U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

June 2012
Division 1

DRAFT REGULATORY GUIDE

Contact: F. Talbot
(301) 415 3146

DRAFT REGULATORY GUIDE DG-1277

(New Regulatory Guide)

INITIAL TEST PROGRAM OF EMERGENCY CORE COOLING SYSTEMS FOR BOILING-WATER REACTORS

A. INTRODUCTION

This guide describes methods that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable to implement Title 10, of the Code of Federal Regulations, Part 50, “Domestic Licensing of Production and Utilization Facilities” (10 CFR Part 50) (Ref. 1), Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criteria (GDC) 4, 5, 33, 34, 35, 36, 37, and 55, with regard to preoperational, low power and power ascension testing features of emergency core cooling systems (ECCSs) for boiling-water reactors (BWRs). The guide also describes testing of the Isolation Condenser System (ICS) and the Reactor Core Isolation Cooling System (RCIC) which support functions that meet alternate water injection during station blackout to meet 10 CFR 50.63, “Loss of All Alternating Current Power” (Ref. 2) for core cooling. This regulatory guide (RG) also describes methods that the NRC staff finds acceptable for initial plant testing of ECCS structures, systems, and components (SSCs), in accordance with the regulations in 10 CFR 50.34(b)(6)(iii), “Plans for Preoperational Testing and Initial Operations,” and 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants” (Ref. 3), Subpart B, “Standard Design Certifications,” and Subpart C, “Combined Licenses,” 10 CFR 52.79(a)(28) “Plans for Preoperational Testing and Initial Operations.”

Nuclear power plant SSCs must be tested to quality standards commensurate with their importance to safety. 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 testing program to identify and perform all tests needed to demonstrate that SSCs will perform satisfactorily in service.

This testing program is to be conducted in accordance with written test procedures that incorporate the requirements and acceptance criteria in applicable design documents. The ECCS functions to be tested are those necessary to ensure that specified design functions of the ECCS are met

---

This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received final staff review or approval and does not represent an official NRC final staff position. Public comments are being solicited on this draft guide (including any implementation schedule) and its associated regulatory analysis or value/impact statement. Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Rules, Announcements, and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; submitted through the NRC’s interactive rulemaking Web page at http://www.nrc.gov; or faxed to (301) 492-3446. Copies of comments received may be examined at the NRC’s Public Document Room, 11555 Rockville Pike, Rockville, MD. Comments will be most helpful if received by August 15, 2012.

Electronic copies of this draft regulatory guide are available through the NRC’s interactive rulemaking Web page (see above); the NRC’s public Web site under Draft Regulatory Guides in the Regulatory Guides document collection of the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/; and the NRC’s Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML113550182. The regulatory analysis may be found in ADAMS under Accession No. ML113550199.

during any condition of normal operation, including abnormal operating occurrences, or because of postulated accident conditions.

RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," (Ref. 3), describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to preoperational, initial criticality, low power, and power ascension testing of nuclear power plant SSCs that perform functions important to safety. This draft regulatory guide (DG) describes initial plant testing acceptable to the staff specifically for ECCSs in BWRs. DG-1277 is applicable to all BWRs licensed under 10 CFR Part 50 and 10 CFR Part 52. In cases in which an SSC is not part of the specific nuclear plant design, the associated testing would not apply.

The NRC issues RGs 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 and licensees. Regulatory guides are not substitutes for regulations and compliance with them is not required.

DG-1277 contains information collection requirements covered by 10 CFR Part 50 and 10 CFR Part 52 that the Office of Management and Budget (OMB) approved under OMB control numbers 3150-0011 and 3150-0151, respectively. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or requirement unless the requesting document displays a currently valid OMB control number. This regulatory guide is a rule as designated in the Congressional Review Act (5 U.S.C. 801-808). However, OMB has not found it to be a major rule as designated in the Congressional Review Act.

B. DISCUSSION

The NRC staff recently updated the guidance in RG 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized-Water Reactors," (Ref. 4) to revise existing ECCS tests for Pressurized Water Reactors (PWRs) and included new ECCS tests for advanced PWRs. As a companion guide to RG 1.79, the NRC staff also concluded that it should provide new guidance about the scope of ECCS initial plant tests for BWRs as a result of the NRC's design certification of the advanced boiling-water reactor (ABWR) and the economic simplified boiling-water reactor (ESBWR). The NRC staff also concluded that this DG should include additional guidance for initial plant tests of ECCSs based on recent operating experience from BWRs.

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 present international good practices and increasingly reflect best practices 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. 5), issued in 2004, addresses design considerations for the ECCS in Sections 4.68 through 4.91. 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 section 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.

C. STAFF REGULATORY GUIDANCE

A comprehensive preoperational, low power and power ascension test program of the ECCS for BWRs should ensure that it will accomplish its intended functions when required. The initial test program should cover all test-related activities, including the following:

1. the development of test descriptions, test objectives, and specific acceptance criteria;
2. the preparation of test procedures;
3. the conduct of the tests and acquisition of system and component performance data; and
4. the resolution of deficiencies and deviations from expected performance.

RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," (Ref. 6) includes regulatory positions on design criteria, performance standards, and analysis methods on ECCS water sources that relate to all water-cooled reactor types (Section C.1.1) and specific to BWRs (Section C.3). The regulatory positions in RG 1.82 support ECCS water source test acceptance criteria for the regulatory positions in DG-1277.

The test program should include prerequisites for completion of construction tests and preoperational tests in coordination with the startup test group approval of test procedures, test configuration, and test initiation.

As a prerequisite to ECCS tests, verify that noncondensable gases in the ECCS systems are kept to an acceptable level. This verification should be accomplished by performing nondestructive examination techniques, by opening vent valves to remove noncondensable gases, or by using methods justified through an engineering evaluation. The engineering evaluation should consider void volume, void transport to pumps, and pump void acceptance criteria and should include performance of void transport analysis. The evaluation should document the rationale and determination that gas intrusion into the ECCS 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 ECCS pumps.

The test program should also include prerequisite tests for motor- and steam-operated pump capabilities and motor- and air-operated valve actuation times. This testing should be performed under the most limiting design-basis conditions and may be verified by either testing or analyses.

1. System Testing for BWR/2-6, the ABWR, and the Passive ESBWR Designs

For BWR plants, initial plant tests of the following ECCS systems are described below.

a. High-Pressure Core Flooder (HPCF)/High-Pressure Core Spray (HPCS) (ABWR—HPCF) (BWR/5-6—HPCS)

(1) Preoperational Flow Tests—Cold Conditions: Verify proper operation of the HPCF/HPCS system, including related auxiliary equipment, pumps, valves, and instrumentation and control (I&C). The suppression pool and condensate storage tank (CST) should be available as pump suction sources, and the reactor vessel should be properly prepared to receive injection flow. The instrument air system, makeup water condensate system, residual heat removal (RHR) system, remote shutdown system, reactor building cooling water system, and appropriate electrical power sources should be

available as needed to support the specified testing and appropriate system configuration. System testing should do the following:

(a) Verify the proper operation of system software-based I&C. Check system functional, performance, and interface requirements as specified in design specifications and hardware/software system specifications.

(b) Verify all component alarms and proper alarm actuation by operating the detectors of the alarm or using simulated alarm signals.

(c) Verify proper operation of motor-operated and air-operated valves, including opening and closing valves with operating switches, valve status indication, and travel timing, if applicable.

(d) Verify proper operation of pumps and motors during continuous run tests. Verify acceptable pump net positive suction head (NPSH) under limited test design flow conditions.

(e) Verify system NPSH requirements under combined operating conditions for pumps, valves, piping, and instruments in the system through the following tests.

(f) Rated Core Flooding Operational Tests: Check proper operation of the system at rated core flooding using the test line for flooding the suppression pool through the minimum flow line until the temperatures of the pump and motor bearing stabilize.

    (1) High-Pressure Flooding Operational Test: Check proper operation of the system at rated core flooding using test lines for injecting into the suppression pool (or CST) to the suppression pool. The test should be performed continuously from the pump motor start to minimum flow with lowest acceptable CST levels.

    (2) Reactor Injection Test: Check proper operation of the system using the core flood line to confirm that core flooding runout is performed.

    (3) Alternate Source Verification Test: Verify that the water source can be transferred from the CST to the suppression pool.

    (4) Automatic Start Test: Verify that the system start time is within safety injection requirements and that water hammer does not occur. Verify proper operation of keep fill and venting components to prevent water hammer damage.

(g) Verify proper pump and motor start sequence and actuation, using all possible start signals, including testing from the remote shutdown panel, operation of interlocks, protective devices, and all components subject to interlocks and protective devices. This includes proper operation of permissive, prohibit, and bypass functions.

(h) Verify proper operation while powered from the primary and alternate power sources, including power transfers.

(i) Verify acceptable pump/motor vibration levels and system piping movement during steady-state and transient operation. This test may also be performed with the expansion, vibration, and dynamics effects preoperational test.

(j) Verify that system flowpaths are acceptable via air testing through core spray spargers. Verify the proper core spray sparger flooding pattern in the reactor vessel.

(2) Power Ascension Flow Test—Hot Conditions: Verify the HPCF/HPCS system shall initiate automatically, when low water levels (Level 1 and 2) are reached during the initial transient following isolation. The minimum capacity and maximum delay time between the time the vessel water level drops below the set point and makeup water enters the vessel shall meet safety analysis requirements.

b. High-Pressure Coolant Injection System (HPCI) (BWR/3-4), High Pressure Feedwater Injection (HPFI) (BWR/2).

(1) Preoperational Flow Tests—Cold Conditions: Verify proper operation of the HPCI system or HFCI system, including related auxiliary equipment, pumps, valves, and I&C. The suppression pool and CST should be available as pump suction sources, and the reactor vessel should be properly prepared to receive injection flow. The preoperational tests should be similar to testing performed in Regulatory Position C.1.a above. However, testing differences may apply because the system injects into the feedwater injection line instead of spray nozzles or spargers above the reactor core. This preoperational test using an auxiliary steam supply to the HPCI turbine should be similar to testing of RCIC in Regulatory Position C.1.d below.

(2) Power Ascension Flow Test—Hot Conditions: Verify proper automatic operation of the HPCI system or HPFI system, including related auxiliary equipment, pumps, valves, and I&C. The suppression pool and CST should be available as pump suction sources, and the reactor vessel should be properly prepared to receive injection flow. The hot functional test should be similar to testing performed in Regulatory Position C.1.a.(2). However, testing differences may apply because the system injects into the feedwater injection line instead of spray nozzles or spargers above the reactor core.

c. Automatic Depressurization System (BWR/2-6, ABWR, and ESBWR)

(1) Preoperational Instrumentation and Control Test—Cold Conditions: This preoperational test should demonstrate proper operation of automatic depressurization system (ADS) I&C subsystem logic functions and safety system logic and control (SSLC) functions.

(a) A number of ADS safety relief valves (SRVs) operate from either ADS loss-of-coolant accident (LOCA) initiation logic signals or safety/relief steam pressure logic signals. The ADS is initiated by high drywell pressure and/or low reactor vessel water level. This test should verify that ADS logic functions meet their design acceptance criteria.

(b) The ADS logic preoperational test verifies integrated automatic decision making and trip logic functions associated with the safety actions of the ADS.

(c) This test should verify that ADS accumulator capacity meets the required number of cycles for operating the SRVs.

