U.S. NUCLEAR REGULATORY COMMISSION **March 2007**
Revision 3

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.68

(Draft was issued as DG-1166, dated November 2006)

INITIAL TEST PROGRAMS FOR WATER-COOLED NUCLEAR POWER PLANTS

A. INTRODUCTION

In Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50), "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 2), 10 CFR 50.34, "Contents of Applications; Technical Information," and 10 CFR 52.79, "Contents of Application, Technical Information in FSAR," require, in part, that an applicant for a license to operate a production or utilization facility provide a safety analysis report (SAR) that includes the principal design criteria for the proposed facility. The introduction to Appendix A, "General Design Criteria 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 requires 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. Since all functions designated in the general design criteria (GDC) are important to safety, all SSCs required to perform these functions need to be tested to ensure that they will perform properly. These functions, as noted throughout the specific GDC, 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 GDC and this guide recognize and provide for successive levels of plant features for achieving the safety of the facility. This is to provide for 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 and high tolerance for system malfunctions and transients; (2) ensure that, in the event of errors, malfunctions, and off-normal conditions, the reactor protection systems and other design features will arrest 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.

This revision incorporates information from 10 CFR Part 52, including initial test program (ITP) information from design certification (DC), manufacturing license (ML), and combined license (COL) applications submitted pursuant to Subparts A – C and F of 10 CFR Part 52. In particular, the DC, ML, and COL applications submitted for new reactors (pursuant to the provisions of 10 CFR Part 52) must include inspections, tests, analyses, and acceptance criteria (ITAAC), which the staff of the U.S. Nuclear Regulatory Commission (NRC) reviews in conjunction with those applications. (If a COL application references a DC or ML, only the site-specific ITAAC are reviewed for the COL application.)

The requirements regarding ITAAC for DC, ML, and COL applications are set forth in 10 CFR 52.47, 52.80, and 52.158. These include the ITAAC overlap with ITP preoperational tests.

Some preoperational tests completed as part of the ITP cover certain ITAAC completed prior to fuel load. For example, testing performed to demonstrate that safety-related SSCs will perform satisfactorily in service must be conducted under a program that satisfies Criterion XI, "Test Control," of Appendix B to 10 CFR Part 50, and may also satisfy testing required by the ITAAC process. The scope of the ITP, however, is not limited solely to safety-related SSCs. Consequently, this guide specifies the scope of plant SSCs to be tested to satisfy the requirements of GDC 1, "Quality Standards and Records" (as specified in Appendix A to 10 CFR Part 50), as well as the quality assurance criteria set forth in Appendix B to 10 CFR Part 50.

While all SSCs important to safety are required to be tested, all of them need not be tested to the same stringent requirements. Specifically, GDC 1 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. A graded approach is also inherent in the testing requirements of Criterion XI of Appendix B to 10 CFR Part 50. 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, test records retention, etc.) contain provisions for the application of such administrative controls in a manner commensurate with the safety-significance of the SSCs within its scope.

Section 50.34 of 10 CFR Part 50 and Section 52.79 of 10 CFR Part 52 for COLs also require, in part, that the applicant include plans for preoperational testing and initial operations in the final safety analysis report (FSAR). Chapter 14 of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (Ref. 3), and Section C.I.14 of Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)" (Ref. 4), provide guidance regarding the ITP-related information to be included in both the preliminary safety analysis report (PSAR) and the FSAR to enable the NRC staff to perform its safety evaluations for construction permits, operating licenses, and combined licenses.

This revision of Regulatory Guide 1.68 describes the general scope and depth that the NRC staff considers acceptable for ITPs for light-water-cooled nuclear power plants. Appendix A to this guide provides a representative listing of the plant SSCs and the design features and performance capabilities that should be demonstrated during the ITP. No particular significance should be attached to the order in which the tests are listed; however, in general, those listed in Section 1, "Preoperational Testing," of Appendix A should precede those listed in Section 2, "Initial Fuel Loading and Precritical Tests," and so on. Appendix B to this guide provides information regarding ITP-related inspections, which will be performed by the NRC, including the appropriate regional offices. Finally, Appendix C to this guide provides guidance regarding the preparation and content of procedures for preoperational tests, fuel loading and precritical tests, startup-to-critical low-power tests, and power-ascension tests.

The COL applicant should describe 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 they can only be performed after fuel load. For example, testing of the main steam isolation valves at high flow conditions, testing involving 100% 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 control document (DCD) or Section 14.2 of the COL FSAR.

This regulatory guide contains information collections that are covered by the requirements of 10 CFR Parts50 and 52, which the Office of Management and Budget (OMB) has 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

The applicant for a construction permit or operating license under 10 CFR Part 50, or combined license 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 to (1) provide additional assurance that the facility has been adequately designed and, to the extent practical, to validate the analytical models and verify the correctness or conservatism of assumptions used to predict plant responses to anticipated transients and postulated accidents and (2) provide assurance that construction and installation of equipment in the facility have been accomplished in accordance with design. Other key objectives are to familiarize the plant's operating and technical staff with the operation of the facility and to verify by trial use, to the extent practical, that the facility operating procedures and emergency procedures are adequate. Initial test programs 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 mentioned in the introduction, the test program 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, those designated as Seismic Category I by Regulatory Guide 1.29, "Seismic Design Classification" (Ref. 5) are considered more important to safety than some other SSCs that are 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, a graded approach to testing should be implemented 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 should also 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 construction permit or COL application. Because of the complexity of these tests and the significant amount of manpower 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 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 manpower 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 multi-unit 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 forces 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 prior to preoperational and/or startup tests of individual systems or components.

