrate that chemical and radiochemical control systems function in accordance with design and sample to establish that reactor coolant system and secondary coolant system limits are not exceeded (25 percent, 50 percent, 75 percent, 100 percent).

bb. Conduct neutron and gamma radiation surveys to establish the adequacy of the airborne activity monitoring system, shielding surveys, access controls, and identification of high-radiation zones as defined in 10 CFR Part 20, Technical Specifications, RG 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants," (Ref. 32) and RG 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants" (Ref. 33) (50 percent, 100 percent).

cc. Demonstrate that gaseous and liquid radioactive waste processing, storage, and release systems operate in accordance with design.

dd. Demonstrate the capability to shut down and maintain the reactor in a hot standby condition from outside the control room, using the minimum shift crew, as well as the potential capability to place the reactor in a cold shutdown condition[16] (greater than or equal to a 10 percent generator load).

ee. Demonstrate that primary containment inerting and purge systems operate in accordance with design, if not previously demonstrated.

ff. Demonstrate or verify that important ventilation and air conditioning systems, including those for the primary containment and steam-line tunnel, continue to maintain their service areas within the design limits (50 percent, 100 percent).

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16 Regulatory Guide 1.68.2, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants," (Ref. 34) should be used as guidance for demonstration of this capability.

gg. If appropriate for the facility design, conduct tests to determine the operability of equipment provided for anticipated transient without scram, if not previously determined (25 percent).

hh. Demonstrate that the dynamic response of the plant to the design load swings for the facility, including step and ramp changes, is in accordance with design (25 percent, 50 percent, 75 percent, and 100 percent).

ii. Demonstrate that the dynamic response of the plant is in accordance with design for limiting reactor coolant pump trips and/or closure of reactor coolant system flow-control valves (BWR). The method for initiating the pump trip or control valve closure should result in the fastest credible coast down in flow for the system (100 percent).

jj. Demonstrate that the dynamic response of the plant is in accordance with design for a condition of loss of turbine generator coincident with loss of all sources of offsite power (i.e., station blackout for passive plants with no emergency diesel generators) (in the range of 10-20 percent power).

kk. Demonstrate that the dynamic response of the plant is in accordance with design for the loss or bypass of the feedwater heater(s) from a credible single failure or operator error that would result in the most severe case of feedwater temperature reduction (50 percent, 90 percent).

ll. Demonstrate that the dynamic response of the plant is in accordance with design requirements for turbine trip. This test may be combined with the dynamic response testing (item "mm" below) if a turbine trip is initiated directly by all remote-manual openings or automatic trips of the generator main breaker (i.e., a direct electrical signal, not a secondary effect such as a turbine over-speed) (100 percent).

mm. Demonstrate that the dynamic response of the plant is in accordance with design for the case of automatic closure of all main steam-line isolation valves. For PWRs, justification for conducting the test at a lower power level, while still demonstrating proper plant response to this transient, may be submitted for NRC staff review (100 percent).

nn. Demonstrate that the dynamic response of the plant is in accordance with design for the full load rejection. The method used to open the generator main breakers (by simulating an automatic or manual trip) should be selected, such that the turbine generator will be subjected to the maximum credible over-speed condition. The test should be initiated with the plant's electrical distribution system aligned for normal full-power operation (100 percent).

oo. Verify by observations and measurements, as appropriate, that piping and component movements, vibrations, and expansions are acceptable for (1) ASME Code Class 1, 2, and 3 systems, (2) other high energy piping systems inside Seismic Category I structures, (3) high energy portions of systems of which failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable level, and (4) Seismic Category I portions of moderate energy piping systems located outside containment. In addition to these activities, licensees will conduct detailed monitoring for flow-induced vibration of piping and components (including the steam dryer in BWR plants) in the reactor coolant, main steam, and main feedwater systems for potential adverse flow effects caused by acoustic resonance and hydraulic loading. For additional details, refer to RG 1.20 for additional details.