(2) Power Ascension Test—Hot Conditions: The objective of this test is to demonstrate that SRVs can be manually opened and closed properly during power operation. The plant should be at the appropriate operational configuration, with prerequisite testing complete. The applicable instructions should be checked or calibrated before testing begins.

(a) Perform a functional test of each SRV during plant heatup to the 50-percent power plateau. Open and close each valve to verify steamflow using discharge tail pipe temperature sensors. The SRV open and close indications and the tailpipe temperature/flow sensor should function as designed

(b) This test should verify that there is no leakage from the SRV by monitoring tail pipe temperature to confirm SRV closure.

(c) This test should verify the steamflow through each SRV should not vary significantly from the average for each valve. Each valve should properly reseat after testing.

The operators may also verify changing indications of SRV position in comparison to changing turbine valve positions and/or generator load output. These changes may also be evaluated to detect anomalies indicating restriction or blockage in a particular SRV tailpipe by making a valve-to-valve comparison.

d. Reactor Core Isolation Cooling (BWR/3-6, ABWR)

(1) Preoperational Flow Test—Cold Conditions: The purpose of this preoperational test is to test the signals to automatically start the reactor core isolation cooling (RCIC) system at low reactor water level or high drywell pressure and the signal for automatic isolation of the RCIC system at low steam pressure to the RCIC pump turbine. This test should be performed using temporary steam supply (e.g., auxiliary boiler), equipment, piping, and instrumentation as necessary for the test. Because preoperational testing is performed using a temporary steam supply, RCIC pump flow may be limited. If this is the case, document the issue and schedule completion of testing during the power ascension test phase. The preoperational tests should include individual component and integrated system tests. These tests should be performed using RCIC system design specifications to demonstrate the following tests:

(a) Verify proper operation of RCIC system software-based I&C. The test should verify system functional, performance, and interface requirements as specified in design specifications and hardware/software system specifications.

(b) Verify all RCIC component alarms and alarm actuations, using actual or simulated alarm signals.

(c) Verify alignment of RCIC system suction from the condensate storage pool and inject water into the reactor through the reactor feedwater line with the reactor at atmospheric conditions. This test should also verify proper operation of all motor-operated and air-operated valves, including operability, position indicators, and timing, if applicable.

(d) Verify proper operation of the RCIC turbine and supporting subcomponents. Perform an RCIC pump turbine quick start under simulated automatic initiation signal with suction from the CST. The test should demonstrate proper system flow rate and time to rated flow with no malfunctions in the system. Verify pump NPSH under limited design flow conditions. If additional testing is needed, this test may also be performed during the power ascension test phase.

(e) Verify acceptable RCIC pump/turbine vibration levels and system piping movements during both transient and steady-state operation. This test may also be performed with the expansion, vibration, and dynamic effects preoperational test.

(f) Verify proper RCIC system operation while powered from the primary and alternate power sources, including transfer, and in degraded modes for which the system is expected to remain operational. The RCIC system should demonstrate its ability to start without the aid of alternating current (AC), except for RCIC direct current/AC inverters.

(g) Verify proper operation of the RCIC system barometric condenser condensate pump and vacuum pump. For newer RCIC designs, the barometric condenser, condensate pump, and vacuum pump may have been eliminated; therefore, testing is not applicable.

(h) Verify the ability of the RCIC system to swap pump suction sources from the CST to the suppression pool without interrupting system operation.

(i) Verify that all RCIC system functions operate from redundant control locations, where appropriate.

(2) Low Power Flow Test—Hot Conditions: The purpose of the RCIC low power test is verify proper operation of the RCIC system over its expected operating pressure and flow ranges, and to demonstrate reliability to automatically start from cold standby with the reactor at power. Test the RCIC System through a full flow test line to the suppression pool and by flow injection directly into the reactor vessel.

(a) Test RCIC capability to inject water into the suppression pool in the manual and automatic start mode and steady-state operation at near rated reactor pressure in the full flow test mode. During this test, throttle pump discharge pressure in order to simulate reactor pressure and the expected pipeline pressure drop. The RCIC turbine speed control loop will be adjusted at near rated reactor pressure conditions.

(b) Test RCIC capability to inject water into the reactor vessel at rated reactor pressure to complete controller adjustments to demonstrate automatic starting from hot standby condition. The test should demonstrate automatic injection into the reactor vessel at rated reactor pressure. The test shall verify satisfactory RCIC system performance under the final set of controller settings after controller adjustment are made by small step changes in speed and flow demand and then verify system response at both low and near rated RCIC pump flow conditions.

(c) After completing RCIC system controller adjustments, test automatic initiation of the RCIC system from cold standby conditions (i.e., 72 hours without RCIC operation) to demonstrate RCIC system reliability. Collect system data in the full flow test mode to provide benchmark data for future surveillance tests. For these tests, evaluate RCIC system and related auxiliary system data. For the RCIC system steam line flow trip setting in the leak detection and isolation system trip logic, collect sufficient operating data to make adjustment and verify correct trip set points.

(d) During low power testing, the RCIC turbine should not trip or isolate during manual or automatic start tests.

(e) Testing should indicate the average RCIC pump discharge flow equal to or greater than the 100% rated value specified in the RCIC system process flow diagram for all operating modes.

(f) Testing should indicate the start time for the RCIC system from receipt of signal to delivering design flow within the limit specified in the applicable RCIC system design specification from low to rated reactor pressure.

(g) During low power testing, the RCIC turbine and pump flow control loops should be adjusted so that the RCIC flow related to variable responses to test inputs are at least quarter damped as stated in the RCIC design specifications.

(h) During low power testing, the RCIC Turbine Gland Seal System should be capable of preventing significant steam leakage to the atmosphere.

(i) For automatic RCIC start tests, the transient first start followed by turbine speed peaks should not exceed the requirements specified in the RCIC vendor Startup Test Specifications document.

(j) The RCIC Turbine Steam Supply line high flow isolation trip should be calibrated to actuate at the value specified in plant Technical Specifications.

e. Gravity-Driven Cooling System (ESBWR)

(1) Preoperational Instrumentation and Flow Tests—Cold Conditions: The gravity-drive cooling system (GDCS) is a unique ESBWR passive cooling system to provide gravity-driven flow into the reactor vessel for emergency core cooling during LOCA conditions. The objective of this test is to verify the operation of all four trains of the GDCS, including valves, logic, and instrumentation. The reactor vessel should be ready to accept GDCS flow. The required electrical power should be available for squib-valve operation. Instrument calibration and checks have been completed. To prevent actuation of single-use squib valves during the logic portion of testing, the valves may be isolated. The tests should do the following:

(a) Verify operation of all instruments and equipment to appropriate design logic combinations and instrument channel trips. Verify all GDCS functions from redundant control locations, where appropriate.

(b) Verify instrumentation and alarms functions used to monitor system operation and availability.

(c) Verify the operation of system valves, including time to open and close. The electrical power supplies should demonstrate their capability to actuate the "explosive chargers" used to open GDCS squib valves.

(d) Verify that GDCS flow from the GDCS pool and suppression pool through the reactor vessel is not obstructed.

(e) Verify that flow passages to the drywell are not obstructed.

(f) Verify the required GDCS design flow rate under the lowest possible suction pressure provided by GDCS pool level.

(g) Verify the adequacy of the instrument channel response times, as measured from process variable input signals to the applicable process actuator confirmation signal.

f. Isolation Condenser System (BWR/2, BWR/3, ESBWR)

(1) Preoperational Flow Test—Cold Conditions: The objective of this test is to verify operation of the isolation condenser system (ICS) loops, including valves, logic, and instrumentation. High-pressure nitrogen must be available to operate the spring-loaded condensate return valves, and nitrogen-operated pneumatic rotary motor isolation valves and electrical power are available to operate valves and controls. Performance should be observed and recorded during a series of individual component and integrated system tests to demonstrate the following:

(a) Verify proper calibration of instrumentation and operation of instrumentation and equipment in appropriate design combinations of logic and instrument channel trip. Verify proper functioning of instrumentation and alarms used to monitor system operation and availability.

(b) Verify proper operation of system valves, including timing. Verify that the steam flowpaths from the inside containment (IC)/passive containment cooling system (PCCS) pools to the atmosphere are not obstructed. Verify that isolation condenser steam and condensate-return piping flow passages are not obstructed.

(c) Verify proper operation of IC/PCCS pool level control. Verify that the isolation condenser pool subcompartment valves are locked open. Verify operation of isolation of the isolation condenser containment isolation valves upon receipt of simulated isolation signal.

(d) Verify acceptable instrument channel response times, as measured from each applicable process variable input signal to the applicable process actuator confirmation signal.

(2) Low-Power Test—Hot Conditions: The objective of this test is to demonstrate operation of the four isolation condensers when supplied with reactor steam at rated pressure. The instrumentation should be checked and calibrated. Any required expansion, vibration,

and temporary flow and temperature measurement instrumentation for ICS piping must be in place.

(a) At 20-percent steady-state power, initiate operation of one ICS train by opening the condensate return valve and condensate return bypass valve. Verify acceptable heat removal capability by measuring ICS steady-state flows, temperatures, and isolation valve/passive containment pool level changes and temperatures.

(b) Perform a heat removal capacity test on only one train of ICS. The objective of this test is to confirm proper startup, operation, and shutdown of one ICS train. Determine proper operation to verify measurement of vibration, displacement, and strain on the ICS heat exchanger, piping, and tubing. Verify measurement of steam inlet and condensate flow return to the reactor.

g. Standby Liquid Control System (ESBWR)

(1) Preoperational Flow Test—Cold Conditions: The objective of this preoperational test is to verify that the operation of the standby liquid control system (SLCS), including accumulator, tanks, control, logic, and instrumentation, is as specified. The reactor vessel should be available to inject water. Required interfacing systems should be available, as needed, to support the specified testing and the appropriate system configurations. To prevent actuation of single-use squib valves during the logic portion of this testing process, the valves may be isolated electrically to prevent actuation. This process of isolation, verification of the firing signal during the test, and subsequent reconnection must be controlled within the test document. Performance should be observed and recorded during a series of individual component and integrated system tests to demonstrate the following:

(a) Verify the proper calibration of instruments and the operation of all instruments and equipment in the required combinations of logic and instrument channel trip. Verify the proper functioning of instruments and alarms used to monitor system operation and availability.

(b) Verify proper functionality of redundant accumulator equipment room electric heaters.

(c) Verify proper operation of system valves, including timing, under expected operating conditions.

(d) Verify proper operation of the nitrogen pressurization system.

(e) Verify proper system flowpaths and discharge. Demineralized water may be used in place of neutron absorber water mixture.

(k) Verify proper operation of interlocks and equipment protective devices in valve controls.

(g) Verify proper operation of the squib-type injection valves.

(h) Verify acceptable instrument channel response times, as measured from each applicable process variable input signal to the applicable process actuator confirmation signal.

(i) Verify proper SLCS tank volume, temperature control, and concentration of the neutron absorber solution, boron-10 enrichment, before entry into a technical specification mode in which SLCS operability is required.

h. Low-Pressure Core Flooder (LPCF) mode of Residual Heat Removal (RHR) (ABWR) (BWR/2-6—Low-Pressure Coolant Injection/Low-Pressure Core Spray)

(1) Preoperational Flow Test—Cold Conditions: The purpose of this preoperational test is to test proper operation of the RHR system in the low-pressure core flooder (LPCF) mode. These tests should verify the following:

(a) Verify adequate NPSH to the RHR pumps in the LPCF operating mode.