(5) Establish early plans for using available information regarding 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 complete testing.

If first-of-a-kind (FOAK), that is, new, unique, or special, principal design features will be used in the facility, the in-plant functional testing requirements necessary to verify their performance should be identified at an early date to permit these 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 or COL applicant for an advanced plant should identify all new FOAK tests in the given plant. Section 6 of Appendix A to this regulatory guide presents examples of FOAK tests. For DC and COL applicants, the NRC will verify that all FOAK tests proposed by the applicant meet the ITAAC and ITP testing requirements. Future COL applicants may propose other FOAK tests not specifically identified in this regulatory guide.

As previously noted, the ITP consists of preoperational and initial startup tests. “Preoperational testing,” as used in this guide, consists of those tests conducted following completion of construction and construction-related inspections and tests, but prior to 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 guide, consists of those test activities that are scheduled to be performed during and following fuel loading. These activities include fuel loading, precritical tests, initial criticality, low-power tests, and power-ascension tests that confirm the design bases and demonstrate, to the extent practical, that the plant will operate in accordance with design and is capable of responding as designed to anticipated transients and postulated accidents as specified in the SAR.

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.

The NRC staff’s safety evaluations of ITPs are based on information provided in the PSAR and FSAR for construction permits and operating licenses issued 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 only in the FSAR. The staff uses this information to support decisions to issue a construction permit, operating license, or combined license. In addition, the NRC uses the information provided in SARs as a basis for the inspection activities associated with ITPs. The satisfactory performance of approved test programs provides confirmation that adequate margins of safety exist, such that there is no undue risk to the health and safety of the public as a result of facility operation.

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 period of time may indicate inadequacies in the applicant'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 radioactive buildup, 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.

Appendix A references existing regulatory guides that are applicable to ITPs. The referenced guides provide detailed guidance for particular tests.

C. REGULATORY POSITION

1. Criteria for Selection of Plant Features To Be Tested

Pursuant to the 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:

(a) will be used for shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period

(b) will be used for shutdown and cooldown 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

(c) will be used to establish conformance with safety limits or limiting conditions for operation that will be included in the facility's technical specifications

(d) are classified as engineered safety features or will be relied on to support or ensure the operation of engineered safety features within design limits

(e) are assumed to function or for which credit is taken in the accident analysis of the facility, as described in the FSAR

(f) will be used to process, store, control, or limit the release of radioactive materials

Appendix A to this guide provides a representative listing of systems to be tested, as well as performance capabilities important to safety, as defined by Appendix A to 10 CFR Part 50, that should be demonstrated for light-water-cooled nuclear power plants. However, applicants should also conduct in-plant testing to verify the adequacy of 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 important SSCs, such as those considered engineered safety features.

2. Prerequisites for Testing

The construction or installation of SSCs should be essentially completed (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 should also be completed before beginning preoperational tests.

Tests designated in the FSAR as preoperational tests should be completed, and the results of such tests should be evaluated and approved by the applicant prior to issuance of the Part 50 operating license. In accordance with 10 CFR 52.103, "Operation Under a Combined License," the COL holder must fulfill the acceptance criteria identified in the license associated with initial plant testing before the initial fuel load. The overall test program should also include surveillance tests necessary to demonstrate proper operation of interlocks, setpoints, and other protective features, systems, and equipment required by the technical specifications. 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 following 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 should also 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 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.

4. Procedures

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 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, should also 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 prior to their intended use.

Before fuel loading commences, results of completed preoperational tests should be evaluated by personnel or groups designated by the applicant. Appropriate remedial actions, including retesting, should be taken if the acceptance criteria are not satisfied.

5. Schedule

Sufficient time should be scheduled to perform orderly and comprehensive testing. Previous applicant's schedules for conducting the preoperational and initial startup phases have typically allowed a minimum time of approximately 9 months and 3 months, respectively. Significantly shorter time periods should be justified.

6. Participation of Plant Operating and Technical Staff

The applicant'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 Operating and Emergency Procedures

Plant operating and emergency procedures should, to the extent practical, be developed, trial-tested, and corrected during the ITP prior to fuel loading to establish their adequacy.

8. Milestones and Power Hold Points

Applicants should establish appropriate hold points at selected milestones throughout the power-ascension test phase to ensure that relevant test results are evaluated and approved by the designated personnel or groups before proceeding with the power-ascension test phase. As a minimum, applicants should establish hold points at approximately 25%, 50%, and 75% power level test conditions for pressurized-water reactors (PWRs), and at appropriate power-to-flow test conditions for boiling-water reactors (BWRs).

9. Test Reports

The preoperational testing procedures and results should be retained as part of the plant's historical record. In addition, a summary of the startup testing should be included in a startup report, as discussed in Regulatory Guide 1.16, "Reporting of Operating Information — Appendix A Technical Specifications" (Ref. 6). This summary should include the following information:

(a) a description of the method and objectives for each test

(b) 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)

(c) 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

(d) justification for acceptance of systems or components that are not in conformance with design predictions or performance requirements

(e) conclusions regarding system or component adequacy

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants regarding the NRC staff's plans for using this regulatory guide. No backfitting is intended or approved in connection with the issuance of this guide.

Except in those cases in which an applicant proposes or has previously established an acceptable alternative method for complying with specified portions of the NRC's regulations, the NRC staff will use the methods described in this guide to evaluate submittals in connection with applications for construction permits and operating licenses under 10 CFR Part 50, as well as DC, ML, and COL applications under 10 CFR Part 52.