pp. Test the heat removal capacity of ultimate heat sink cooling chains including heat exchangers, cooling towers or other cooling sources (e.g., lake, river, and ocean).

qq. Verify through data collection that the design capacity of the ultimate heat sink will meet design acceptance criteria under normal plant conditions and design basis accident conditions.

rr. Verify safety-related bus voltages obtained from the voltage analyses (by using analytical techniques and assumptions) by comparing with that of actual measured safety-related bus voltages.

ss. Demonstrate the main generator is designed to accept a load rejection without a turbine trip and continue to supply plant loads without interruption (island mode plant operation).

A-6. First-of-a-Kind (FOAK) Testing

First-of-a-kind (FOAK) tests are defined as new, unique, or special tests for new design features in plants licensed under 10 CFR Part 50 or 10 CFR Part 52. If certain SSCs are new, unique, or special, principal design features that will be used in the facility, the ITP functional testing requirements necessary to verify FOAK test performance should be identified at an early date in the SDA, DC, ML, COL, or OL application process to account for the final design of the plant.

Some new plant designs licensed under 10 CFR Part 52 have passive plant design features and FOAK tests for systems that are safety-related or important to safety. For example, some new passive plant designs use large pyrotechnic-actuated valves with new designs in safety significant applications. Consequently, each new SDA, DC, ML, COL, or OL application for an advanced plant licensed under 10 CFR Part 50 or 10 CFR Part 52 should identify new FOAK tests in the given plant. This regulatory guide does not identify specific FOAK tests but new SDA, DC, ML, COL, or OL applicants may identify specific FOAK tests for new, unique or special plant design features that require additional design verification testing during the ITP to verify as-built test acceptance criteria are met.

A-7. Design Qualification Tests for Advanced Reactors

Design qualification tests for advanced reactors are used to demonstrate the performance of new safety features for nuclear reactor plants that differ significantly from evolutionary light water reactors or that use simplified, inherent, passive, or other innovative means to accomplish their safety functions. As discussed in the statements of consideration (SOC) for "Licenses, Certifications and Approvals for Nuclear power Plants – Final Rule," Federal Register Notice (FRN) Volume 72, No. 166, dated August 28, 2007, (Ref. 35), pages 49369 and 49370, states, in part, design qualification test requirements were included in 10 CFR Part 52 to demonstrate the performance of new safety features for nuclear power plants (advanced reactors) that differ significantly from evolutionary light water reactors or that use simplified, inherent, passive or other innovative means to accomplish their safety functions. Design qualification test requirements were included in 10 CFR Part 52 to ensure that new safety features will perform as predicted in the licensee's FSAR, to provide sufficient data to validate analytical codes, and to ensure that the effects of systems interactions are acceptable.

The design qualification testing requirements may be met with either separate effects or integral system tests, prototype tests, or a combination of tests, analyses, and operating experience. These requirements implement the Commission's policy on proof of performance testing for all advanced reactors with the goal of resolving all safety issues before authorizing construction.

Some prototype plant tests or FOAK tests may not be resolved until after fuel load and plant startup. The supplier of a DC may use integral system test facilities to demonstrate through scale model testing and simulation codes that certain as-built DC SSCs will meet their design and test acceptance criteria. If integral system tests and simulation codes cannot adequately demonstrate proof of performance testing, then the COL applicant may identify additional prototype plant tests or FOAK tests during the ITP to verify as-built test acceptance criteria are met.

Under 10 CFR 50.2, "Prototype plant" means a nuclear reactor that is used to test design features, such as the testing required under §50.43(e). The prototype plant is similar to FOAK or standard plant design in all features and size, but may include additional safety features to protect the public and the plant staff from the possible consequences of accidents during the testing period.

