U.S. NUCLEAR REGULATORY COMMISSION

September 2012
Revision 2

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.68.1

(Draft was issued as DG-1265, dated March 2012)

Initial Test Program of Condensate and Feedwater Systems for Light-Water Reactors

A. INTRODUCTION

This regulatory guide describes methods that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable when developing preoperational, initial plant startup, and power ascension tests for light water reactor (LWR) power plant condensate, feedwater (FW), startup feedwater (SFW), auxiliary feedwater (AFW), and emergency feedwater (EFW) systems. It also includes recommended tests for new reactor condensate and FW systems for the advanced boiling water reactor (BWR), economic simplified BWR, U.S. evolutionary power reactor (EPR), U.S. advanced pressurized water reactor (PWR), and advanced passive 1000 (AP1000), including the AFW/EFW systems for the U.S. EPR and the U.S. advanced PWR. It also includes tests for active defense-in-depth system functions such as the SFW system in the AP1000 design.

This guide describes the methods that the NRC staff considers acceptable to implement multiple criteria in Appendix A, "General Design Criteria for Nuclear Power Plants," of Title 10, of the *Code of Federal Regulations* (CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities" (10 CFR Part 50) (Ref. 1), including, but not limited to, Criterion 4, "Environmental and Dynamic Effects Design Basis," Criterion 11, "Reactor Inherent Protection," Criterion 14, "Reactor Coolant Pressure Boundary," and Criterion 15, "Reactor Coolant System Design" regarding the initial test program (ITP) for condensate and FW systems, including condensate storage and supply, for LWRs and for SFW, AFW, or EFW systems for PWRs.

This guide also describes methods that the NRC staff finds acceptable for testing condensate, FW, SFW/AFW/EFW (PWR only) structures, systems, and components (SSCs) in operating plants and new facilities in accordance with Subpart B, "Standard Design Certifications," and Subpart C, "Combined Licenses," of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 2).

Criterion XI, "Test Control," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 requires licensees to establish a test program to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

Regulatory Guide (RG) 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants" (Ref. 3), identifies acceptable ITP test methods for nuclear power plant SSCs. RG 1.68 describes tests for LWR power conversion SSCs to ensure that these SSCs will perform their design basis functions and to aid in minimizing the probability of SSC malfunctions during subsequent plant operations. This regulatory guide describes in more detail the type and nature of LWR condensate, FW, and SFW/AFW/EFW system tests that the NRC staff considers acceptable.

This revision to RG 1.68.1 incorporates lessons learned information and operating experience gained since the last revision of this guide for the types of problems that can be detected in condensate, FW, and AFW/EFW systems during preoperational, initial plant startup, and power ascension tests.

The NRC issues regulatory guides to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required.

This regulatory guide contains information collection requirements covered by 10 CFR Part 50 and 10 CFR Part 52 that the Office of Management and Budget (OMB) approved under OMB control number 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. This regulatory guide is a rule as designated in the Congressional Review Act (5 U.S.C. 801-808). However, the NRC has determined this regulatory guide is not a major rule as designated by the Congressional Review Act and has verified this determination with the OMB.

B. DISCUSSION

Reason for Change

Regulatory guide 1.68.1 (Ref. 4) was initially developed to provide guidance on preoperational and startup testing of condensate and feedwater systems for BWRs. The guide was first revised in 1977 to incorporate information from the U.S. Atomic Energy Commission Report WASH-1260, "Evaluation of Incidents or Primary Coolant Release from Operating Boiling Water Reactors," dated October 30, 1972 (Ref. 5). Revision 2 is being issued for two reasons: (1) to expand the scope of the guide to encompass preoperational, initial plant startup, and power ascension tests for the condensate and feedwater systems in all types of light water reactor facilities licensed under 10 CFR Parts 50 and 52; and (2) to incorporate lessons learned by the NRC staff since the 1977 revision.