REGULATORY ANALYSIS / BACKFIT ANALYSIS

The regulatory analysis and backfit analysis for this regulatory guide are available in Draft Regulatory Guide DG-1166 "Initial Test Programs for Water-Cooled Nuclear Power Plants" (Ref. 7). The NRC issued DG-1166 in October 2006 to solicit public comment on the draft of this Revision 3 of Regulatory Guide 1.68.

REFERENCES

1. *U.S. Code of Federal Regulations,* Title 10, *Energy,* Part 50, "Domestic Licensing of Production and Utilization Facilities."[1]

2. *U.S. Code of Federal Regulations,* Title 10, *Energy,* Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."[1]

3. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

4. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

5. Regulatory Guide 1.29, "Seismic Design Classification," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

6. Regulatory Guide 1.16, "Reporting of Operating Information — Appendix A Technical Specifications," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

7. Draft Regulatory Guide DG-1166 "Initial Test Programs for Water-Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.[3]

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[1] All NRC regulations listed herein are available electronically through the Public Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/reading-rm/doc-collections/cfr/. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; email PDR@nrc.gov.

[2] All regulatory guides listed herein were published by the U.S. Nuclear Regulatory Commission. Where an ADAMS accession number is identified, the specified regulatory guide is available electronically through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. All other regulatory guides are available electronically through the Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/. Single copies of regulatory guides may also be obtained free of charge by writing the Reproduction and Distribution Services Section, ADM, USNRC, Washington, DC 20555-0001, or by fax to (301)415-2289, or by email to DISTRIBUTION@nrc.gov. Active guides may also be purchased from the National Technical Information Service (NTIS) on a standing order basis. Details on this service may be obtained by contacting NTIS at 5285 Port Royal Road, Springfield, Virginia 22161, online at http://www.ntis.gov, by telephone at (800) 553-NTIS (6847) or (703) 605-6000, or by fax to (703) 605-6900. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4209, by fax at (301) 415-3548, and by email to PDR@nrc.gov.

[3] Draft Regulatory Guide DG-1166 is available electronically under Accession #ML062750316 in the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4209, by fax at (301) 415-3548, and by email to PDR@nrc.gov.

8. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, NY, 1992.[4]

9. Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

10. Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

11. Regulatory Guide 1.41, "Preoperational Testing of Redundant Onsite Electric Power Systems To Verify Proper Load Group Assignments," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

12. Regulatory Guide 1.68.1, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling-Water Reactor Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

13. Regulatory Guide 1.10, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

14. Regulatory Guide 1.9. "Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

15. Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized-Water Reactors," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

16. 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," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

17. NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.[5]

18. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.[5]

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[4] Copies may be purchased from the American Society of Mechanical Engineers, Three Park Avenue, NewYork, NY 10016-5990; phone (212) 591-8500; fax (212) 591-8501; www.asme.org.

[5] All NUREG-series reports listed herein were published by the U.S. Nuclear Regulatory Commission. Copies are available for inspection or copying for a fee from the NRC's Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; email PDR@nrc.gov. Copies are also available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328, telephone (202) 512-1800; or from the National Technical Information Service (NTIS) at 5285 Port Royal Road, Springfield, Virginia 22161, online at http://www.ntis.gov, by telephone at (800) 553-NTIS (6847) or (703) 605-6000, or by fax to (703) 605-6900. NUREG-0612 and NUREG-1793 are also available electronically through the Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/ reading-rm/doc-collections/nuregs/.

19. Regulatory Guide 1.68.3, "Preoperational Testing of Instrument and Control Air Systems," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

20. Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling-Water Reactors," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

21. *U.S. Code of Federal Regulations,* Title 10, *Energy,* Part 20, "Standards for Protection Against Radiation."[1]

22. Regulatory Guide 1.68.2, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

23. NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling-Water Reactor Design," U.S. Nuclear Regulatory Commission, Washington, DC.[5]

24. NUREG-1793, "Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design," U.S. Nuclear Regulatory Commission, Washington, DC.[5]

25. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

APPENDIX A

INITIAL TEST PROGRAM

This appendix incorporates information relevant to design certification (DC), manufacturing license (ML), and combined license (COL) applications submitted under the applicable appendix to Title 10, Part 52, of the *Code of Federal Regulations* (10 CFR Part 52), “Licenses, Certifications, and Approvals for Nuclear Power Plants” (Ref. 2). This discussion reflects new DC, ML, and COL applicant design information; first-of-a-kind (FOAK) tests for these new reactor designs; and preoperational test program overlap with the inspection, test, analysis, and acceptance criteria (ITAAC) requirements under 10 CFR Part 52, which must be completed prior to fuel load for these new reactor designs. For DC applicants, the staff has reviewed the proposed testing associated primarily with the preoperational phase of the ITP. The COL applicant is responsible for identifying the remaining ITP phases.

Sections 1–5 of this appendix respectively 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, Section 6 identifies examples of FOAK tests, but does not encompass all such tests that are necessary for new plant designs. Since information on operating or testing experiences may not be available for these new plant designs, FOAK tests may be necessary to obtain such experience on which to base further testing. An example of a FOAK test is the AP1000 in-containment refueling water storage tank heatup test, which should be associated with the applicable testing or FOAK sections of the application.