(b) Verify proper operation of LPCF flow control valves and pumps within each train and cross-connecting train of the RHR system from all control locations.

(c) Verify proper operation while powered from the primary and alternate power sources, including transfer, automatic startup, timing, and sequencing. This test may also be performed with the integrated ECCS loss-of-offsite-power/LOCA preoperational test.

(d) Verify proper flow-induced vibration preoperational testing of the LPCF sparger structure and end bracket attachments.

i. Residual Heat Removal Systems (BWR/2-6, ABWR, ESBWR)

(1) Preoperational Flow Test—Cold Conditions: The objective of this preoperational test is to verify individual component and integrated RHR system tests for the RHR system shutdown cooling mode of operation. The RHR system LPCF mode is tested in Regulatory Position C.1.h above. The RHR system specification tests to support the shutdown cooling mode of operation include the following:

(a) Verify proper operation of instrumentation and system controls in all combinations of instrument channel trip logic.

(b) Verify proper operation of the RHR water leg pump with flow through a bypass loop around the water leg pump. The water leg pump discharges to the RHR main line to keep it and all of its branches filled with water. Any makeup water needed is drawn from the RHR pump suction line’s open path to the suppression pool. The ability to keep the RHR line filled with water prevents damaging water hammer during system transient. Tests should confirm the RHR water leg pump trip on startup of the RHR pump.

(c) Verify proper operation of the RHR system during shutdown cooling. Verify adequate NPSH to the RHR pumps from the suppression pool.

(d) Verify proper operation of interlocks and equipment protective devices in pump and valve controls, including those designed to protect low-pressure portions of the system from the reactor coolant system at high pressure.

(e) Verify proper operation of permissive, prohibit, and bypass functions.

(f) Verify acceptable pump and motor vibration levels and system piping movements during transient and steady-state operation.

(g) Verify the proper operating flow conditions of each RHR pump during continuous operation at design-rated flow, with a flowpath from the suppression pool through the RHR heat exchanger and return to the suppression pool.

(h) Verify proper operation of the RHR heat exchangers. This test may also be performed during the startup test phase when load test conditions from reactor steam are available.

(2) Low-Power Test—Hot Conditions: This low power test will demonstrate the ability of the RHR system to remove decay heat from the nuclear boiling system and safely place the plant in the shutdown cooling mode of operation. Pertinent system parameters will be monitored in the suppression pool cooling and shutdown cooling modes to verify that overall system operation and heat removal capabilities are in accordance with design requirements.

(a) Verify that the RHR system is capable of operating in the suppression pool cooling and shutdown cooling modes at the heat exchanger capacity as determined by flow rates and temperature differentials indicated on the RHR system process flow diagram.

(3) Low-Power Test—Hot Conditions (ESBWR reactor water cleanup shutdown cooling mode): This low power test will demonstrate the ability of the reactor water cleanup (RWCU) nonregenerative heat exchanges to remove decay heat from the nuclear boiling system to place the plant in a safe-shutdown cooling mode of operation.

(a) Verify that the heat removal capacity of the RWCU non-regenerative heat exchangers as determined by flow rates and the temperature differential indicated on the RWCU system process flow diagram.

2. Component Testing

The components of the systems involved in the system tests described in Regulatory Position C.1 should be tested, either in conjunction with the system tests at the appropriate test phase or by independent component tests. Components that are common to the ECCS and other systems should be tested to the more stringent criteria. Performance data should be recorded and the following items verified:

a. Instrumentation

(1) Verify that design acceptance criteria are met for operation of initiating instrumentation in various combinations of logic and instrument channel trip.

(2) Verify that design acceptance criteria are met for functioning of instrumentation and alarms used to monitor system availability. Instruments and alarms should be calibrated and tested before plant startup.

b. Valves

(1) Verify that the design acceptance criteria are met for operation of system valves, including response times with the energy source (e.g., air/nitrogen supply or electric power source) at its limiting design condition. This should include visual verification of valve position as well as proper control room indication.

(2) Verify valve operation under maximum expected differential pressure conditions (consistent with system test limitations).

(3) Verify operability at maximum expected pressure and temperature (consistent with system test limitations).

c. Pumps and Motors

(1) Verify proper operation of injection pumps and motors in all design operating modes.

(2) Verify that design acceptance criteria are met for NPSH performance under maximum system flow and temperature conditions. The test should also verify, by inspection, that no foreign material has entered into the pump, to ensure that performance degradation does not occur, and it should verify that the pump suction strainer is not clogged with debris, so that pump failures or other system degradation does not occur.

(3) Verify that design acceptance criteria are met for individual pump capacity and discharge head.

(4) Verify that design acceptance criteria are met for pump response time (time to reach rated flow conditions) under minimum design voltage and frequency.

(5) Verify that design acceptance criteria are met for pump motor start sequence, overspeed protection, and adequate margins between motor running currents and protective breaker ratings. RG 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," (Ref 7), Regulatory Position C.3.3, "Circuit Analysis," provides additional guidance for the coordination and testing of protective breakers to prevent thermal overload of electrical motors.

(6) Verify that design acceptance criteria are met for vibration levels. Direct contact accelerometers or noncontacting vibration measurement methods are acceptable.

d. Controls

(1) Verify that design acceptance criteria are met for operation of controls, including controls that transfer pump suction. The tests should also verify separately and independently

each channel or bus to identify any failures or losses of redundancy. Testing should include all backup and redundant controls.

(2) Verify that design acceptance criteria are met for the operation of interlocks and equipment protective devices in pump and valve controls.

e. Power Supplies

(1) Verify that design acceptance criteria are met for operation of normal and all alternative electric power supplies used for system valves, pumps, and motors.

(2) Verify that design acceptance criteria are met for operation of automatic and manual power transfer switches.

f. System Piping and Supports

Verify that design acceptance criteria are met for system piping movements under system startup conditions and during steady-state operation. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Sections NB/NC/ND-3620, "Design Considerations," and NB/NC/ND-3622.3, "Vibration" (Ref. 8), provide a robust methodology for testing, monitoring, evaluating, and controlling piping system vibration.

3. Documentation

The initial test program for BWRs should be documented in a summary report and retained as part of the plant historical record. This summary report should include the following:

a. a listing and description of the objectives of each test;

b. a description of how each test was conducted;

c. the parameters monitored;

d. complete comparisons and evaluations against design predictions or system performance requirements for the HPCF, HPCS, HPCI, HPFI, RCIC, ICS, LPCF, LPCS, LPCI, and RHR flow and isolation tests and ADS steam flow tests.

e. any discrepancies or deficiencies noted;

f. system modifications and corrective actions required;

g. appropriate justification for acceptance of systems or components not in conformance with design predictions or performance requirements;

h. any unexpected or unusual conditions during test observations; and

i. conclusions.

Retention of the test procedures, data, and summaries by the licensee should be consistent with paragraph 9 of Appendix C to RG 1.68 (Ref. 3) and in accordance with General Design Criterion 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50 and Criteria XI and XVII, "Quality Assurance Records," of Appendix B to 10 CFR Part 50.

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

---

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 OF ACRONYMS

ABWR—advanced boiling-water reactor

ADAMS—Agencywide Documents Access and Management System

AC—alternating current

ADS—automatic depressurization system

ASME—American Society of Mechanical Engineers

BWR—boiling-water reactor

CFR—*Code of Federal Regulations*

CST—condensate storage tank

DCD—design certification document

ECCS—emergency core cooling system

ESBWR—economic simplified boiling-water reactor

GDCS—gravity driven cooling system (ESBWR only)

HPCF—high-pressure core flooder (ABWR only)

HPCI—high-pressure coolant injection

HPCS—high-pressure core spray

HPFI—high-pressure feedwater injection

IAEA—international atomic energy agency

IC—inside containment

IC/PCCS – inside containment/passive containment cooling system

I&C—instrumentation and control

ICS—isolation condenser system

LOCA—loss-of-coolant accident

LPCF—low-pressure core flooder (ABWR only)

NPSH—net positive suction head

NRC—U.S. Nuclear Regulatory Commission

OMB—Office of Management and Budget

PCCS—passive containment cooling system (ESBWR only)

PWR—pressurized water reactor

RCIC—reactor core isolation cooling

RG—regulatory guide

RHR—residual heat removal

RWCU—reactor water cleanup system

SRV—safety relief valve

SSLC—safety system logic and control

SLCS—standby liquid control system

SSC—structure, system, and component

The following list of additional acronyms is used in Appendix A of this guide including acronyms used in Figures A-1 through A-11:

AO—air operated

ATWS—anticipated transient without scram

DIV—division

DPV—depressurization valves

EH— electro hydraulic

ESF— engineered safety feature

FE—flow element

FP —fuel pool

FPC—fuel pool cooling

FPCU—fuel pool cooling and cleanup

FCS—flammability control system

HPCF—high pressure core flooder system (ABWR only)

HX— heat exchanger

IC/PCCS— inside containment/passive containment cooling system

LPCF—low pressure core flooder system (ABWR only)

M— motor

MO—motor operated

MS— main steam

MUCW—makeup cooling water

NBS — nuclear boiler system

NNS—non nuclear system

P— pressure

RCIC— reactor core isolation cooing system

RCS—reactor coolant system

RCW—reactor building cooling water system

RHR—residual heat removal system

RPV— **r**eactor pressure vessel

R—redundant

SLC—standby liquid control

SO— solenoid operated

S/P— suppression pool

SPCU—suppression pool cleanup system

T—temperature

Note: Some acronyms may be defined more than once in this guide since they were used with slightly different vendor terms.

REFERENCES[3]

1. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission, Washington, D.C.

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,", U.S. Nuclear Regulatory Commission, Washington, DC.

4. Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors," U.S Nuclear Regulatory Commission, Washington, D.C.

5. IAEA Safety Standard Series Safety Guide 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]

6. Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission, Washington, D.C.

7. Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, D.C.

8. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III[5], Sections NB/NC/ND-3620, "Design Considerations," and NB/NC/ND-3622.3, "Vibration" 2010 (with 2011 Addendum).