Background

Abnormal occurrences associated with FW systems at operating nuclear power plants have resulted from excessive vibration of system components and piping. Thorough testing during the ITP could have identified these operating problems. Vibration-induced pipe cracking was the most frequent recurring problem with condensate and FW systems. In many instances, reportable occurrences from problems with pipes and pipe fittings were responsible for safety related events and outage times at LWRs. For additional details on vibration testing at power, see regulatory position C.2.h in this guide.

Since the NRC last revised RG 1.68.1, additional equipment failures have been identified with both non-safety related and safety related condensate and FW systems in LWRs and safety related AFW/EFW systems in PWRs. As discussed in the information notices and generic letters referenced in this regulatory guide, the ITP should be capable of detecting the conditions that cause these types of equipment failures. These systems possess both safety related and non-safety related functions related to the performance and control of the reactor vessel level in BWRs and steam generator level in PWRs. Based on these lessons learned, the NRC staff concluded that it should provide further guidance on the ITP for condensate, FW, SFW, AFW, and EFW systems identified in this guide.

Harmonization with International Standards

The International Atomic Energy Agency (IAEA) has established a series of safety guides and standards constituting a high level of safety for protecting people and the environment. IAEA safety guides are international standards to help users striving to achieve high levels of safety. Pertinent to this regulatory guide, IAEA Safety Guide NS-G-1.9, “Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants” (Ref. 6), issued in 2004, addresses design considerations for the FW systems in LWRs and AFW/EFW systems in PWRs in Section 4. In addition, Draft Specific Safety Guide DS 446, a pending revision of IAEA Safety Guide NS-G-2.9, “Commissioning for Nuclear Power Plants” (Ref. 7), issued in 2003, addressees commissioning tests for plant systems. The NRC has an interest in facilitating the harmonization of standards used domestically and internationally. In this case there are many similar elements between this regulatory guide and the corresponding sections of the safety guide. This regulatory guide consistently implements and details the principles and basic safety aspects provided in IAEA Safety Guide NS-G-1.9 and NS-G-2.9.

C. STAFF REGULATORY GUIDANCE

Licensees and applicants should develop preoperational, initial plant startup, and power ascension tests for the condensate and FW systems of LWRs. The testing programs should be designed to ensure that these systems will accomplish their required functions under normal and transient conditions as stated in the safety analysis report.

All preoperational activities should be verified complete before testing begins. Preoperational activities include proper system physical installation, system hydrostatic testing, system cleanliness verification, and initial verification of safety devices (e.g., relief valves and electrical trip devices). Preoperational testing should be completed before fuel loading to determine component operability and performance and to verify that system installation meets the design. Although the licensee should perform as much of this type of testing as practicable during this phase, the unavailability of reactor power, system flow paths, or other factors may limit the ability to conduct some system-level testing and component testing. Thus, the licensee should complete these tests as part of its ITP. Testing evolutions should take note of overly sensitive controls and switches that could be problematic during routine plant operation. Vibration monitoring should include the visual inspection of possible wear points in the systems.

The licensee may not need to repeat some tests that it satisfactorily completed during preoperational testing during the startup or power ascension test phase. Engineering evaluations should be used to ensure that changing plant system and environmental conditions during the startup and power ascension phases would not change the results of tests conducted earlier during preoperational testing. Preoperational and low power testing should be evaluated for or performed under the harshest environmental conditions postulated for the design. In most cases, only evaluations are needed to determine if the design functions under the harshest design basis accident conditions. Testing should demonstrate that the licensee has met design assumptions for the worst case energy source (i.e., voltage/frequency and steam/air pressure).

Consideration should be given to integral testing at the system or plant level whenever possible. Integrated testing of complete systems under the harshest environmental conditions practical will help verify system reliability and operability in accordance with the design basis.

As a prerequisite to preoperational, startup, and power ascension tests, the licensee should verify that noncondensable gases and steam voids in the condensate, FW, SFW, and AFW/EFW systems are kept to acceptable levels. Normal system filling and venting procedures should be completed before checking for air or steam voids in the system. The licensee should verify that noncondensable gases precipitating out of the system or steam voids do not collect into piping pockets and degrade system performance. This verification should be completed by either performing nondestructive examination techniques, opening vent valves to remove noncondensable gases or by methods justified through an engineering evaluation. The engineering evaluation should consider void volume, void transport to pumps, pump void acceptance criteria and include performance of void transport analysis. The evaluation should document the rationale and determination that gas intrusion or steam voids into the system would not adversely affect the ability of the system to perform its function. If noncondensable gases are vented through high point vent valves, verify closure of the valves before starting the pumps.