Under §50.43(e), applications for a DC, COL, ML, or OL that propose nuclear reactor designs that differ significantly from light-water-reactor designs that were licensed before 1997 or use simplified, inherent, passive, or other innovative means to accomplish their safety functions, will be approved only if:

> *(1)(i) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;*
>
> *(ii) Interdependent effects among the safety features of the design are acceptable, as demonstrated by analysis, appropriate test programs, experience, or a combination thereof; and*
>
> *(iii) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions;*
>
> *or*
>
> *(2) There has been acceptable testing of a prototype plant over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If a prototype plant is used to comply with the testing requirements, then the NRC may impose additional requirements on siting, safety features, or operational conditions for the prototype plant to protect the public and the plant staff from the possible consequences of accidents during the testing period.*

In §52.137(b), an application for approval of a SDA, which differs significantly from the light-water reactor designs of plants that have been licensed and in commercial operation before April 18, 1989, or uses simplified, inherent, passive, or other innovative means to accomplish its safety functions, must meet the requirements of §50.43(e).

Thus, the regulations in §50.43(e) may also apply to SDA, DC, ML and COL applications submitted under §52.47(a)(2)(iii), §52.47(c)(2), §52.79(a)(2)(iii), §52.79(a)(24), §52.79(a)(28) and §52.137(b).

In §52.47(a)(2), applicants for DC of a nuclear power reactor design must contain in the FSAR a description of the facility that presents the design bases and the limits on its operations and presents a safety analysis of the SSCs and of the facility as a whole, including the unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive material. While §52.79(a)(2)(iii) states, "the extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive material," §52.47(c)(2), states, in part "that a DC application that uses simplified, inherent, passive or other innovative means to accomplish their safety function, the DC applicant must meet the requirements in 10 CFR 50.43(e)." The regulations in §52.79(a)(28) require that a COL applicant describe plans for preoperational testing and initial operations. The NRC staff considers this to include testing to verify that unique, unusual or enhanced safety features systems will meet their design and test acceptance criteria.

In §52.47(a)(2), applicants for certification of a nuclear power reactor design must include, in the FSAR, a description of the facility including the design basis and limits on its operations. The applicants must also submit a safety analysis of the SSCs and facility as a whole, including unique, unusual, or enhanced safety features that have a significant bearing on the probability or consequences of the accidental release of radioactive material. Subsection 52.79(a)(2)(iii) requires that the COL FSAR also identify unique, unusual, or enhanced safety features to the extent that the DC is not already incorporated by reference. The requirements in 10 CFR 52.79(a)(24) state, in part, that if the COL application uses simplified, inherent, passive, or other innovative means to accomplish their safety function, the application must describe how the design meets the requirements in 10 CFR 50.43(e). The regulations in §52.79(a)(28) require a COL application to describe plans for preoperational testing and initial operations, including testing to verify that unique, unusual, or enhanced safety features systems will meet their design and test acceptance criteria.

The requirements in §52.79(a)(28) state that the FSAR is required to contain the plans for preoperational testing and initial operations of the new reactor. The requirements in §50.43(e) may apply to preoperational and initial operation tests, including prototype plant or FOAK design features tests in the new reactor. These regulations do not require the use of a prototype plant for design qualification testing. Rather, they provide that, if a prototype plant is used to qualify an advanced reactor design, then additional conditions may be required for the licensed prototype plant to compensate for any uncertainties with unproven safety features. Also, the prototype plant can be used for commercial operation. After prototype plant or FOAK tests are complete, these tests may not be needed on subsequent plants. The holder of an SDA, DC, COL, ML, or OL may use operating and testing experience to justify not performing the tests on subsequent plants or may choose to perform the test on subsequent plants if additional operating and testing experience is needed.

Based on the above, the SDA, DC, COL, ML or OL applications shall identify design qualification tests for new advanced reactors. These tests may include separate effects tests, integral systems tests, prototype plant tests, and FOAK tests or a combination of tests for new, unique, unusual or special SSCs that are used to protect public health and safety and plant staff from the consequences of accidents during the ITP test period. These tests will address components with new designs or applications, such as pyrotechnic-actuated squib valves or nozzle check valves, from currently licensed nuclear power plants. These tests may include preoperational, startup, low power, and power ascension tests for new safety-related SSCs that are active or passive design features in new advanced reactors.