---

[3] Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at: http://www.nrc.gov/reading-rm/doc-collections/. The documents can also be viewed on-line 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

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

BIBLIOGRAPHY

U.S. Nuclear Regulatory Commission Documents

Generic Letters

GL 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," December 3, 1985. (ADAMS Accession No. ML031150731)

GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," July 14, 1998. (ADAMS Accession No. ML031110081)

GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors, September 13, 2004." (ADAMS Accession No. ML042360586)

GL 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems," January 11, 2008. (ADAMS Accession No. ML072910759)

Information Notices

IN 92-85, "Potential Failures of Emergency Core Cooling Systems Caused by Foreign Material Blockage," December 23, 1992. (ADAMS Accession No. ML031190717)

IN 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," April 26, 1993 (ADAMS Accession No. ML031070498)

IN 93-34, Supplement 1, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," May 6, 1993. (ADAMS Accession No. ML031210149)

IN 94-57, "Debris in Containment and the Residual Heat Removal System," August 12, 1994. (ADAMS Accession No. ML031060503)

IN 96-10, "Potential Blockage by Debris of Safety System Piping Which Is Not Used During Normal Operation or Tested during Surveillances," February 13, 1996. (ADAMS Accession No. ML031060270)

IN 97-76, "Degraded Throttle Valves in Emergency Core Cooling System Resulting from Cavitation-Induced Erosion during a Loss-of-Coolant Accident," October 30, 1997. (ADAMS Accession No. ML031050058)

IN 2006-20, "Foreign Material Found in the Emergency Core Cooling System," October 16, 2006. (ADAMS Accession No. ML062440467)

IN 2006-21, "Operating Experience Regarding Entrainment of Air into Emergency Core Cooling and Containment Spray Systems," September 21, 2006. (ADAMS Accession No. ML062570468)

Bulletins

Bulletin 93-02, Supplement 1, “Debris Plugging of Emergency Core Cooling Suction Strainers,” May 11, 1993. (ADAMS Accession No. ML031190684)

Bulletin 96-03, “Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors,” May 6, 1996. (ADAMS Accession No. ML993410152)

Licensee Event Reports

LER 50-333/2004-001, “Inadvertent Actuation of Emergency Core Cooling Systems and Emergency Diesel Generators While in Refueling Mode,” January 19, 2005. (ADAMS Accession No. ML050270178)

BWR Technical Specification Amendments Related to ECCS

James A. Fitzpatrick Nuclear Power Plant, Improved Standard Technical Specification (ITS) Conversion, “Emergency Core Cooling System (ECCS) Instrumentation,” Justification for Differences from NUREG-1433, Revision 1, April 7, 1995. (ADAMS Accession No. ML011640094)

James A. Fitzpatrick Nuclear Power Plant Improved Standard Technical Specification (ITS) Conversion, “ECCS – Shutdown, Discussion of Changes to CTS.” April 7, 1995 (ADAMS Accession No. ML011630179)

LaSalle County Station Units 1 and 2, Application for Amendment to Appendix A Technical Specification Section 3/4.5.1, “ECCS Operating,” Action C, “RCIC Declared Operable when Reactor Steam Dome Pressure Greater than 150 psig,” April 12, 2000, (ADAMS Accession No. ML003704192)

Cooper Nuclear Station, Revise Technical Specification 3.5.1 to Incorporate Technical Specification Task Force (TSTF) 318 for One Low Pressure Coolant Injection (LPCI) Pump Inoperable in Each of the Two ECCS Divisions, August 25, 2003 (ADAMS Accession No. ML032450233)

Columbia Generating Station, Revised TS 3.3.5.2, “RCIC System Instrumentation,” and TS 3.5.2, “ECCS Shutdown to Increase Storage Tank Level,” Amendment 210, September 30, 2008 (ADAMS Accession No.ML082610056)

APPENDIX A

DESIGN DESCRIPTIONS OF EMERGENCY CORE COOLING SYSTEMS

A.1. Emergency Core Cooling System for the Current Boiling-Water Reactor Fleet of Plants

The boiling-water reactor (BWR)/2-6 emergency core cooling system (ECCS) typically includes a high-pressure coolant injection (HPCI) or high-pressure core spray (HPCS) subsystem, one or more low-pressure coolant injection (LPCI) and/or low-pressure core spray (LPCS) subsystems, and an automatic depressurization system (ADS). The HPCI or HPCS subsystem is intended to respond to small-break loss-of-coolant accidents (LOCAs), while the LPCI and/or LPCS subsystems are intended to respond to large-break LOCAs that rapidly depressurize the reactor coolant system (RCS). The ADS is used to depressurize the RCS when the HPCI or HPCS subsystem fails to perform adequately, thereby allowing the LPCI or LPCS subsystems to provide core cooling. The BWRs also use either an isolation condenser system (ICS) or a reactor core isolation cooling (RCIC) system when feedwater is isolated from the RCS and ICS or RCIC is needed for hot shutdown cooling to cold shutdown. The LPCI mode of residual heat removal (RHR) is then used to cool the reactor to cold shutdown. For additional details, see NUREG/CR-5640, "Overview and Comparison of U.S. Commercial Nuclear Power Plants," dated September 1990. Specifically, see Section 8.9, "Emergency Core Cooling Systems," for BWRs.

The BWR ECCS meets acceptance criteria that are based on meeting the relevant requirements of Title 10 of the *Code of Federal Regulations* General Design Criteria 4, 5, 33, 34, 35, 36, 37 and 55 of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." In addition, the ICS and RCIC support functions that meet alternate water injection during station blackout to meet 10 CFR 50.63, "Loss of All Alternating Current Power."

A.2 Advanced Boiling-Water Reactor

ECCSs for the advanced boiling-water reactor (ABWR) consist of the following:

- the low-pressure core flooder (LPCF) mode of RHR,
- the high-pressure core flooder (HPCF),
- RCIC, and
- the ADS subsystem of the nuclear boiler system (NBS).

Low-Pressure Core Flooder (LPCF) Mode of Residual Heat Removal

Figures A-1 to A-3 show the three-train system. Each of the three RHR divisions can be initiated manually (LPCF mode). The RHR system channel measurements are provided to the safety system logic and control (SSLC) for signal processing, setpoint comparisons, and generation of trip signals. The RHR system is automatically initiated when either a high drywell pressure or a low reactor water level condition exists (i.e., a LOCA signal). An RHR initiation signal is provided to the systems. The SSLC processors use a two-out-of-four voting logic for RHR system initiation.

After receipt of an initiation signal, the RHR system automatically initiates and operates in the low-pressure core flooder (LPCF) mode to provide emergency makeup to the reactor vessel. The initiation signal starts the pumps, which run in the minimum flow mode until the reactor depressurizes to less than the pump's developed head pressure. A low-reactor-pressure permissive signal occurs above the pump's developed head pressure, which signals the injection valve to open. As the injection valve opens, the testable check valve contains the reactor pressure until the pressure becomes less than the pump's developed head pressure in the minimum flow mode, at which time injection flow begins. This sequence satisfies the response requirements for all potential LOCA pipe breaks when the injection valve opens within seconds after receiving the low-reactor-pressure permissive signal. The LPCF injection flow for each division begins when the reactor vessel pressure is no less than 1.55 megapascals (MPa) (225 psig) above the drywell pressure. When the reactor vessel pressure is no less than 0.275 MPa (40 psig) greater than the drywell pressure, the LPCF injection flow for each division is 954 cubic meters per hour, minimum. Since these are ABWR design certification document (DCD) values, these pressures may vary for each individual plant design. All three divisions of the RHR system accomplish the LPCF mode by transferring water from the suppression pool to the reactor pressure vessel (RPV) via the RHR heat exchangers. The system automatically aligns to the LPCF mode of operation from the test mode, the suppression pool cooling, or wetwell spray mode upon receipt of an initiation signal. The wetwell spray mode is applicable for division B or C. If a drywell spray valve is open in division B or C, that division automatically aligns to the LPCF mode in response to the injection valve beginning to open. The RPV injection valve in each division requires a low-reactor-pressure permissive signal to open and closes automatically upon receipt of a high-reactor-vessel-pressure signal.