1. Preoperational Testing

For new plants licensed in accordance with the requirements of 10 CFR Part 52, some design, construction, and testing activities are subject to ITAAC. For these plants, the NRC has developed a new construction inspection program (CIP). The ITAAC, which are incorporated into 10 CFR Part 52, are a design-specific pre-approved set of verifications that the COL applicant must meet prior to fuel load. ITAAC are required to be submitted as part of the DC and COL applications and are reviewed by the NRC staff in conjunction with those applications. Successful completion of all ITAAC demonstrates that the facility is constructed in accordance with the applicable regulations. However, preoperational testing not associated with ITAAC should be performed in accordance with this regulatory guide. In cases where preoperational tests are performed under both the ITP and the ITAAC, the test results will be recorded under both programs.

Following plant construction, testing should be accomplished to demonstrate the proper performance of structures, systems, and components (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 essentially been completed, and the designated construction tests and inspections have been satisfactorily completed. Construction and preliminary tests and inspections typically consist of activities such as initial instrument calibration, flushing, cleaning, wiring continuity and separation checks, hydrostatic pressure tests, and functional tests of components.

Preoperational tests should demonstrate that SSCs will operate in accordance with design in all operating modes and throughout the full design operating range. Testing should include, as appropriate, manual 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 should also 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. System vibration, expansion (in discrete temperature step increments), and restraint tests should also be conducted. This testing should include verification (by observations and measurements), as appropriate, that piping and component movements, vibrations, and expansions are acceptable for (1) Class 1, 2, and 3 systems, as defined by the Boiler and Pressure Vessel Code (Ref. 8) promulgated by the American Society of Mechanical Engineers (ASME); (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.

The SSCs and tests listed in subsections a. through o. of this section (below) are representative of the plant features that should undergo preoperational testing. This listing is provided to indicate the extent of testing necessary to demonstrate that the facility can be operated in accordance with design requirements. In general, sections a. through o. 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 listing provided in sections a. through o., since additional or different tests may be dictated by the particular plant design and/or the nomenclature applied to plant systems and features.

a. **Reactor Coolant System**

The reactor coolant system includes all pressure-containing components (such as pressure vessels, piping, pumps, and valves) within the reactor coolant pressure boundary, as defined in Section 50.2, “Definitions,” of 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities” (Ref. 1):

(1) *Integrated Systems Test*. Expansion and restraint tests to confirm acceptability of clearances and displacements of vessels; piping; piping hangers; and seismic and other holddown, support, or restraining devices in the as-built system during normal hot functional testing plant conditions. Hot and/or cold testing of the system with simultaneous operation of auxiliary systems.

(2) *Component Tests*. Appropriate tests and measurements of the following components of the reactor coolant system:

- (a) pressurizer
- (b) pumps, motors, and associated power sources
- (c) steam generators
- (d) pressure relief 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 internals, including reactor internals vent valves
(i) safety valves
(j) jet pumps

(3) *Vibration Tests*. Vibration monitoring of reactor internals[6] and other components, such as piping systems, heat exchangers, and rotating machinery.

(4) *Pressure Boundary Integrity Tests*. Hydrostatic tests to obtain baseline data for subsequent inservice testing.

b. <u>Reactivity Control Systems</u>

(1) *Control Rod System Tests*. Demonstrate normal operation and scram capability of the control rods (BWR) and control rod drive system. Demonstrate proper operation of functions such as control rod withdrawal inhibit features, runback features, rod withdrawal sequence control devices, and rod worth minimizers. 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. Demonstrate proper operation, including correct failure mode on loss of power, for the control rod drive system and proper operation of system alarms.

(2) *Chemical Control System Tests*. 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. Demonstrate proper rate of injection into the reactor coolant system and rate of dilution from the primary system. Verify redundancy, electrical independence, and operability of system components. Demonstrate correct failure mode on loss of power to system components.

(3) *Standby Liquid Control System Tests*. Demonstrate proper operation of the system with demineralized water, and verify proper solution mixing and adequacy of the sampling system. Demonstrate operability of instrumentation, controls, interlocks, and alarms. Verify operability of heaters, air spargers, and heat tracing. Conduct test firings of squib-actuated valves, and demonstrate design injection capability. Conduct tests, as appropriate, to verify redundancy and electrical independence.

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[6] Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing" (Ref. 9), should be used as guidance for vibration monitoring of reactor internals and other components.

c. Reactor Protection System and Engineered Safety Features Actuation Systems

Verify (by testing) the response time of each of the protection channels, including sensors.[7] 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 (snubbers, sensing lines, flow-limiting devices, etc.). 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. Demonstrate redundancy, electrical independence[8], coincidence, and safe failure on loss of power. If appropriate for the facility design, demonstrate operability of backup scram solenoid valves and devices, including detectors, logic, and final control elements to protect the facility for anticipated transients without scram (ATWS).

d. Residual or Decay Heat Removal Systems

Verify 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. Verify 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. Tests should be conducted, as appropriate, to verify redundancy and electrical independence (see footnote 8). The following list illustrates the systems and components that should be tested:

(1) turbine bypass valves
(2) steam line atmospheric dump valves
(3) relief valves
(4) safety valves
(5) decay or residual heat removal system
(6) reactor core isolation cooling system
(7) main steam isolation valves, branch steam isolation valves, and nonreturn valves
(8) auxiliary feedwater systems
(9) condensate storage system
(10) emergency cooling towers
(11) cooling water systems

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 steam generator levels. Operability of system pumps, valves, controls, and instrumentation should be demonstrated and, to the extent practical, testing should provide reasonable assurance that flow instabilities (e.g., "water hammer") will not occur in system components or piping, or inside the steam generators, during normal system startup and operation.

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7 Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems" (Ref. 10), provides a test criterion that is also acceptable for preoperational testing of protection channels, including sensors.