APPENDIX B

INSPECTION BY
THE OFFICE OF NUCLEAR REACTOR REGULATION,
THE OFFICE OF NEW REACTORS, AND REGIONAL OFFICES

For new plant OL applicants under the requirements in 10 CFR Part 50, the NRC will implement inspection of the ITP beginning before preoperational testing and continuing through initial criticality, startup, low power and power ascension testing. The NRC inspectors will verify that the OL meets the requirements in §50.34(b)(6)(iii) and §50.43(e). NRC inspectors will follow Inspection Manual Chapter (IMC) 2513, Appendix A, "Light Water Reactor Preoperational Testing Phase," (Ref. 36), Appendix B, "Light Water Reactor Operational Preparedness Phase," (Ref. 37), IMC 2514, "Light Water Reactor Inspection Program – Startup Testing Phase," (Ref. 38), and IMC 2514, Appendix A, "Startup Test Program Inspection Procedures," (Ref. 39)

The NRC inspectors will inspect the licensee's ITP (i.e., preoperational, fuel loading, initial criticality, lower power and power ascension tests), as described in the FSAR, to determine if the ITP is adequately implemented and whether test results demonstrate that the plant, procedures, and personnel are ready for safe operation. The inspections will focus on the manner in which the licensee fulfilled FSAR commitments, NRC license conditions and regulations related to the ITP, and implemented ITP procedures used to adequately demonstrate completion of ITP test results.

For new plants licensed in accordance with the requirements of 10 CFR Part 52 the NRC will implement construction inspection programs and conduct inspections of ITPs beginning before preoperational testing and continuing through fuel loading, initial criticality, startup, low power and power ascension testing. The NRC inspectors will verify that the COL meets the requirements in §52.79(a)(28) and §50.43(e).

The ITAACs incorporated into 10 CFR Part 52 COL are the specific preapproved set of verifications that licensees must meet before fuel load. The NRC staff's inspections are intended to determine, on a risk-informed sample basis, whether the licensee's ITPs, as described in the FSAR, are adequately implemented, and whether the test results demonstrate that the plant, procedures, and personnel are ready for safe operation. Toward that end, the inspections focus on the manner in which the licensee has fulfilled its commitments to ensure that adequate programs have been developed and carried out, as exemplified by the methods used to establish procedures and the results those methods have produced.

For plants licensed under 10 CFR Part 52, NRC inspectors will inspect the overall preoperational test program and ITAAC as part of the construction program inspections included in IMC 2503 "Construction Inspection Program: Inspections of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Related Work," (Ref. 40) and IMC 2504, "Construction Inspection Program – Inspection of Construction and Operational Programs," (Ref. 41). In most cases, licensees have testing requirements in ITAAC that may also be associated with the preoperational tests completed as part of the ITP. NRC inspectors will sample preoperational tests completed as part of the ITAAC through a risk-informed sampling process. The inspectors will verify that preoperational tests satisfy both the ITP and the ITAAC.

The licensee should have copies of its test procedures available for examination by the NRC's regional office personnel approximately 60 days before the scheduled preoperational tests, and copies of procedures for fuel loading, initial criticality, initial startup testing, and supporting activities not less than 60 days before the scheduled fuel loading date. Examination by NRC personnel does not constitute approval of the procedures. The possession of such procedures by NRC personnel should not impede the revision, review, and refinement of the procedures by the licensee.

The inspections by NRC personnel generally include the following activities:

1. Examine the methods being used to prepare, review, and approve procedures; control test performance; record, evaluate, review, approve, and retain test data and results; and identify and correct deficiencies noted in systems and procedures. For the most part, this examination will be carried out before the start of the formal test program and is intended to determine whether the licensee has established a set of administrative procedures that will ensure that the programs are carried out in accordance with the methods described in the final safety analysis report.