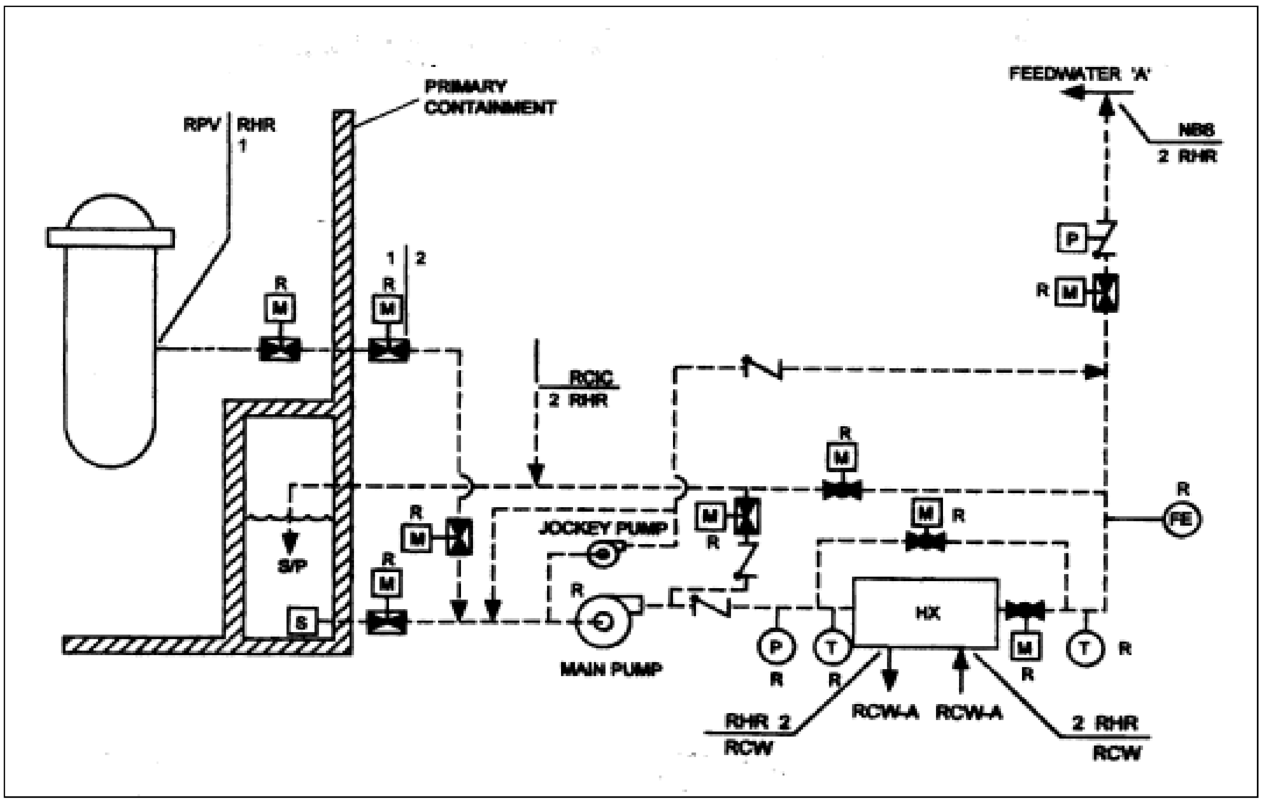

Figure A-1. ABWR low-pressure core flooder mode of residual heat removal—train A

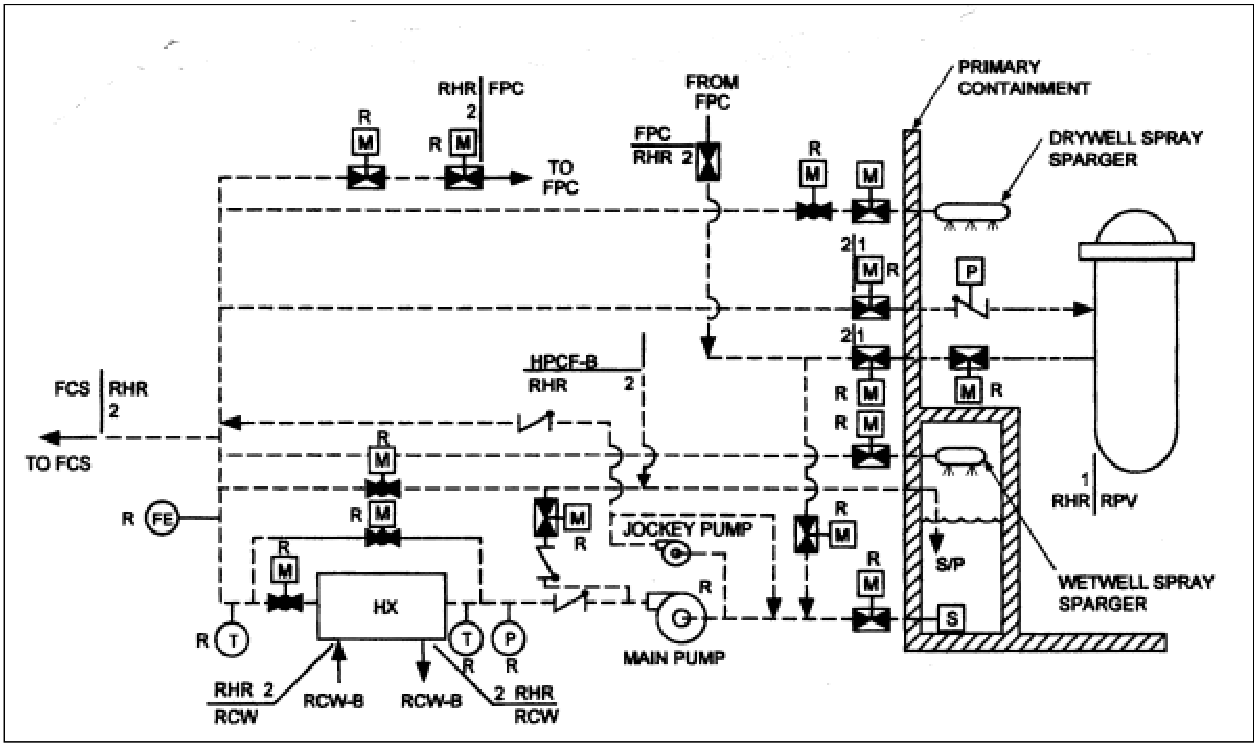

Figure A-2. ABWR low-pressure core flooder mode of residual heat removal—train B

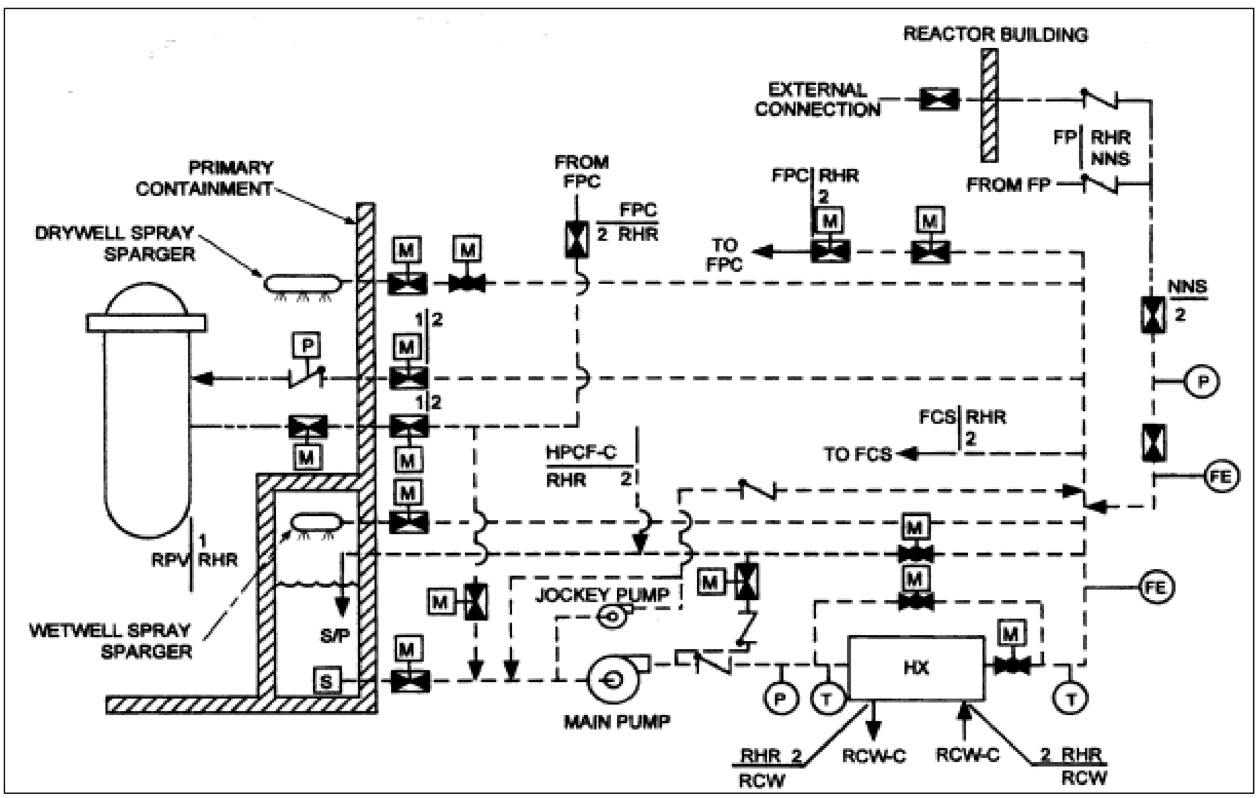

Figure A-3. ABWR low-pressure core flooder mode of residual heat removal—train C

High-Pressure Core Flooder

The HPCF system (Figure A-4) consists of two HPCF loops (B and C) flooding water to the RPV above the core. Each of the two loops belongs to a separate division; electrical and mechanical separation between the two divisions is complete. Locating each division in a different area of the reactor building ensures physical separation. The two loops are both high-pressure pumping systems (i.e., they are capable of injecting water into the reactor vessel over the entire operating pressure range). The reference pressure for the operating performance of the system at high pressure is the lowest spring (safety) setpoint of the safety relief valves (SRVs).

Both HPCF divisions take primary suction from the condensate storage tank (CST) and secondary suction from the suppression pool. In the event that the CST water level falls below a predetermined setpoint or the suppression pool water level rises above a predetermined setpoint, the pump suction will automatically transfer from the CST to the suppression pool. Both HPCF system loops have suction lines that are separate from the RHR loops.

The HPCF pumps are at an elevation below the water level in the suppression pool. This ensures flooded pump suction conditions. The motor-operated valve in the suction line from the suppression pool on each division is normally closed because primary suction is taken from the CST. This valve automatically opens upon receipt of either of the suction transfer signals noted above. The suppression pool suction valves on each loop can be closed from the control room if a leak develops in the system piping downstream of the isolation valves.

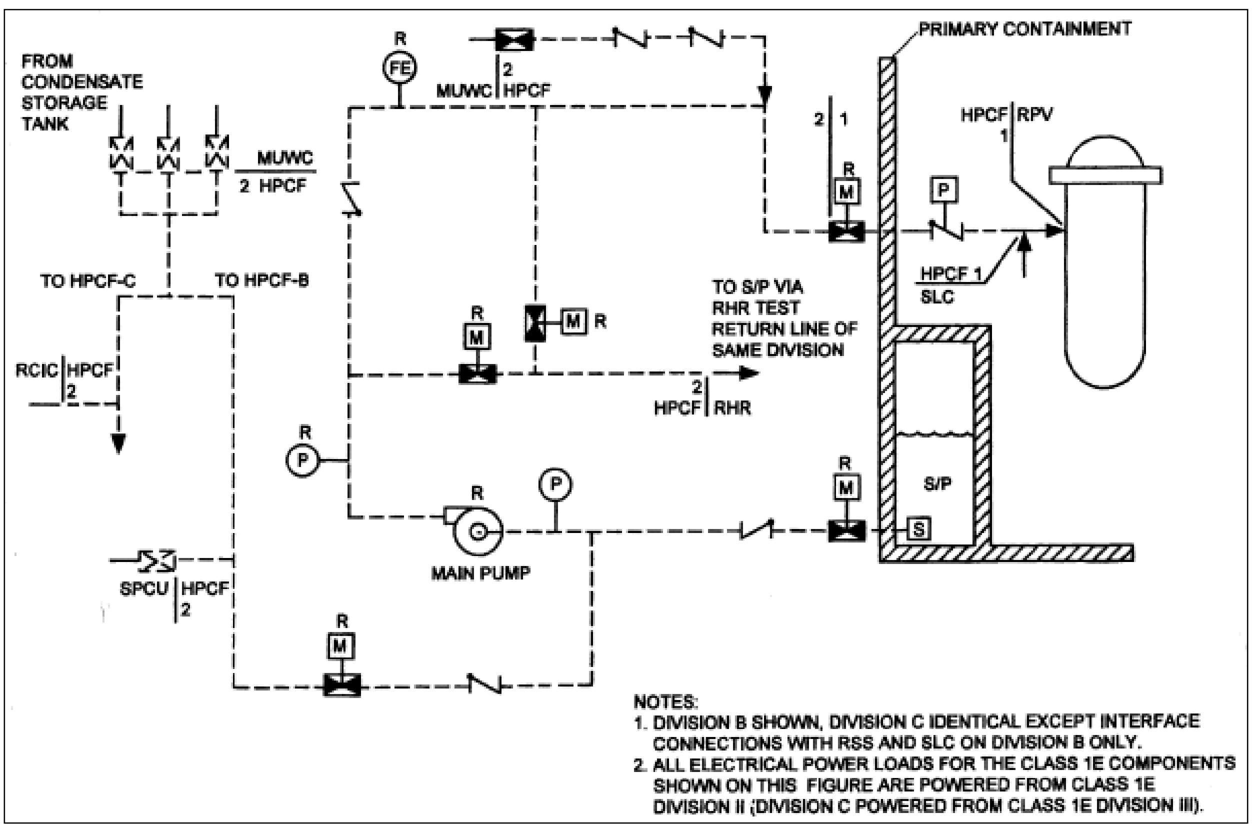

Figure A-4. ABWR high-pressure core flooder

The RHR system is a closed system consisting of three independent pump loops that inject water into the vessel and/or remove heat from the reactor core or containment. Each of the pump loops contains the necessary piping, pump valves, and heat exchangers. In the core cooling mode, each loop draws water from the suppression pool and injects the water into the vessel outside the core shroud (via the feedwater line on one loop and via the core cooling subsystem discharge return line on two loops). In the heat removal mode, pump suction can be taken from either the suppression pool or the RPV.

With the pump suction taken from the suppression pool, the pump discharge within these loops provides a flowpath to the following points:

- the RPV (via feedwater on one loop and via the core cooling subsystem return lines on the other two loops) and
- the wetwell and drywell spray spargers (on two loops only).

With the pump suction taken from the RPV via the shutdown cooling lines (in the shutdown cooling mode), the pump discharge in these loops provides a flowpath back to the following points:

- the reactor vessel (via the core cooling discharge return lines and feedwater line) and
- the upper reactor well, via the fuel cooling system (on two loops only).

Reactor Core Isolation Cooling

The purpose of the RCIC system (Figure A-5) is to supply high-pressure makeup water to the RPV when the reactor is isolated from the main condenser and the condensate and feedwater system is not available. The system is started automatically upon receipt of a low-water-level signal or manually by the operator. Water is pumped to the core by a turbine pump driven by reactor steam.