1. Preoperational Testing

The preoperational phase of the ITP should include tests and measurements, where applicable to that design, to verify that:

a. Pumps used to provide FW flow (condensate, condensate booster, FW booster, FW, and SFW pumps), including pump subsystems such as lube oil and seal water, operate properly. Tests should confirm that the pumps satisfy performance requirements, including required suction head, flow rate, net positive suction head (NPSH), and overspeed characteristics. Although preoperational testing does not require an alternate source of steam for turbine drive pumps, testing should include, to the extent practical, all flow paths and pump alignments used during normal, abnormal, and emergency conditions.

b. Tests the adequacy of the automatic condensate makeup flow from the condensate storage tank to the hotwell. Tests the adequacy of the automatic recirculation function for all associated pumps and valves.

c. Tests system logic and correct setpoints of trips, automatic starts, and permissive and prohibitive interlocks in the starting and shutdown controls for the pump drivers. Verify the proper operation of all controls used for the manual and automatic starting and stopping of the pump drivers after a loss of power and degraded modes of operation. The condensate, FW, and SFW pumps trip on actuation of the following signals:

    1. high steam generator level in any steam generator,

    2. low suction pressure, and

3. a number of protective trips (e.g., lubricating oil, turbine overspeed, low turbine exhaust, excessive thrust bearing wear, high discharge pressure, suction isolation valve not fully open, and condensate pump trip).

d. The testing of safety related digital instrumentation and control (I&C) system software logic should follow the guidance in RG 1.171, "Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants" (Ref. 8).

e. I&Cs and valves used for trips and bypass flow of the filter demineralizers/polishers, condensate and FW heaters, and condensate and FW heater drain pumps operate properly.

f. All valves used for adjusting the condensate, FW and SFW flow rates operate properly. Tests should verify the proper response of valves for the design operating range and correct operation of protective features such as thermal overload devices and under-voltage sensing devices incorporated in the design of valve operators and associated control circuitry.

g. Tests should verify proper check valve design flow rates. Tests should verify the correct valve orientation for valves that are dependent on specific flow direction. Ensure that normal valve positioning is used during leak rate tests. Tests should verify that check valves function as designed when subjected to worst case design interruptions of flow or differential pressure. In some cases, the lowest system differential pressure may be limiting for leakage.

h. Tests or inspections or both should ensure that there is no evidence of water hammer under the most limiting conditions during the testing of various system alignments, component trips, and automatic component starts.

i. Sensors and associated instrumentation that provide inputs to the FW control systems operate properly. Tests should verify stable and accurate outputs in response to test signals. The testing of safety-related digital I&C system software logic should follow the guidance in RG 1.171 (Ref. 8). Tests should verify that external signals cannot influence the operation of the system (i.e., signal optical isolation functions are working between different I&C logic platforms).

j. The condensate, condensate makeup, FW and SFW control systems operate properly. Tests should verify the proper response of individual components in the control system (including programmers, summers, and signal modifiers) and the overall response of the control system, including the final element control. Tests should verify control system functions under simulated malfunctions (i.e., FW controller failure to minimum and maximum flow demand) and under simulated plant transients (i.e., main steamline isolation valve closure at full flow conditions and turbine trip without bypass at full flow conditions). Tests should also be conducted under the normal and most restrictive control modes (e.g., single- and multiple-element control).

k. Instrumentation and alarms used to monitor the performance of condensate, FW and SFW systems operate properly.

l. Tests all controls, instruments, alarms, protection functions, and redundant features have train separation and verify that train separation is acceptable.

m. FW containment isolation valves satisfy General Design Criterion (GDC) 55, "Reactor Coolant Pressure Boundary Penetrating Containment," GDC 56, "Primary Containment Isolation," and GDC 57, "Closed Systems Isolation Valves," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 for the automatic and remote manual valve isolation functions.