2. Examine selected test procedures to ascertain whether the tests are designed to satisfy their objectives, whether test procedures contain appropriate acceptance criteria, and whether the procedures require documentation of sufficient information to permit adequate evaluation of the test results. This examination will also determine, on a selective basis, whether changes to approved test procedures have been reviewed and authorized.

3. Examine the fuel loading and startup procedures to ascertain whether prerequisites, prescribed operations, and limitations are appropriately included to control the operation, and whether the licensee has implemented administrative controls identified in Item (1) above.

4. Verify that the licensee has evaluated the test results and either concluded that those results are satisfactory and meet the acceptance criteria or initiated corrective action.

5. Verify that the licensee has reviewed the results of the fuel loading and initial operations.

6. Conduct an independent examination of the results of selected tests important to safety. This examination is primarily intended as an independent, selective audit to determine whether the licensee is appropriately documenting and evaluating information, and whether the resulting technical conclusions are valid.

7. Witness parts of preoperational, fuel loading, startup, low power, and power ascession tests to determine whether they are being conducted in the manner described in the licensee's administrative and test procedures, and whether they are being performed in a technically competent manner that satisfies the NRC regulations.

8. Verify that the licensee completes FOAK tests in the licensee's ITP, NRC license conditions and requirements in the FSAR.

9. Verify that test results demonstrate that acceptance criteria are met.

10. Verify that the licensee completes §50.79(a)(28) preoperational tests consistent with the ITAAC requirements in 10 CFR 52.103.

APPENDIX C

PREPARATION OF PROCEDURES

This appendix provides guidance on the preparation and content of procedures for preoperational tests, fuel loading and pre-critical tests, startup to critical, low power tests, and power ascension tests.

C-1. Preoperational Test Procedures

a. <u>Prerequisites</u>

Each test of the operation of a system normally requires that certain other activities be performed first (e.g., completion of construction, construction and/or preliminary tests, inspections, and certain other preoperational tests or operations). The preoperational testing procedures should include, as appropriate, these specific prerequisites, as illustrated by the following typical examples:

1. Confirm that construction activities associated with the system have been completed and documented. Field inspections should have been conducted to ensure that the equipment is ready for operation, including inspection for proper fabrication and cleanness, checkout of wiring continuity and electrical protective devices, adjustment of settings on torque limiting devices and calibration of instruments, verification that all instrument loops are operable and respond within required response times, and adjustment and settings of temperature controllers and limit switches.

2. Confirm that test equipment is operable and properly calibrated.

3. Conduct tests and surveillances of individual components or subsystems to demonstrate that they meet their functional requirements. The following items should typically be considered for common types of equipment:

    a. Valves[1]
    - leakage
    - opening and closing times
    - valve stroke
    - valve flow direction
    - valve seat orientation
    - position indication that confirms valve obturator position is accurately specified by control room lights
    - torque- and travel limiting settings
    - operability against pressure

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1 American Society of Mechanical Engineers (ASME) *Operation and Maintenance of Nuclear Power Plants* (Ref. 7) provides requirements for in-service testing of pumps, valves and dynamic restraints in nuclear power plants. The applicant should examine these requirements for applicability to its preoperational test programs.

- diagnostic testing of power-operated valves with measurements of valve operating requirements and actuator output under static conditions and against system differential pressure and flow to demonstrate design-basis capability
- pyrotechnic-actuated valve actuation control circuitry operational readiness, and sample testing of pyrotechnic charges to demonstrate design-basis capability
- check valve performance capability in both the forward and reverse flow directions
- safety and relief valve performance in accordance with design requirements
- examination of external condition and internal parts for structural integrity and presence of foreign material and fluids, and to confirm operational readiness (for example, surveillance requirements for pyrotechnic-actuated valves and nozzle check valves)