The functional classification of the RCIC system is as a safety-related and engineered safety feature (ESF) system. For the ABWR design, the RCIC system is considered to be part of the ECCS. For earlier BWR designs, RCIC was not considered to be part of the ECCS. However, the RCIC system is considered an ESF system because of its role in mitigating the consequences of a control rod drop accident. If a rod drop accident occurs, it is possible that the main steamlines might isolate on a high radiation signal. The RCIC system then performs its normal isolation cooling function. The RCIC system function is completely backed up by HPCF, HPCS, or HPCI, depending on the BWR design (BWR 3-6, ABWR).

The RCIC system consists of a steam-driven pump with associated valves and piping capable of delivering water flow to the RPV. The turbine is driven by steam produced from decay heat. The RCIC system takes water from the CST or the suppression pool and delivers it to the RPV to maintain adequate RPV level. The turbine exhaust is directed to the suppression pool, where it is condensed. If the condensate and feedwater system is isolated from the RPV, the RCIC system will start automatically when decay heat boils coolant to low reactor water level (the Level 2 initial signal). The RCIC system supplies sufficient inventory to allow complete shutdown without compromising fuel clad integrity.

The RCIC system is also used in conjunction with the RHR system in the steam condensing mode to pump condensate from the RHR heat exchangers back to the RPV. The RCIC system also has alternate paths to allow recirculation back to the CST for testing purposes, discharge to the suppression pool to ensure minimum flow through the pump, and recirculation for turbine lube oil cooling.

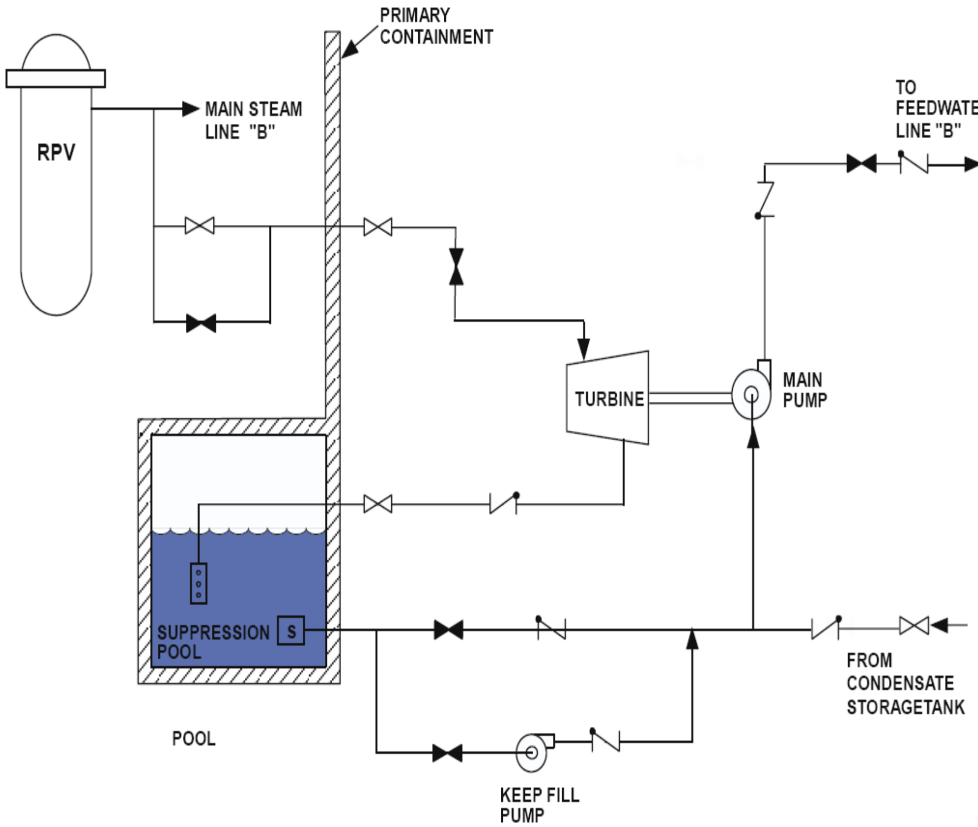

Figure A-5. BWR3-6, ABWR reactor core isolation cooling

__Automatic Depressurization System__

If the RCIC and HPCF systems cannot maintain the reactor water level, the ADS (Figure A-6), which is independent of any other ECCS, reduces the reactor pressure so that flow from the RHR system operating in the low-pressure flooder mode enters the reactor vessel in time to cool the core and limit fuel cladding temperature. The ADS employs nuclear SRVs to relieve high-pressure steam to the suppression pool.

The NBS channel measurements are provided for the safety-related instrumentation and control system SSLC for signal processing, setpoint comparisons, and generation of trip signals. Except for the pump running interlock, the SSLC uses a two-out-of-four voting logic for ADS initiation. The ADS logic is automatically initiated when a low-reactor-water-level signal is present. If the RPV low-water-level signal is present concurrent with a high-drywell-pressure signal, both the main ADS timer (less than or equal to 29 seconds) and the high-drywell-pressure bypass timer (less than or equal to 8 minutes) are initiated. Since these are ABWR DCD values, the ADS timers may vary for each individual plant design. Absent a concurrent high-drywell-pressure signal, only the ADS high-drywell-pressure bypass timer is initiated. On the timeout of the ADS high-drywell-pressure bypass timer, concurrent with the RPV low-water-level signal, the main ADS timer is initiated, if not already initiated. The main timer continues to completion and times out only in the continued presence of an RPV low-water-level signal. On timeout of the main ADS timer, concurrent with positive indication by pump discharge pressure of at least one RHR or one HPCF pump running, the ADS function is initiated.

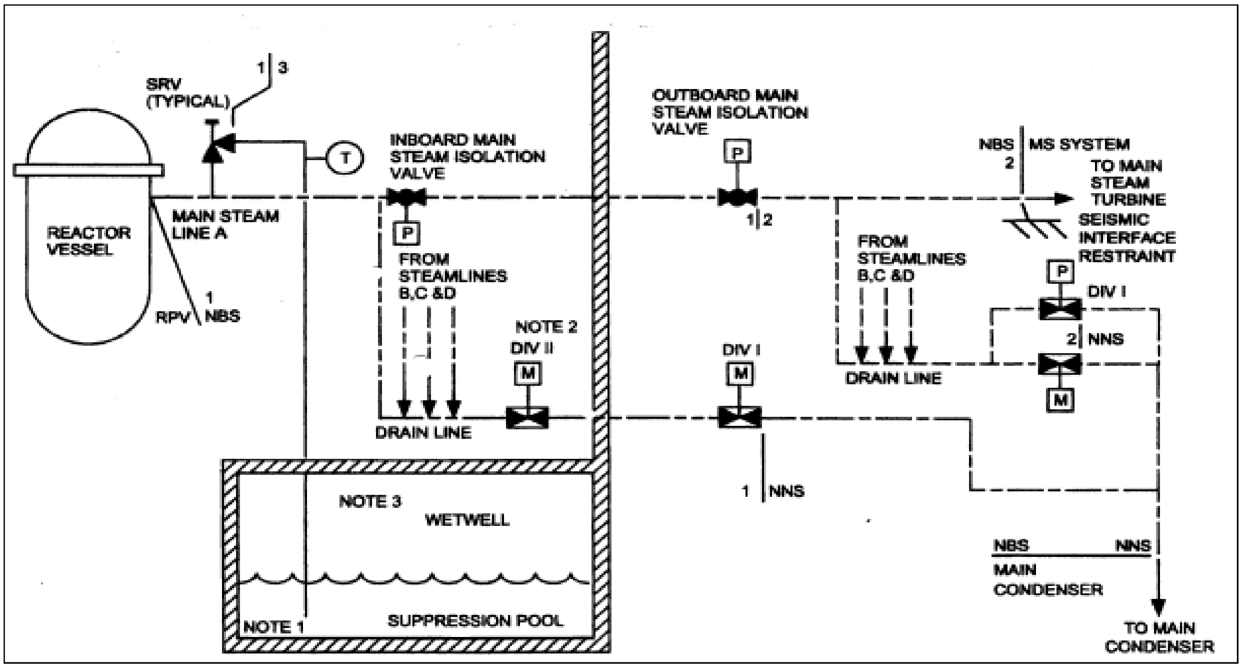

Figure A-6. ABWR automatic depressurization system

Signals from all four divisions for low reactor water level and high drywell pressure and the Division I control logic signal actuate one set of pilots, and sensors from all four divisions for low reactor water and high drywell pressure and the division II control logic signal actuate the second set of pilots, either of which initiates the opening of the ADS SRVs.

Redundant trip channels arranged in two divisionally separated logics that control two separate solenoid-operated pneumatic pilots on each ADS SRV accomplish ADS initiation. Either pilot can operate the ADS valve. These pilots control the pneumatic pressure applied by the accumulators and the high-pressure nitrogen gas supply system. The direct-current power for the logic is obtained from SSLC divisions I and II.

For mitigation of anticipated transient without scram (ATWS), the ADS has an automatic and manual inhibit of the automatic initiation. Automatic initiation of the ADS is inhibited unless there is a coincident low-reactor-water-level signal, and an average power range monitors the ATWS permissive signal from the neutron monitoring system. There are main control room switches for the manual inhibit of automatic initiation of the ADS.

The ADS can be initiated manually. On a manual initiation signal, concurrent with positive indication that at least one RHR or one HPCF pump is running, the ADS function is initiated.

The ADS automatically actuates in response to the ECCS initiation signals. A two-out-of-four-level initiation logic is used to activate the SRVs and depressurization valves (DPVs). The 10-second delay to confirm the level initiation signal ensures that momentary system perturbations do not actuate the ADS when it is not required. The two-out-of-four logic ensures that a single failure does not cause spurious system actuation and that a single failure cannot prevent initiation.

A.3 Economic Simplified Boiling-Water Reactor Systems

ECCSs for the economic simplified boiling-water reactor (ESBWR) consist of the following:

- the ICS,
- the gravity-driven cooling system (GDCS),
- the ADS, and
- the standby liquid control system (SLCS).

<u>Isolation Condenser System</u>

The ESBWR ICS (Figure A-7) is the system most comparable to the BWR RCIC system. The ESBWR is a passive plant relying almost exclusively on natural recirculation to drive plant flow, which differs significantly from the BWR RCIC, which relies heavily on active systems to accomplish its functions.