n. Condensate and FW isolation valves close on simulated high steam generator or reactor vessel level, reactor trip and low T_{avg} simulated signals in reactor coolant system loops.

o. Tests the AFW/EFW system instrumentation and alarms used to monitor, properly operate and verify the performance of the system. Sensors and associated instrumentation that provide inputs to the AFW/EFW control systems operate properly. Tests should verify stable and accurate outputs in response to test signals. The testing of safety-related digital I&C system software logic should follow the guidance in RG 1.171 (Ref. 8). Tests should verify that external signals cannot influence the operation of the system (i.e., signal optical isolation functions are working between different I&C logic platforms).

p. Tests should verify the proper operation of the AFW/EFW crosstie and alternate flow path valves to ensure that system redundancy is available for each train. The AFW/EFW water source and alternate sources are greater than, or equal to, the design rated volume needed for safety related use. Tests should verify that the AFW/EFW pumps can deliver the design rated flow to each of the steam generators at the acceptable design water level range in the condensate storage tank.

2. Startup and Power Ascension Testing

The startup phase of the ITP should include at least tests and measurements, where applicable to that design, to verify the following:

a. The condensate and FW systems, including, but not limited to, FW heater performance, filtration, deaerator, and demineralization, during each phase of the power ascension test program (25-, 50-, 75-, and 100-percent reactor power consistent with RG 1.68) are operating properly.

b. The condensate, FW, FW booster, and SFW pumps during system transient or off-normal suction source conditions (i.e., conditions that affect the temperature, static or pressure head, and vapor pressure of source liquids) at low power (less than, or equal to, 15-percent reactor power) and during each phase of the power ascension test program (25-, 50-, 75-, and 100-percent reactor power consistent with RG 1.68) have proper NPSH.

c. The FW control system responds properly in the automatic and manual mode of control to the reactor pressure vessel or steam generator. Verify the adequacy of the control system response to positive or negative design power step changes and ramp rates. Tests should verify that the system can be operated in the manual mode and that transfer to and from the automatic mode can be accomplished in accordance with design requirements (at low power less than, or equal to, 15-percent reactor power) and during each phase of the power ascension test program (25-, 50-, 75-, and 100-percent reactor power consistent with RG 1.68).

d. For variable speed pumps, the pump speed controller operates within predetermined limits. The stability and response characteristics of the automatic control system meet the performance requirements for normal plant operation. Tests should verify that the acceptance criteria for maximum and minimum water levels in the reactor vessel (BWR) or steam generator (PWR) are not exceeded as a result of plant transients, such as turbine trip and main steam isolation valve closure, with the control system in the automatic control mode (25-, 50-, 75-, and 100-percent reactor power consistent with RG 1.68).

e. Adequate condensate and FW chemistry control exists during each phase (25-, 50-, 75-, and 100-percent reactor power consistent with RG 1.68) of the power ascension test program. This test includes verification that condensate filters, demineralizers, and polishers are performing their intended function at all power levels.

f. The response of the FW system meets performance requirements following the loss of a condensate or FW pump (100-percent reactor power). This test should include a condensate or FW pump trip at the reactor power designed for pump operation, including any automatic controlled runbacks to the appropriate power level. This test should be performed after check valve leak rate testing.

g. Condensate, FW, SFW, AFW and EFW piping is arranged and supported with consideration for vibration. Vibration levels for safety related and non-safety related system components and piping are within predetermined limits. Piping movements during heatup, steady-state, and transient operation are also within predetermined limits. Vibration and acoustic testing should include the monitoring of high-velocity flow rates and cavitation.

h. Adequate margins exist between system variables and setpoints of instruments that monitor these variables to prevent spurious actuation or loss of system pumps and motor-operated valves. RG 1.105, "Setpoints for Safety-Related Instrumentation" (Ref. 9), provides guidance for testing of instrument setpoints and instrument uncertainty determination for nuclear safety-related instrumentation. This guidance may also be used for non-safety-related SSCs to reduce spurious trips and improve reliability.