8 Regulatory Guide 1.41, "Preoperational Testing of Redundant Onsite Electric Power Systems To Verify Proper Load Group Assignments" (Ref. 11), should be used as guidance for appropriate tests.

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[9] 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 auxiliaries used for maintaining condenser vacuum

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 8). 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) cooling towers and associated auxiliaries
(3) raw water and service water cooling systems

g. **Electrical Systems**

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 protective devices, initiating devices, relaying and logic, transfer and trip devices, permissive and prohibit interlocks, instrumentation and alarms, and load-shedding features. Testing should also be conducted to demonstrate proper operation and load-carrying capability of breakers, motor controllers, switchgear, transformers, and 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, etc.). Redundancy and electrical independence should be demonstrated where appropriate.

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[9] Regulatory Guide 1.68.1, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling-Water Reactor Power Plants" (Ref. 12), should be used as guidance for appropriate tests.

Tests should demonstrate that the integrated system will perform as designed in response to simulated partial and full losses of offsite power sources. Tests should also demonstrate the design capability to transfer from onsite to offsite power sources.

(2) *Emergency AC Power Distribution System.* Demonstrate proper operation of protective devices, relaying and logic, transfer and trip devices, permissive and prohibit interlocks, instrumentation and alarms, and load-shedding or stripping features. Testing should also be conducted to demonstrate proper operation and load-carrying capability of breakers, motor controllers, switchgear, transformers, and 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(s) 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 tests (see footnote 8).

Loads supplied from the system, such as motor-generator (m-g) sets with flywheels, that 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 m-g sets. Full-load tests for vital buses should be conducted using normal and emergency power supplies to the bus. Testing should also 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 demonstrate system reliability,[10] 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 also 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 should also demonstrate the proper logic, correct setpoints for trip devices, and proper operation of initiating devices and permissive and prohibit interlocks, and should also demonstrate redundancy and electrical independence (see footnote 8). Emergency loads supplied should be confirmed to be in agreement with design sizing assumptions used for the power supplies.[11]

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10 Regulatory Guide 1.10, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants" (Ref. 13), should be used as guidance for applicable tests.

11 Regulatory Guide 1.9. "Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants" (Ref. 14), should be used as guidance for applicable tests.

(4) *DC System.* Demonstrate proper calibration and trip settings of protective devices, including relaying, and proper operation of permissive and prohibit interlocks. Demonstrate the design capability of battery chargers, transfer devices and inverters, and emergency lighting systems. Testing should also be conducted to demonstrate proper operation of breakers, transfer devices, inverters, and cables. This testing should simulate, as closely as practical, actual service conditions. Demonstrate operation of instrumentation and alarms and ground detection instrumentation. Also, demonstrate redundancy and electrical independence (see footnote 8) and show that actual total system amperage loads are in agreement with design loads. In addition, 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.

h. Engineered Safety Features

Engineered safety features 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, engineered safety features 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). Since engineered safety features vary for different plant designs, the listing below is only illustrative of those that are commonly used to prevent, limit, or mitigate the consequences of postulated accidents. If the subject plant design provides engineered safety features in addition to (or other than) those listed below, they should also be appropriately tested. Additionally, it should be noted that other categories of systems listed in Section 1 of this appendix include plant features commonly designated as engineered safety features, which should be appropriately tested, such as the emergency ac power distribution system [Section 1.g.(2)], emergency or standby ac power supplies [Section 1.g.(3)], the dc system [Section 1.g.(4)], and primary and secondary containments (Section 1.i).

The testing of engineered safety features should demonstrate that such features will perform satisfactorily in all expected operating configurations or modes. Testing should include demonstrations of correct logic and setpoints, 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. Concurrent testing of systems or features provided to ensure or support the operation of engineered safety features should also be conducted to demonstrate that they meet design requirements with the minimum number of operable components available for 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 engineered safety features; 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 leaktight covers, structures, or housings (low-pressure pneumatic or vacuum tests) provided to protect engineered safety features 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 8). The following list illustrates the systems and components that should be tested:

(1) *Emergency Core Cooling Systems (ECCS).*[12]

  (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 over-pressure 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 long-term cooling.

(2) *Auto-Depressurization System*. Testing should include such factors as accumulator capacity, relief valves, and operability using all alternative power and pneumatic supplies.

(3) *Containment Post-Accident Heat Removal Systems*. Testing of the containment spray system should include demonstrations that the spray nozzles, spray headers, and piping 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 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).[13]

(8) *Tanks and Other Sources of Water Used for 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*.

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[12] Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized-Water Reactors" (Ref. 15), provides specific guidance for PWRs.

[13] 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. 16).

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, the secondary containment ventilation system in BWRs is required 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.

Due consideration should also 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 8), 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 listing illustrates the systems, features, and performance demonstrations that should be included in the test program:

(1) containment design over-pressure structural tests[14] and vacuum tests for sub-atmospheric containments

(2) containment isolation valve functional and closure timing tests

(3) containment isolation valve leak rate tests[15] and in-leakage tests for sub-atmospheric containments

(4) containment penetration leakage tests[14]

(5) containment airlock leak rate tests[14]

(6) integrated containment leakage tests[14]

(7) main steamline leakage sealing systems

(8) primary and secondary containment isolation initiation logic tests

(9) containment purge system tests

(10) containment and containment annulus vacuum-breaker tests

(11) containment supplementary leak collection and exhaust system tests

(12) containment air purification and cleanup system tests

(13) containment inerting system tests

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14 Over-pressure structural tests should be conducted in accordance with Section III of the ASME Code (Ref. 8).

15 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.

(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)

(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 outgassing from concrete or components within the containment will not affect the test results.