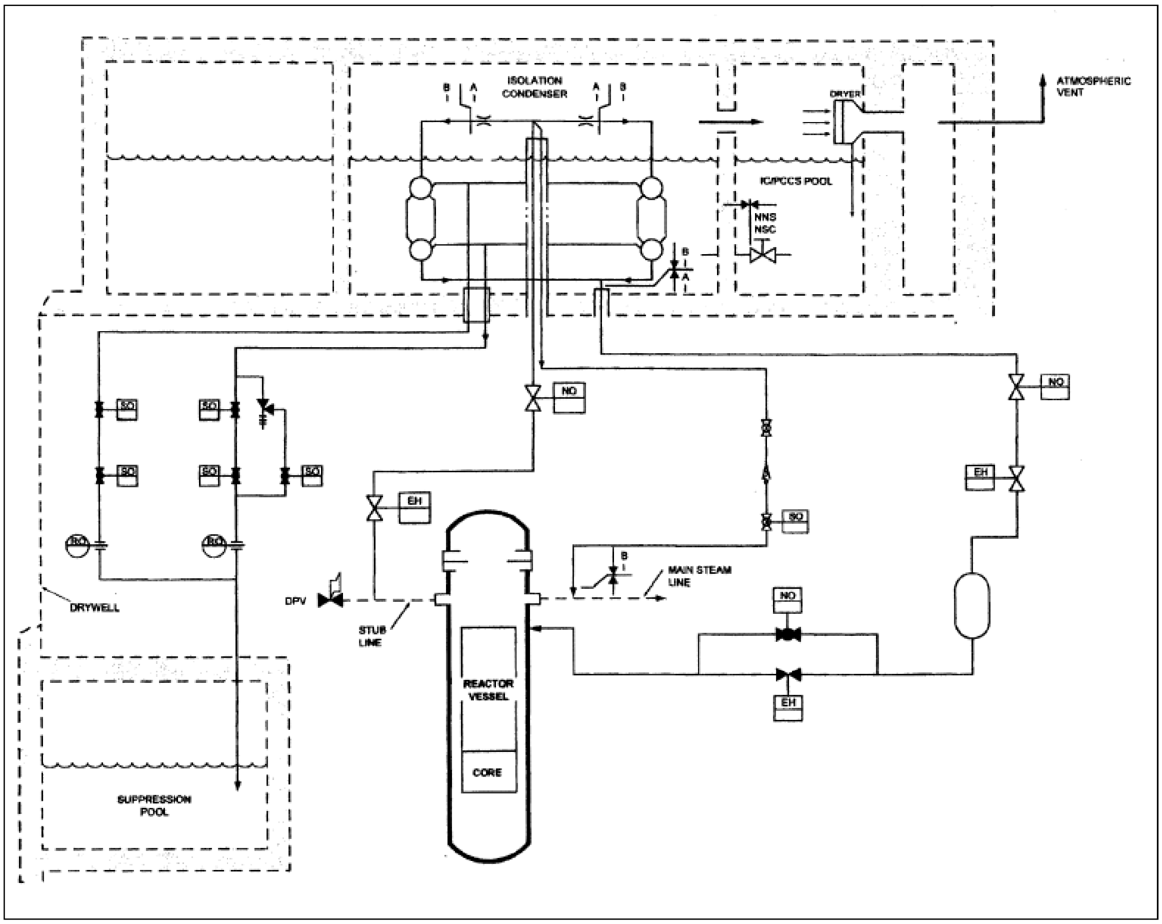

Figure A-7. ESBWR isolation condenser system

The ESBWR passive decay heat removal systems (isolation condensers) are capable of achieving and maintaining safe, stable conditions for at least 72 hours without operator action after non-LOCA events. Operator action is credited after 72 hours to refill isolation condenser pools or initiate nonsafety shutdown cooling.

The ICS removes residual sensible and core decay heat from the reactor, in a passive way and with minimal loss of coolant inventory from the reactor, when the normal heat removal system is unavailable after any of the following events:

- station blackout (i.e., unavailability of all alternating-current power),
- ATWS, and
- LOCA.

The ICS functions to avoid unnecessary use of other ESFs for RHR and in the event of a LOCA. The ICS also provides additional liquid inventory upon opening of the condensate return valves to initiate the system. In the event of ICS initiation by a reactor level below Level 2, the ICS removes core heat, causing initial depressurization of the reactor before the ADS initiates. Because of this vessel pressure reduction with return of condensed steam plus the additional initial ICS stored condensate inventory, the ADS can initiate from a lower reactor water level to complete the vessel depressurization.

The ICS is designed as a safety-related system to remove reactor decay heat after reactor shutdown and isolation. It also prevents unnecessary reactor depressurization and operation of other ESFs, which can also perform this function.

In the event of a LOCA, the ICS provides additional liquid inventory upon opening of the condensate return valves to initiate the system. The ICS also provides initial depressurization of the reactor before ADS in the event of loss of feedwater, such that the ADS can take place from a lower water level.

To ensure that an adequate inventory of cooling water is available for at least 72 hours after an accident, the ICS uses automatically opening connections between the equipment storage pool and isolation condenser/passive containment cooling system pools.

Gravity-Driven Cooling System

The GDCS (Figure A-8) provides emergency core cooling after an event that threatens the reactor coolant inventory. Once the reactor has been depressurized, the GDCS is capable of injecting large volumes of water into the depressurized RPV to keep the core covered for at least 72 hours after a LOCA.

The GDCS also drains its pools to the lower drywell in the event of a core melt sequence that causes failure of the lower vessel head and allows the molten fuel to reach the lower drywell cavity floor. This action is accomplished by detection of elevated temperatures registered by thermocouples in the lower drywell cavity, and by logic circuits that actuate squib-type valves on independent pipelines draining GDCS pool water to the lower drywell region. Because inadvertent actuation of the automatic logic circuits could result in loss of GDCS pool inventory and the consequent unavailability of water for injection into the reactor vessel on a valid GDCS actuation signal, a set of safety-related temperature switches are used to inhibit deluge actuation as long as the drywell temperature is less than a preset value.

The GDCS requires no external electric power source or operator intervention. The GDCS initiation signal is the receipt of a confirmed ECCS initiation signal from the NBS. This signal initiates ADS and GDCS injection valve timers as well as longer equalization valve timers in the GDCS logic.

After injection valve timer duration, squib valves are activated in each of the injection lines leading from the GDCS pools to the RPV, making GDCS flow possible. The actual GDCS flow delivered to the RPV is a function of the differential pressure between the reactor and the GDCS injection nozzles. The loss of the GDCS provides short-term post-LOCA water makeup to the annulus region of the reactor through eight injection line nozzles, and by gravity-driven flow from three separate water pools in the drywell at an elevation above the active core region. The system provides long-term post-LOCA water makeup to the annulus region of the reactor through four equalization nozzles and lines connecting the suppression pool to the RPV. During severe accidents (i.e., if the core melts through the RPV), the GDCS floods the lower drywell region directly via four GDCS injection drain lines (one each from two pools and two from the third pool) through a deluge system.

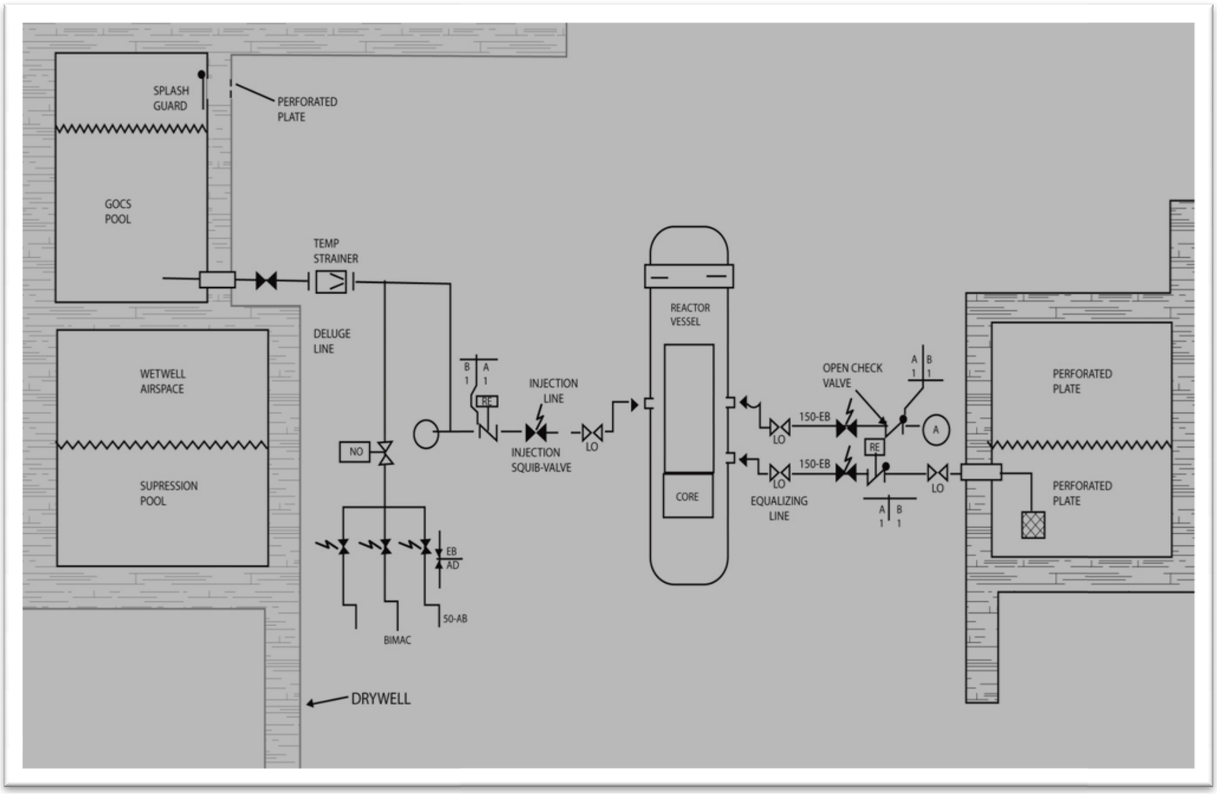

Figure A-8. ESBWR gravity-driven cooling system

Automatic Depressurization System

The ADS (Figures A-9 and A-10) is a part of the ECCS. It operates to depressurize the reactor so that the low-pressure GDCS is able to make up coolant to the reactor. The ADS is a function of the NBS. The depressurization function is accomplished using SRVs and DPVs.

The ADS automatically actuates in response to the ECCS initiation signals. A two-out-of-four-level initiation logic is used to activate the SRVs and DPVs. The 10-second delay to confirm level initiation signal ensures that momentary system perturbations do not actuate the ADS when it is not required. The two-out-of-four logic ensures that a single failure does not cause spurious system actuation and that a single failure cannot prevent initiation.