i. During power ascension testing of the FW system, proper reactor pressure vessel and steam generator water level are independently verified to check for instrument errors in the cold reference leg vessel level. At 100-percent power, verify that water-level instrument lines reach steady-state temperature conditions before assessing instrument errors.

j. Moisture separator reheaters effectively maintain a specific steam quality to ensure proper heating of FW heaters during each phase (25-, 50-, 75-, and 100-percent reactor power consistent with RG 1.68) of the power ascension test program. Tests should verify that the heater drain system collects steam from the moisture separator reheaters and from the high- and low-pressure FW heaters and returns cascading hot water back to the condensate and FW system. Tests should verify proper operation of all associated pumps and valves.

k. The SFW pumps perform their defense-in-depth function during the loss of normal FW with the steam generators at their normal operating pressure range.

l. The condensate and FW systems respond properly to an inadvertent automatic start of an SFW, AFW, or EFW pump at 100-percent power.

m. If applicable, the SFW system, including pump controls, operates during the loss of normal FW systems with the steam generators at their normal operating pressure and temperature. This test should verify that the SFW pumps can deliver the design rated flow to each of the steam generators from all water sources at the lowest available NPSH.

n. The AFW/EFW system operates properly during loss of normal FW or other residual heat removal systems with the steam generators at their normal operating pressure range. Tests should verify that the AFW/EFW pumps can deliver the design rated flow to each of the steam generators at the acceptable design water level range in the condensate storage tank. Tests should confirm that the AFW/EFW pumps satisfy performance requirements, including required suction head, flow rate, NPSH, and overspeed characteristics and meet the requirements in 10 CFR Part 50, Appendix A, General Design Criteria 34. "Residual Heat Removal."

o. If applicable, the AFW/EFW system, including pump controls, operates during the loss of normal FW systems with the steam generators at their normal operating pressure and temperature. This test should verify that the AFW/EFW pumps can deliver the design rated flow to each of the steam generators from all water sources at the lowest available NPSH.

p. Verify proper operation of all AFW/EFW controls used for the manual and automatic starting and stopping of the pumps. Test the AFW/EFW pumps start on engineering safety feature actuation signal.

q. In accordance with NUREG-0800, SRP Section 10.4.9, "Auxiliary Feedwater System (PWR)," (Ref. 10) the AFW/EFW pumps should complete a 48 hour endurance test. The test should be completed consistent with the most limiting operating scenario for the AFW/EFW pumps, followed by immediate pump shut down, cool down, restart and 1 hour of restart operation. The test provisions should be based on test acceptance criteria (e.g., bearing and bearing oil temperature, vibration, and pump room environmental qualification limits.)

r. Integral system testing of the SFW/AFW/EFW pumps, the main steam atmospheric dump valves and the turbine bypass valves to the condenser can depressurize the reactor coolant system by removing heat from the steam generators. For additional details on testing main steam atmospheric dump valves and turbine bypass valves, see RG 1.68, Appendix A, Sections 4.r and 5.s.

3. Test Reports

See Regulatory Position 9 in RG 1.68 for guidance on preoperational and initial startup test reports.

D. IMPLEMENTATION

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

Use by Applicants and Licensees

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

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

Use by NRC Staff

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

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

The NRC staff does not intend or approve any imposition or backfitting of the guidance in this regulatory guide. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this regulatory guide, unless the licensee makes a change to its licensing basis. The NRC staff does not expect or plan to request licensees to voluntarily adopt this regulatory guide to resolve a generic regulatory issue. The NRC staff does not expect or plan to initiate NRC regulatory action which would require the use of this regulatory guide. Examples of such unplanned NRC regulatory actions

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

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

include issuance of an order requiring the use of the regulatory guide, requests for information under 10 CFR 50.54(f) as to whether a licensee intends to commit to use of this regulatory guide, generic communication, or promulgation of a rule requiring the use of this regulatory guide without further backfit consideration.