b. Pumps[1]
- direction of rotation
- vibration
- motor load versus time
- seal or gland leakage
- seal cooling
- flow and pressure characteristics that demonstrate design-basis capability to deliver flow requirements at design pressure
- lubrication
- acceleration and coast down

c. Dynamic Restraints (Snubbers)[1]
- visual examination
- load rating
- location
- orientation
- position setting
- configuration
- swing clearance
- fluid level

d. Motors and Generators
- direction of rotation

- vibration
- thermal overload protection, margins between set points, and full load running amps
- lubrication, cooling, heating, ventilation, fueling, starting , and exhaust
- Megger or hi-pot tests
- supply voltage and frequency
- phase to phase checks
- neutral current
- acceleration under load
- temperature rise
- phase currents
- load acceptance capability versus both time and load (generators)
- divisional redundancy and electrical independence
- alarms, interlocks, and control functions

e. Piping and Vessels
- hydrostatic test
- leak-tightness
- cleaning, flushing, and layup[2]
- clearance of obstructions
- support adjustments
- proper gasketing
- bolt torque
- insulation
- filling and venting

f. Electrical and Instrumentation and Control
- verification that sensing lines are clear for process sensors and that instrument root valves are open
- voltage (i.e., operating ranges)
- frequency
- current

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2 Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," (Ref. 42) should be used as guidance.

- circuit breaker operation
- power source identification
- bus transfer schemes
- trip settings
- operation of interlocks, prohibits, and permissives
- operation of standby power system logic and its loading sequence
- calibration of relays (degraded grid voltage and loss of voltage)
- control transformer settings
- temperature effects
- range checks
- response times

b. <u>Test Objectives</u>

Objectives of the test should be clearly stated. Many systems tests will be intended to demonstrate that each of several initiation events will produce one or more expected responses. These initiating events and the corresponding responses should be identified.

c. <u>Special Precautions</u>

The test procedure should highlight and clearly describe any and all special precautions needed to ensure a reliable test or the safety of personnel or equipment.

d. <u>Initial System Conditions</u>

Where appropriate, instructions should be given pertaining to the system configuration, components that should or should not be operating, and other pertinent conditions that might affect the operation of the given system.

e. <u>Environmental Conditions</u>

Most tests will be run at ambient conditions; however, procedures should include provisions to test the equipment under environmental conditions as close as practical to those the equipment will experience in both normal and accident situations.

f. <u>Acceptance Criteria</u>

The test procedure should clearly identify the criteria against which the success or failure of the test will be judged and should account for measurement errors and uncertainties. In some cases, these will be qualitative criteria. Where applicable, quantitative values, with appropriate tolerances, should be used as acceptance criteria.

g. Data Collection

The test procedure should prescribe the data to be collected and the form in which the data are to be recorded. All entries should be permanent. The administrative controls should include an acceptable method for correcting an entry.

h. Detailed Procedures

Detailed step by step procedures should be provided for each test. To the extent practical, the test procedures should use approved normal plant operating procedures.

Each procedure should require necessary nonstandard arrangements to be restored to their normal status after the test is completed. Control measures such as jumper logs and check off lists should be specified. Nonstandard bypasses, valve configurations, and instrument settings should be identified and highlighted for return to normal. Nonstandard arrangements should be carefully examined to ensure that temporary arrangements do not invalidate the test by interfering with proper testing of the as built system.

i. Documentation of Test Results

Records should identify each observer and/or data recorder participating in the test, as well as the type of observation, identifying numbers of test or measuring equipment, results, acceptability, and action taken to correct any deficiencies. Administrative procedures should specify the retention period of test result summaries and should require permanent retention of documented summaries and evaluations.

C-2. Fuel Loading

This section provides guidance on typical information to be included in the detailed fuel loading procedure.

a. Prerequisites for Fuel Loading

1. Specify the composition, duties, and emergency procedure responsibilities of the fuel handling crew including plans for completion of fuel loading milestones.