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 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.

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 8).

The following listing 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

(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 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) process computers

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) process, criticality, effluent, and area radiation monitors

(2) personnel monitors and radiation survey instruments

(3) laboratory equipment used to analyze or measure radiation levels and radioactivity concentrations

(4) high-efficiency particulate air (HEPA) filters and charcoal absorbers (see footnote 13)

Tests should be conducted, as appropriate, to verify redundancy and electrical independence (see footnote 8).

l. 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. Testing or calculations should include, as appropriate, verification of tank volumes, capacities, holdup times, and proper operation and calibration of associated instrumentation. Spiked samples of the typical media or sources should be used, where necessary, 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) solid waste handling systems
(4) isolation features for steam generator blowdown
(5) isolation features for condenser offgas systems
(6) isolation features for ventilation systems
(7) isolation features for liquid radwaste effluent systems
(8) plant sampling systems

Solidification system tests should include verification that no free liquids are present in packaged wastes.

m. Fuel Storage and Handling Systems

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 8). The following list illustrates the equipment and component tests that the program should include:

(1) spent fuel pit cooling system tests, including testing of anti-siphon devices, high-radiation alarms, and low-water level alarms

(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[16] of cranes, hoists, and associated lifting and rigging equipment, including the fuel cask handling crane

(5) fuel transfer devices

(6) irradiated fuel pool or building ventilation system tests

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% of rated load, and full operational testing should be at 100% of rated load.

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[16] NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants" (Ref. 17), and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" (Ref. 18), should be used as guidance for tests on single-failure-proof overhead crane handling systems.

n. Auxiliary and Miscellaneous Systems

The applicant 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 8). The following list illustrates the types of systems and features for which performance should be demonstrated by testing:

(1) service and raw water cooling systems

(2) closed loop cooling water systems

(3) component cooling 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, and suppression 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[17] 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:

    (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

(15) shield cooling systems

(16) cooling and heating systems for the refueling water storage tank

(17) equipment and controls for establishing and maintaining sub-atmospheric pressures in sub-atmospheric containments

(18) heat tracing and freeze protection systems

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[17] Regulatory Guide 1.68.3, "Preoperational Testing of Instrument and Control Air Systems" (Ref. 19), provides detailed guidance on testing of instrument and control air systems.

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.

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 leaktightness of ducts and flow rates, proper direction of airflows, and proper control of space temperatures.

o. <u>Reactor Components Handling Systems</u>

Include the following:

(1) dynamic and static load tests (see footnote 16) of cranes, hoists, and associated lifting and rigging equipment (e.g., slings and strongbacks used during refueling or the preparation for refueling)

(2) demonstration of the operability of protective devices and interlocks

(3) demonstration of the operability of safety devices on equipment

Static testing of cranes, hoists, and associated lifting and rigging equipment should be at 125% of rated load, and full operational testing should be at 100% of rated load.

2. Initial Fuel Loading and Precritical Tests

Licensees should cautiously conduct initial fuel loading to preclude inadvertent criticality. To load on this basis, licensees should establish and follow specific safety measures, such as (a) ensuring that all applicable technical specification requirements and other prerequisites have been satisfied, (b) establishing requirements for continuous monitoring of the neutron flux throughout core loading so that all changes in the multiplication factor are observed, (c) establishing requirements for periodic data-taking, and (d) 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 of 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 (BWR), 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 guide describes typical prerequisites, precautions, and details that should be included in the initial fuel loading and precritical 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 listing illustrates the types of tests and verifications that should be conducted during or following initial fuel loading:

a. shutdown margin verification for partially (BWR) 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 for BWRs) 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[18] and boron concentration (PWR) of the reactor coolant system

f. reactor coolant system flow tests to establish that (i) vibration levels are acceptable, (ii) differential pressures across the fully loaded core and major components in the reactor coolant system are in accordance with design values, and (iii) 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

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. 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 coastdown. 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.

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[18] Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling-Water Reactors" (Ref. 20), discusses BWR design features to maintain water quality.

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 at least ½ count per second should register on the startup channels before startup begins, and the signal-to-noise ratio should be known to be greater than two. 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 technical specifications should be observed and complied with.

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

4. Low-Power Testing

Following initial criticality, licensees should conduct appropriate low-power tests (normally at less than 5% power) to (a) 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 (b) 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 listing 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 “PWR” for pressurized-water reactors and “BWR” for boiling-water reactors):

a. determination of boron and moderator temperature reactivity coefficients over the temperature and boron concentration ranges in which the reactor may initially be taken critical (PWR)
b. measurements of control rod and control rod bank reactivity worths 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 (PWR)
c. pseudo-rod-ejection test to verify calculational models and accident analysis assumptions (PWR)
d. determination of adequate overlap of source- and intermediate-range neutron instrumentation, and verification that proper operation of associated protective functions and alarms provides for plant protection in the low-power range (if not previously performed)
e. determination of 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 mispositioned or uncoupled control rods (measurements may be performed at a higher power level, depending on the sensitivity of in-core flux instrumentation)

f. neutron and gamma radiation surveys

g. determination of proper response of process and effluent radiation monitors (to the extent practical, responses of installed process and effluent radiation monitors should be verified by laboratory analyses of samples-from the process and/or effluent systems)

h. 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

i. demonstration of the 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

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

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

m. demonstration of the operability of the main steamline isolation valve leakage control system (BWR, during hot standby conditions)