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

Conclusion

This regulatory guide is not being imposed upon current licensees and may be voluntarily used by existing licensees. In addition, this regulatory guide is issued in conformance with all applicable internal NRC policies and procedures governing backfitting. Accordingly, the NRC staff issuance of this regulatory guide is not considered backfitting, as defined in 10 CFR 50.109(a)(1), nor is it deemed to be in conflict with any of the issue finality provisions in 10 CFR Part 52.

If a licensee believes that the NRC is either using this regulatory guide or requesting or requiring the licensee to implement the methods or processes in this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfit appeal with the NRC in accordance with the guidance in NUREG-1409 and NRC Management Directive 8.4.

GLOSSARY ACRONYMS

ADAMS—Agency Document Access and Management System

AFW—auxiliary feedwater system

BWR—boiling water reactor

CFR—code of federal regulations

EFW—emergency feedwater system

FW—feedwater

I&C—instrumentation and control

ITP—initial test program

LWR—light water reactor

NPSH—net positive suction head

NRC—Nuclear Regulatory Commission

OMB—Office of Management and Budget

PWR—pressurized water reactor

RG—regulatory guide

SFW—startup feedwater

SSC—structure, system and component

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2. 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, D.C.

3. Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 3, U.S. Nuclear Regulatory Commission, Washington, D.C.

4. Regulatory Guide 1.68.1, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling-Water Reactor Power Plants," Revision 1, dated January 1977, U.S. Regulatory Commission, Washington D.C.

5. WASH-1260, "Evaluation of Incidents of Primary Coolant Release from Operating Boiling-Water Reactors," Directorate of Regulatory Operations, October 30, 1972, U.S. Atomic Energy Commission, Washington D.C.

6. International Atomic Energy Agency (IAEA) Safety Standard No. NS-G-1.9, "Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants," International Atomic Energy Agency, Vienna, Austria, 2004.[4]

7. International Atomic Energy Agency (IAEA) Draft Specific Safety Guide DS 446, a pending revision of IAEA Safety Guide NS-G-2.9, "Commissioning for Nuclear Power Plants," issued in 2003, International Atomic Energy Agency, Vienna, Austria.

8. Regulatory Guide 1.171, "Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, D.C.

9. Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation," U.S. Nuclear Regulatory Commission, Washington, D.C.

10. NUREG-0800, Standard Review Plan (SRP), Section 10.4.9, "Auxiliary Feedwater System (PWR)." March 2007, U.S. Nuclear Regulatory Commission, Washington, D.C.

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

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

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IN 83-077, "Air/Gas Entrainment Events Resulting in System Failures," November 14, 1983. (ADAMS Accession No. ML083240150)

IN 84-006, "Steam Binding of Auxiliary Feed Water Pumps," January 25, 1984. (ADAMS Accession No. ML082970341)

IN 84-032, "Auxiliary Feed Water Sparger and Pipe Hanger Damage," April 18, 1984. (ADAMS Accession No. ML082840775)

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IN 89-080, “Potential for Water Hammer, Thermal Stratification, and Steam Binding in High-Pressure Coolant Injection Piping,” December 1, 1989. (ADAMS Accession No. ML031190089)

IN 90-045, “Overspeed of the Turbine-Driven Auxiliary Feedwater Pumps and Overpressurization of the Associated Piping Systems,” July 6, 1990. (ADAMS Accession No. ML082610386)

IN 91-038, “Thermal Stratification in Feedwater System Piping,” June 13, 1991 (ADAMS Accession No. ML031190533)

IN 91-050, “A Review of Water Hammer Events after 1985,” August 20, 1991. (ADAMS Accession No. ML031190397)

IN 91-058, “Dependency of Offset Disc Butterfly Valve’s Operation on Orientation with Respect to Flow,” September 20, 1991. (ADAMS Accession No. ML031190251)

IN 93-012, “Off-Gassing in Auxiliary Feedwater System Raw Water Sources,” February 11, 1993. (ADAMS Accession No. ML031080158)

IN 93-048, “Failure of Turbine-Driven Main Feedwater Pump To Trip because of Contaminated Oil,” July 6, 1993. (ADAMS Accession No. ML031070473)