2. Test the radiation monitors, nuclear instrumentation, manual initiation, and other devices and verify that they are operable to actuate the building evacuation alarm and ventilation control.

3. Specify the status of all systems required for fuel loading.

4. Conduct inspections of fuel, control rods, and poison curtains.

5. Ensure that nuclear instruments are calibrated, operable, and properly located (source fuel detector geometry). One operating channel should have audible indication or annunciation in the control room.

6. Require a response check of nuclear instruments to a neutron source within "*N*" hours prior to loading (or resumption of loading, if delayed for "*N*" hours or

more), where "*N*" is consistent with the technical specification surveillance frequency for source range nuclear instruments in the refueling mode (typically 8 or 12 hours).

7. Specify and establish the status of containment.

8. Specify the status of the reactor vessel. Components should be either in place or out of the vessel, as specified, to make it ready to receive fuel.

9. Establish the vessel water level and prescribe the minimum level for fuel loading and unloading.

10. Specify and establish coolant circulation for borated reactors and take precautions (such as valve and pump lockouts) to prevent deboration.

11. Ensure that the emergency boron addition system (or other negative reactivity insertion system) is operable.

12. Check the fuel handling equipment and perform dry runs.

13. Prescribe and verify the status of protection systems, interlocks, mode switch, alarms, and radiation protection equipment. For reactors that have operable control rods during fuel loading, the high flux trip points should be set for a relatively low power level (normally not greater than 1 percent of full power).

14. Establish water quality and identify limits.

15. Establish and verify the minimum fuel loading boron concentration.

b. **<u>Procedure Details</u>**

The procedure should include instructions or information for the following areas:

1. loading sequence and pattern for fuel, control rods, poison curtains, and other components, with guidance on fuel addition increments (in general, the procedure should require constituting the core so that the reactivity worth of added individual fuel elements becomes less as the core is assembled);

2. maintenance of a display for indicating the status of the core and fuel pool, as well as appropriate records of core loading;

3. proper seating and orientation of fuel and components (the procedure should specify a visual check of each assembly in each core position);

4. functional testing of each control rod immediately following fuel loading (BWR);

5. nuclear instrumentation and neutron source requirements for monitoring subcritical multiplication, including source or detector relocation and normalization of the count rate after relocation (normally, a minimum of three

source range monitors on a BWR and two on a PWR should be operable whenever operations are performed that could affect core reactivity);

6. flux monitoring, including counting times and frequencies and rules for plotting inverse multiplication and interpreting plots (the counting period for count rates should be specified, and an inverse multiplication plot should be maintained);

7. the expected subcritical multiplication behavior;

8. determination of adherence to the minimum shutdown margin and rod worth tests in un-borated reactors and the frequency of determination (the minimum shutdown margin should be proved periodically during loading and at the completion of loading, and shutdown margin verifications should not involve a planned approach to criticality using nonstandard rod patterns or with operational interlocks bypassed);

9. determination of the boron concentration in borated reactors and the frequency of determination (the frequency of determination should be commensurate with the worst possible dilution capability, as determined by consideration of piping systems that attach to the reactor coolant system);

10. actions (especially those pertaining to flux monitoring) for periods when fuel loading is interrupted;

11. maintenance of continuous voice communication between the control room and loading station;

12. the minimum crew required to load fuel (the procedure should require the presence of at least two persons at any location where fuel handling is taking place, and a senior reactor operator with no other concurrent duties should be in charge);

13. crew work time (if personnel are scheduled for consecutive daily duty, they should not normally be expected to work more than 12 hours out of each 24);

    and

14. approvals required for changing the procedure.

c. <u>Limitations and Actions</u>

1. Establish criteria for stopping fuel loading. Some circumstances that might warrant this are unexpected subcritical multiplication behavior, loss of communications between the control room and fuel loading station, an inoperable source range detector, and inoperability of the emergency boration system.