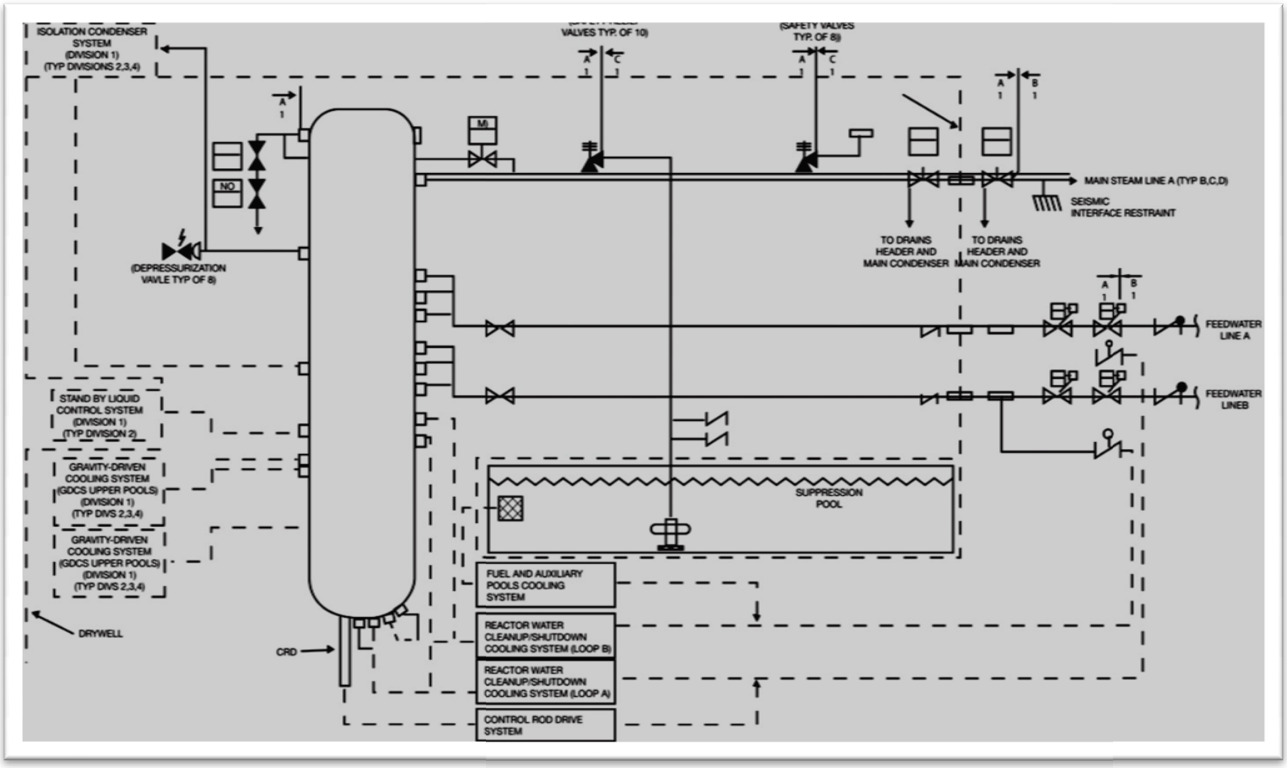

Figure A-9. ESBWR automatic depressurization system

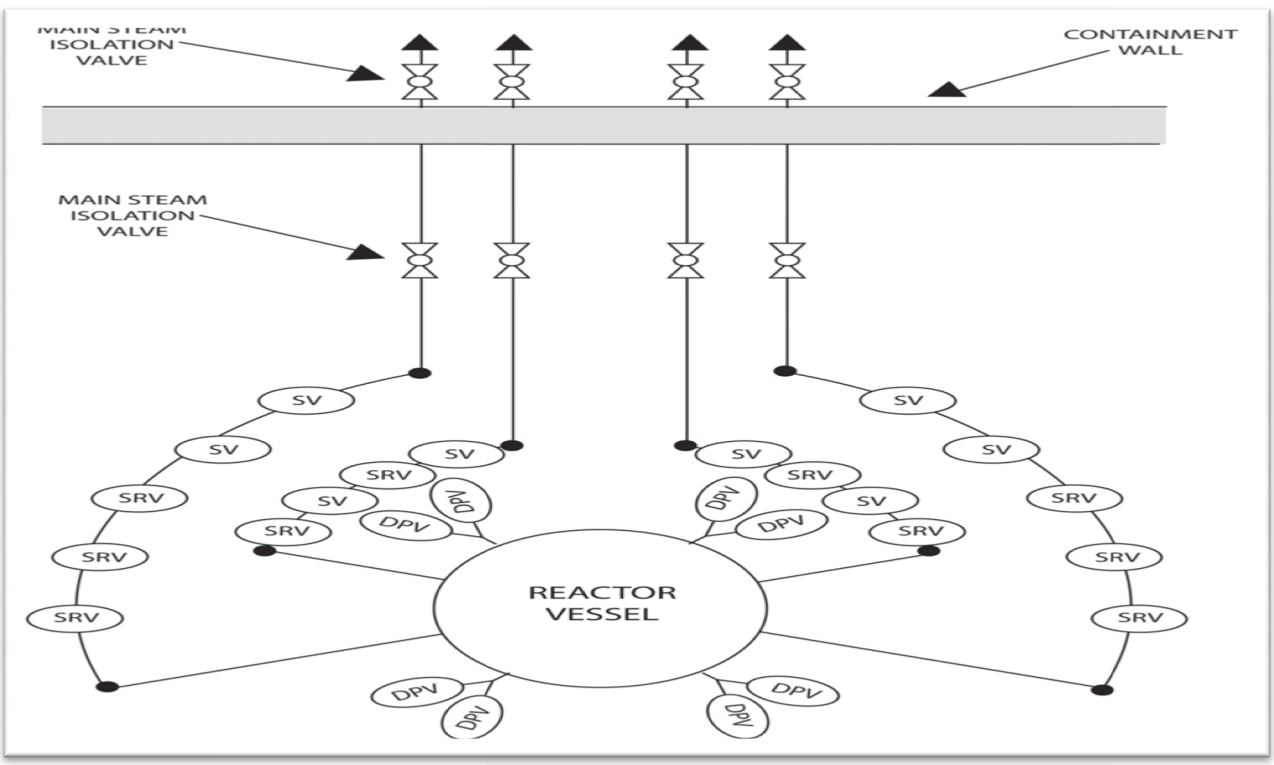

Figure A-10. ESBWR automatic depressurization system

Use of Dual-Function Components

The ECCS ADS and GDCS are designed to accomplish only one function, to cool the reactor core after a LOCA. The ECCS SLCS is designed for use during an ATWS, and the ECCS ICS is designed to avoid unnecessary use of other ESFs for RHR. Both the SLCS and the ICS provide additional liquid inventory on actuation. To this extent, components or portions of these systems, except for the pressure relief function of SRVs, are not required for operation of other systems. Because the SRV opens either on an ADS initiating signal or by spring-actuated pressure relief in response to an overpressure condition, no conflict exists.

Standby Liquid Control System

The shutdown function of the SLCS (Figure A-11) is manually initiated. In addition, the system is automatically initiated for ATWS and LOCA events.

The SLCS contains two identical and separate trains. Each train provides 50-percent injection capacity. All components of the SLCS in contact with the boron solution are constructed of or lined with stainless steel. The safety-related portions of the SLCS are listed in the ESBWR design control document. The SLCS requires support from safety-related interfaces.

The SLCS includes a nitrogen charging subsystem that consists of a liquid nitrogen tank, a vaporizer, a high-pressure pump, and associated valves and piping. This subsystem is used for initial accumulator charging and makeup for normal system losses during normal plant operations. It is a nonsafety-related subsystem (but it has safety-related piping and valves) inside the reactor building, extending from the makeup valves downstream to the accumulators. The nonsafety-related high-pressure cryogenic nitrogen equipment is outside the reactor building at grade elevation.

The core bypass spargers are in the reactor vessel and penetrate through the shroud to the core. The portions of the standby liquid control injection line downstream of each squib valve contain only stagnant reactor water. The major components of the SLCS, which are necessary for injection of sodium pentaborate solution (neutron absorber) into the reactor, are in the reactor building. Reactor building heating, ventilation, and air conditioning controls the temperature and humidity conditions in the SLCS equipment rooms to prevent solute precipitation in the accumulators and injection lines, thereby ensuring proper system operation. This system readiness function is nonsafety-related.

The SLCS performs safety-related functions; therefore, it is classified as safety-related and is designed as a seismic Category I system. The SLCS meets the following safety design bases:

- Provides a diverse backup capability for reactor shutdown that is independent of normal reactor shutdown provisions. The SLCS provides makeup water to the RPV to mitigate the consequences of a LOCA. The sodium pentaborate in the SLCS solution is credited for buffering to ensure that the iodine chemical distribution assumed in the LOCA dose consequence analysis remains valid.

- Injects a boron solution, which performs as a neutron absorber, at multiple locations into the core bypass region at high velocity. This ensures adequate mixing and total injection of the solution to accomplish reactor shutdown. The injection geometry ensures balancing of reaction forces.

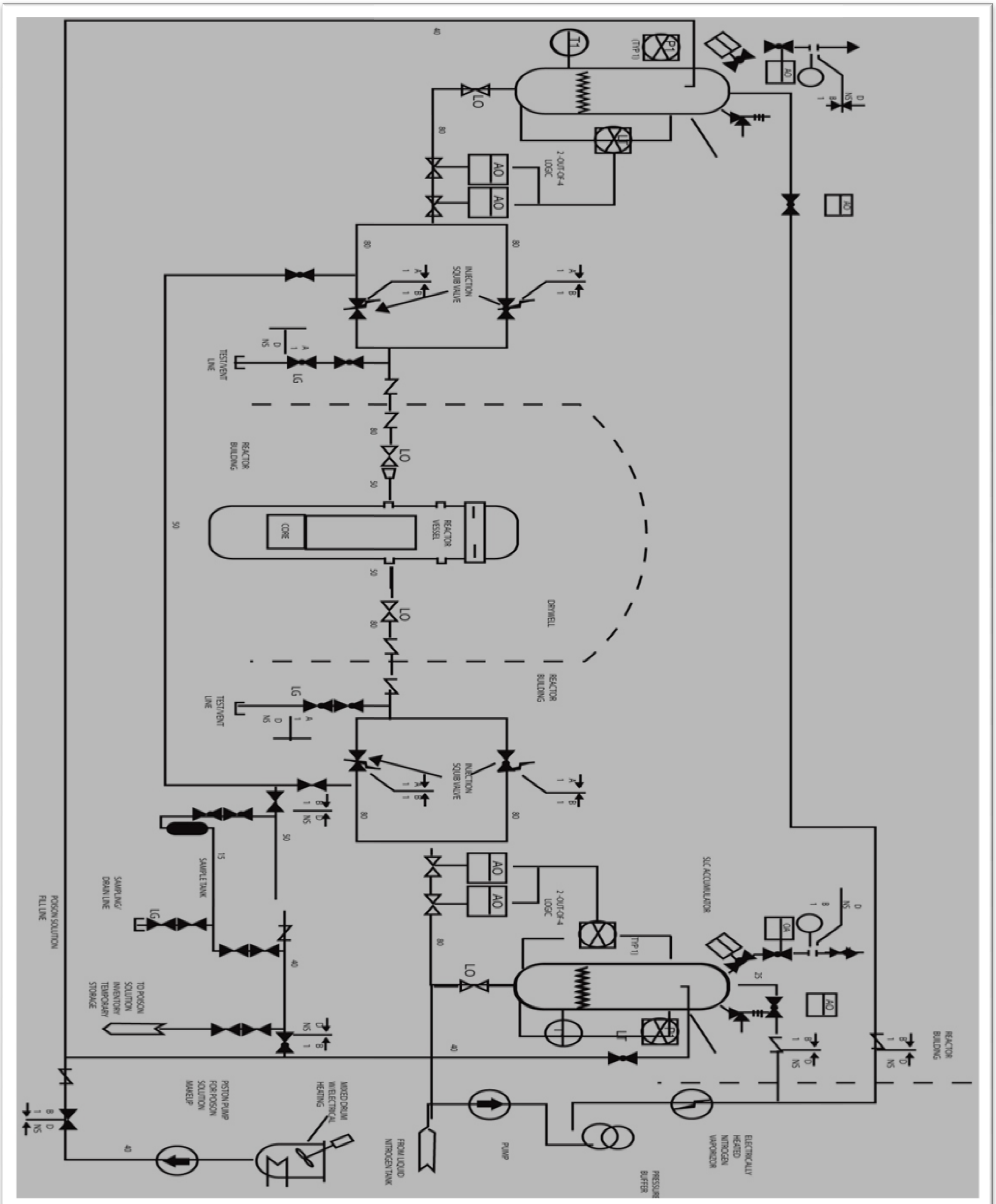

Figure A-11. ESBWR standby liquid control system