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

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

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

q. demonstration of the operability of residual or decay heat removal systems, including atmospheric steam dump valves (PWR) and turbine bypass valves

r. demonstration of the operability of reactor coolant system purification and cleanup systems

s. vibration measurements of reactor vessel internals (see footnote 6) and reactor coolant system components (if not previously conducted)

t. performance of natural circulation tests of the reactor coolant system to confirm that the design heat removal capability exists, or to verify that flow (without pumps) or temperature data are comparable to prototype designs for which equivalent tests have been successfully completed (PWR)

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

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. To validate the analytical models used to predict plant responses to anticipated transients and postulated accidents, these tests should establish that measured responses are in accordance with predicted responses. The predicted responses should be developed using real or expected values of such attributes as beginning-of-life core reactivity coefficients, flow rates, pressures, temperatures, pump coastdown 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 also 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 listing 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 "PWR" for pressurized-water reactors, and "BWR" for boiling water reactors:

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

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 (DNBR), radial and axial power peaking factors, maximum average planar linear heat generation rate (MAPLHGR), minimum critical power ratio (MCPR), quadrant power tilt, and other important parameters are in accordance with design values throughout the permissible range of power-to-flow conditions. (25%, 50%, 75%, 100%)

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). (BWR)

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% – 85%) (PWR) Results of xenon oscillation tests performed at plants of essentially identical design can be used to substitute or supplement this testing.

e. Conduct a pseudo-rod-ejection test to validate the rod ejection accident analysis. (Greater than 10% power with control rod banks at the full power rod insertion limit) (PWR) This test need not be repeated for facilities using calculational models and designs identical to prototype facilities.

f. 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%) (PWR)

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

h. 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.

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

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

k. Demonstrate that ECCS high-pressure coolant injection systems 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% – 50% range for BWRs with steam- or electric-driven pumps, if not previously conducted. For PWRs, the testing should be in accordance with Regulatory Guide 1.79 (Ref. 15).

l. 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 (RHR) system in steam condensing mode, reactor core isolation cooling (RCIC) system, and auxiliary feedwater system. Testing of the auxiliary 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. (Prior to exceeding 25% power)

m. 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.

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

o. Calibrate instrumentation and demonstrate the proper response of reactor coolant leak detection systems, if not previously demonstrated.

p. Conduct vibration monitoring of reactor internals during steady-state and transient operation to establish that design limits are not exceeded (see footnote 6), if this test has not previously been completed.

q. Verify the proper operation of failed fuel detection systems. (25%, 100%)

r. Verify by review and evaluation of printouts and/or cathode ray tube (CRT) 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%, 50%, 75%, 100%)

s. Calibrate, as necessary, and verify the performance of major or principal plant control systems, including T-average controller; automatic reactor control system; boron addition systems (PWR); integrated control system; pressurizer control system; reactor coolant flow control system; main, auxiliary, and emergency feedwater control systems; hotwell level control systems; steam pressure control systems; and reactor coolant makeup and letdown control systems. (25%, 50%, 75%, 100%)

t. If not previously accomplished, verify, as appropriate, the operability, response times, relieving capacities, setpoints, and reset pressures for pressurizer relief valves; main steamline relief valves; atmospheric steam dump valves; turbine bypass valves; and turbine stop, intercept, and control valves. (25%) During transient tests, verify operability, setpoints, and reset pressures of relief valves.

u. Verify operability and response times of main and branch steamline 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%)

v. Verify that the main steam and feedwater systems operate in accordance with design performance requirements. (25%, 50%, 75%, 100%)

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%)

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%, 100%)

y. Calibrate, as required, and verify the proper operation of important instrumentation systems, including reactor coolant system flow; core flow, level, and temperature; in-core and ex-core neutron flux; and instruments and systems used to calculate thermal power level (heat balance) of the reactor. (25%, 50%, 75%, 100%)

z. Demonstrate that process and effluent radiation monitoring systems are responding correctly by performing independent laboratory 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%, 50%, 75%, 100%)

bb. Conduct neutron and gamma radiation surveys to establish the adequacy of shielding and identify high-radiation zones as defined in 10 CFR Part 20, “Standards for Protection Against Radiation” (Ref. 21). (50%, 100%)

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.[19] (Greater than or equal to 10% 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 steamline tunnel, continue to maintain their service areas within the design limits. (50%, 100%)

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

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%, 50%, 75%, 100%)

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 coastdown in flow for the system. (100%)

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). (In the range of 10–20% 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%, 90%)

ll. Demonstrate that the dynamic response of the plant is in accordance with design requirements for turbine trip. This test may be combined with item nn. (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 overspeed). (100%)

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%)

nn. Demonstrate that the dynamic response of the plant is in accordance with design for the case of 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 overspeed condition. The test should be initiated with the plant's electrical distribution system aligned for normal full-power operation. (100%)

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. Tests performed earlier in the test program need not be repeated.

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[19] Regulatory Guide 1.68.2, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants" (Ref. 22), should be used as guidance for demonstration of this capability.

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

FOAK tests are defined as new, unique, or special tests for new design features associated with SSCs that are part of a new reactor design under 10 CFR Part 52. The ITP functional testing requirements necessary to verify FOAK test performance should be identified at an early date in the DC, ML, and COL application process to account for the final design of the plant.

For example, the staff accepted the ITP proposed by the applicant in the final safety evaluation report (FSER) for the the Advanced Boiling-Water Reactor (ABWR), which the NRC published as NUREG-1503 (Ref. 23). During the ABWR DC review, the staff did not specifically identify individual FOAK tests. However, several examples exist in ABWR test abstracts identified in Section 14.2 of the DCD. The COL applicant should identify all FOAK tests in the FSAR.