2. Establish criteria for emergency boron injection.

3. Establish criteria for containment evacuation.

4. Outline actions to be followed in the event of fuel damage.

5. List actions to be followed or approvals to be obtained before routine loading may resume after one of the above limitations has been reached or invoked.

C-3. Initial Criticality Procedures

This section provides some specific guidance for the detailed procedure for operations associated with bringing the reactor critical for the first time. The guidance provided in Section 1, "Preoperational Test Procedures," of this appendix is also considered applicable. The initial criticality procedure should include steps to ensure that the startup will proceed in a deliberate and orderly manner, changes in reactivity will be continuously monitored, and inverse multiplication plots will be maintained and interpreted. A critical rod position (boron concentration) should be predicted so that any anomalies may be noted and evaluated. All systems needed for startup should be aligned and in proper operation. The emergency liquid poison system should be operable and in readiness. Technical specification requirements must be met.

Nuclear instruments should be calibrated. A neutron count rate (of at least 1/2 counts per second) should register on startup channels before the startup begins, and the signal to noise ratio should be known to be greater than 2. A conservative startup rate limit (no shorter than approximately a 30-second period) should be established. High flux scram trips should be set at their lowest value (approximately 5 percent to 20 percent).

Containment closure procedures should be implemented before changing plant modes that allow the reactor to start the approach to critical. The radiation monitoring program should be implemented to include operation of radiation barriers, airborne radiation monitors, air sampling, and baseline surveys before pulling control rods for the approach to critical. In accordance with TS requirements, all containment penetrations, the equipment hatch, and containment airlocks should be closed.

C-4. Low-Power and Power-Ascension Procedures

This section provides guidance for planning and preparing procedures for use in conducting the initial ascension to rated power. The guidance provided in Section 1, "Preoperational Test Procedures," of this appendix is also considered applicable. The program should be planned to increase power in discrete steps. Major testing should be performed at power levels of approximately 25 percent, 50 percent, 75 percent, and 100 percent.

If tests intended to verify that movements and expansion of equipment are in accordance with design are not conducted during hot functional tests and must be delayed until generation of nuclear heat, the first power level for conducting such tests should be as low as practical (approximately 5 percent).

Individual test procedures should include instructions and precautions for establishing the special conditions necessary for conducting tests. The individual procedures should highlight these special conditions and specifically provide for restoration to normal following the test. The overall or governing power-ascension test plan should typically require the following operations to be performed at appropriate steps in the power ascension test phase:

1. Conduct any tests that are scheduled at the test condition or power plateau.

2. Examine the radial flux for symmetry and verify that the axial flux is within expected values.

3. Determine reactor power by heat balance, calibrate nuclear instruments accordingly, and determine the existence of adequate instrumentation overlap between the intermediate- and power range detectors.

4. Reset high flux trips, just before ascending to the next level, to a value no greater than 20 percent beyond the power of the next level unless TS limits are more restrictive.

5. Perform general surveys of plant systems and equipment to determine that they are operating within expected values.

6. Check for unexpected radioactivity in process systems and effluents.

7. Perform reactor coolant leak checks and ESF system leakage checks are within FSAR Chapter 15 accident analysis limits.

8. Perform primary to secondary leakage detection testing through steam generators (PWR only).

9. Verify that acoustic resonance criteria are not reached in the main steam system.

10. Monitor potential adverse flow effects from acoustic resonance and hydraulic loading on reactor internals (including the steam dryer for BWR plants) and flow-induced vibration of piping and components in the reactor coolant, main steam, and main feedwater systems with procedural instructions to reduce power to an acceptable level where evaluation of plant data indicates that structural limits for piping or components might be reached during power ascension. Structural capability of piping and components will be demonstrated prior to continuing power ascension.

11. Review the completed testing program at each plateau; perform preliminary evaluations, including extrapolation of minimum departure from nucleate boiling ratio and maximum linear heat rate values to the high flux trip set point for the next power level; and obtain the required management approvals before ascending to the next power level or test condition.