IN 93-051, “Repetitive Overspeed Tripping of Turbine-Driven Auxiliary Feedwater Pumps,” July 9, 1993. (ADAMS Accession No. ML031070374)

IN 93-062, “Thermal Stratification of Water in BWR Reactor Vessels,” August 10, 1993. (ADAMS Accession No. ML031070192)

IN 94-008, “Potential for Surveillance Testing To Fail To Detect an Inoperable Main Steam Isolation Valve,” February 1, 1994. (ADAMS Accession No. ML031070034)

IN 94-044, “Main Steam Isolation Valve Failure To Close on Demand because of Inadequate Maintenance and Testing,” June 16, 1994. (ADAMS Accession No. ML091590421)

IN 94-066, “Overspeed of Turbine-Driven Pumps Caused by Binding in Stems of Governor Valves,” Supplement 1, June 16, 1995. (ADAMS Accession No. ML031060370)

IN 95-011, “Failure of Condensate Piping because of Erosion/Corrosion at a Flow-Straightening Device,” February, 24, 1995. (ADAMS Accession No. ML031060332)

IN 96-041, “Effects of a Decrease in Feedwater Temperature on Nuclear Instrumentation,” July 26, 1996. (ADAMS Accession No. ML031060009)

IN 99-014, “Unanticipated Reactor Water Draindown at Quad Cities Unit 2, Arkansas Nuclear One Unit 2, and Fitzpatrick,” May 5, 1999. (ADAMS Accession No. ML031040444)

IN 99-019, “Rupture of the Shell Side of a Feedwater Heater at the Point Beach Nuclear Plant,” June 23, 1999. (ADAMS Accession No. ML031040409)

IN 00-001, “Operational Issues Identified in Boiling Water Reactor Trip and Transient,” February 11, 2000. (ADAMS Accession No. ML003682692)

IN 01-009, “Main Feedwater System Degradation in Safety-Related ASME Code Class 2 Piping inside the Containment of a Pressurized Water Reactor,” June 12, 2001. (ADAMS Accession No. ML011490408)

IN 02-018, “Effect of Adding Gas into Water Storage Tanks on the Net Positive Suction Head for Pumps,” June 6, 2002. (ADAMS Accession No. ML021570158)

IN 04-001, “Auxiliary Feedwater Pump Recirculation Line Orifice Fouling—Potential Common Cause Failure,” January 21, 2004. (ADAMS Accession No. ML040140460)

IN 04-006, “Loss of Feedwater Isokinetic Sampling Probes at Dresden Units 2 and 3,” March 26, 2004. (ADAMS Accession No. ML040711214)

IN 05-023, “Vibration-Induced Degradation of Butterfly Valves,” August 1, 2005. (ADAMS Accession No. ML051740299)

IN 06-015, “Vibration-Induced Degradation and Failure of Safety-Related Valves,” July 27, 2006. (ADAMS Accession No. ML061790443)

IN 07-021, “Flow-Induced Vibration Could Cause Reflective Metal Insulation To Damage Nuclear Power Plant Piping,” June 11, 2007. (ADAMS Accession No. ML071150051)

<u>Notices of Violation Related to Condensate, Feedwater, and Auxiliary Feedwater</u>

NOV 50-390-90-19, dated October 15, 1990. (ADAMS Accession No. ML072600609)

Description: During work activities associated with the sandblasting and repainting of the Watts Bar Nuclear Plant, Unit 1, condensate storage tank, the licensee failed to cover the suction pipes to the AFW system. Failure to adequately isolate this suction pipe allowed sandblast material to become entrained in this section of the line, thereby creating a potential condition adverse to system operability and system design.

NOV 50-390-97-02, dated May 22, 1997. (ADAMS Accession No. ML072750078)

Description: An isolation valve at the Watts Bar Nuclear Plant, Unit 1 (motor-driven AFW pump steam generator No. 1 level control valve), was closed and was not tracked by any configuration control process for being out of its required position. Isolation of the valve would have prevented the motor-driven AFW pump from feeding the steam generator during an accident.