Another example relates to the AP1000 standard design, for which the staff accepted the ITP proposed by the applicant in the related FSER, which the NRC published as NUREG-1793 (Ref. 24). The AP1000 design utilizes new passive design features, which have not been implemented in previous commercial nuclear power plants. In some cases, testing is only required for the first AP1000 plant to be constructed; in other cases, testing is only required for the first three AP1000 plants to be constructed. The following are some examples of FOAK tests that the staff accepted:

a. Verify that the passive core cooling system performs its design function by completing the in-containment refueling water storage tank (IRWST) heatup test during the preoperation test phase (first plant only).

b. Verify natural circulation of reactor coolant through the steam generators and residual heat removal heat exchangers during the low-power test phase (first plant only).

c. Verify the natural circulation function of the passive core cooling system by completing the core makeup tank heatup recirculation tests during the preoperational test phase (first three plants only).

d. Verify the blowdown function of the reactor coolant system to the IRWST by completing the automatic depressurization system blowdown test during the preoperational hot functional testing phase (first three plants only).

APPENDIX B

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

For new plants licensed in accordance with the requirements of 10 CFR Part 52, the NRC will implement construction inspection programs (CIPs) and conduct inspections of initial test programs (ITPs) beginning before preoperational testing and continuing through startup. The inspections, tests, analyses, and acceptance criteria (ITAAC) incorporated into 10 CFR Part 52 comprise a design-specific pre-approved set of verifications that applicants must meet prior to fuel load. These inspections are intended to determine, on a selective basis, whether the applicant's ITPs, as described in the final safety analysis report (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 applicant 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. In cases where NRC test verification satisfies both the ITAAC verification program and the ITP described in this regulatory guide, the NRC will record acceptable test results under both programs.

For the NRC to implement inspection programs, the applicant should have copies of its test procedures available for examination by the NRC's regional office personnel approximately 60 days prior to the scheduled preoperational tests, and copies of procedures for fuel loading, initial startup testing, and supporting activities not less than 60 days prior to 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 applicant.

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 prior to the start of the formal test program and is intended to determine whether the applicant has established a set of administrative procedures that will ensure that the programs are carried out in accordance with the methods described in the FSAR.

(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 applicant has implemented administrative controls identified in item (1) above.

(4) Confirm that the applicant has evaluated the test results and either concluded that those results are satisfactory and meet the acceptance criteria, or initiated corrective action.

(5) Confirm that the applicant 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 applicant is appropriately documenting and evaluating information, and whether the resulting technical conclusions are valid.

(7) Witness parts of preoperational, fuel loading, and startup tests to determine whether they are being conducted in the manner described in the applicant's administrative and test procedures, and whether they are being performed in a technically competent manner.

APPENDIX C

PREPARATION OF PROCEDURES

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

1. Preoperational Test Procedures

a. Prerequisites

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 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[20]
   - leakage
   - opening and closing times
   - valve stroke
   - position indication
   - torque- and travel-limiting settings
   - operability against pressure

   (b) pumps[20]
   - direction of rotation
   - vibration
   - motor load versus time
   - seal or gland leakage
   - seal cooling
   - flow and pressure characteristics
   - lubrication
   - acceleration and coastdown

[20] Section XI of the ASME Boiler and Pressure Vessel Code (Ref. 8) provides requirements for inservice testing of pumps and valves in nuclear power plants. The applicant should examine these requirements for applicability to its preoperational test programs.

(c) motors and generators
- direction of rotation
- vibration
- thermal overload protection, margins between setpoints, and full load running amps
- lubrication
- Megger or hi-pot tests
- supply voltage
- phase-to-phase checks
- neutral current
- acceleration under load
- temperature rise
- phase currents
- load acceptance capability versus both time and load (generators)

(d) piping and vessels
- hydrostatic test
- leaktightness
- cleaning, flushing, and layup[21]
- clearance of obstructions
- support adjustments
- proper gasketing
- bolt torque
- insulation
- filling and venting

(e) electrical and instrumentation and control
- verification that sensing lines are clear for process sensors and instrument root valves are open
- voltage
- frequency
- current
- circuit breaker operation
- power source identification
- bus transfers
- trip settings
- operation of interlocks, prohibits, and permissives
- operation of logic systems
- calibration
- control transformer settings
- temperature effects
- range checks
- response times

[21] Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" (Ref. 25), should be used as guidance.

b. Test Objectives

Objectives of the test should be 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. Special Precautions

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. Initial System Conditions

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. Environmental Conditions

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. Acceptance Criteria

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 designated 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 checkoff 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. <u>Documentation of Test Results</u>

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.

2. Fuel Loading

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

a. <u>Prerequisites for Fuel Loading</u>

(1) Specify the composition, duties, and emergency procedure responsibilities of the fuel handling crew.

(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 have been calibrated, and are 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% of full power).

(14) Establish water quality and identify limits.

(15) Establish and verify the fuel loading boron concentration.

b. Procedure Details

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 regarding 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 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 unborated 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 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) 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)

(14) approvals required for changing the procedure

c. **Limitations and Actions**

(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, 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.

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. This 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 ½ count per second) should register on startup channels before the startup begins, and the signal-to-noise ratio should be known to be greater than two. 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% – 20%).

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%, 50%, 75%, and 100%.

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%).

Individual test procedures should include instructions and precautions for establishing 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:

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

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

c. 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.

d. Reset high-flux trips, just prior to ascending to the next level, to a value no greater than 20% beyond the power of the next level unless technical specification limits are more restrictive.

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

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

g. Perform reactor coolant leak checks.

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