19K PRA-Based Reliability and Maintenance

19K.1 Introduction

In this appendix, the results of the PRA are reviewed to determine the appropriate reliability and maintenance actions that should be considered throughout the life of an ABWR plant so that the PRA remains an adequate basis for quantifying plant safety. These actions comprise a part of the plant's reliability assurance program (RAP).

Paragraph 8.8, "Maintenance and Surveillance", of the ABWR Licensing Review Bases (Reference 19K-1), reads in part, "GE is to provide in the SSAR the reliability and maintenance criteria that a future applicant must satisfy to ensure that the safety of the as-built facility will continue to be accurately described by the certified design." This appendix provides the PRA based reliability and maintenance actions which should be considered for incorporation into the future applicant's (i.e., the applicant referencing the ABWR design) operating and maintenance procedures required by Standard Review Plan (SRP) Subsection 13.5.2. As indicated in Table 1.8-19, SRP 13.5.2 is an interface requirement to be provided by the utility applicant referencing the ABWR design.

19K.2 General Approach

To determine the appropriate reliability and maintenance-related activities that should be considered to assure that plant safety is maintained as operation proceeds, results of PRA and other analyses were reviewed. The objective of the review was to determine the relative importance of prevention and mitigation features of the ABWR in satisfying the key PRA goals related to core damage frequency (CDF) and frequency of offsite release. Also considered were the initiating events that had significant impact on CDF. From this review (Subsection 19K.3), the most important plant features were identified.

The PRA was further reviewed (Subsections 19K.4 through 19K.10) for other important features, the failure of which was not addressed directly in Subsection 19K.3, to supplement the above list. Finally (Subsection 19K.11), the individual features identified in Subsections 19K.3 through 19K.10 were reviewed to determine appropriate maintenance and surveillance actions.

19K.3 Determination of "Important Structures, Systems and Components" for Level 1 Analysis

To determine which plant structures, systems and components (SSCs) are the most important with respect to CDF, the Level 1 analysis results were analyzed. The SSCs were listed in order of Fussell-Vesely (FV) importance, or the percent of cutsets that contribute to the CDF, as calculated by the CAFTA code. A second criterion for selecting SSCs was to consider those SSCs with high "risk achievement worth" (RAW), or the increase in CDF if that SSC always fails. The top SSCs, ranked by FV importance greater than 0.5% in value, are shown in Table 19K-1. Not shown in Table 19K-1 are several human error contributions. Significant human errors are addressed in Subsection 19D.7 of Reference 19K-4.

The SSCs in Table 19K-1 were further evaluated to eliminate those with a combination of low values for both FV importance and risk achievement worth. A value of 2.0 for RAW was considered to be low. The SSCs meeting this criterion are so indicated.

The remaining SSCs of Table 19K-1 should be included with important SSCs being considered for periodic testing and/or preventive maintenance (PM) as part of the Reliability Assurance Program (RAP) of the plant owner/operator. The reliability and maintenance actions suggested for the listed SSCs are identified in Subsection 19K.11.

A second table, Table 19K-2, was prepared to show those SSCs with small to moderate values of risk achievement worth. Most of these have very low Fussell-Vesely importance, with the exception of some common cause failures, indicating a low probability of failure. However, if they fail, the impact on CDF is not negligible. Many of these SSCs have very similar risk achievement worths because their failure would result in failure of the same system to perform its functions.

Initiating events that are significant contributors to CDF in the Level 1 analysis are listed in Table 19K-3. The isolation/loss of feedwater initiating events and the loss of offsite power are the biggest contributors to CDF. Station blackout is a subset of the loss of offsite power event. These are followed by reactor shutdown and the turbine trip initiators. The loss of coolant accidents and other initiators are very small contributors to the plant CDF.

Systems that are most important to limiting the frequency of isolation/loss of feedwater are the Feedwater and Feedwater Control (FWC) Systems. The FWC System is triply redundant, having digital logic with self-checking. The automatic checking of the FWC System assures that its reliability remains high throughout operation. The COL applicant should assure that maintenance and test activities for risk-significant components in the FW System, the FW pumps and motors, are appropriate to assure high reliability.

The components within the control of the COL applicant that are of most significance to limiting the frequency of station blackout are the diesel generators, DC batteries, and the combustion turbine. The COL applicant should assure that maintenance and test activities for these components are appropriate to assure high reliability. The recovery of offsite power and diesel generators is also important.

Unplanned manual reactor shutdowns occur with a relatively short time for preparation, in contrast with a planned shutdown. To assure that the unplanned shutdowns will not cause undue risk to the plant, the training procedures should include adequate training, including simulator exercises, for such events so the operating crews can respond to plant conditions during such shutdowns on short notice.

The Reactor Core Isolation Cooling (RCIC) System, the High Pressure Core Flooder (HPCF) System, the Residual Heat Removal System (RHR), and their support systems are important systems. The Containment Overpressure Protection System (COPS) is also judged to be important. These systems are discussed in more detail in subsequent subsections.

The RAP activities for important SSCs identified by consideration of initiating events are included in Table 19K-4.

The relative importance of some ABWR features is not established by the Level 1 analysis described above because some important SSCs are not treated in the Level 1 calculation. To identify other important SSCs, the Level 2, seismic, fire, flood and shutdown analyses results were carefully reviewed by knowledgeable engineers who identified additional SSCs for the RAP. The important SSCs identified in these other studies are given in Subsections 19K.4 through 19K.10, and RAP activities are in Subsection 19K.11.

19K.4 Determination of "Important Structures, Systems and Components" for Level 2 Analysis

The Level 2 analysis evaluates the offsite release of fission products following core damage. Those analyses related to the consequences of core damage were reviewed, including source term sensitivity studies, deterministic analysis of plant performance, and containment event trees. Those systems which would be important with regard to mitigating a core damage event were considered as potential risk-significant SSCs. The following features were identified:

(1) The Automatic Depressurization System (ADS)

The ADS depressurizes the RPV so that the low pressure systems can inject water. Even if no water injection is available, the depressurization via one safety/relief valve (SRV) eliminates the potential for direct containment heating in event of RPV failure. The SRVs are important SSCs for the ADS since they are the components that function to release steam to reduce RPV pressure.

(2) The AC-independent Water Addition (ACIWA) System

The ACIWA System has two major benefits. First, it can inject water into the RPV to prevent core damage or facilitate in-vessel recovery. Second, it helps protect the containment by flooding the lower drywell (diverse from LDF) to cool corium in event of core melt and vessel failure. The ACIWA System can also be used to reduce high drywell temperature when operated in the drywell spray mode.

Also, for sequences with loss of containment heat removal, the ACIWA System adds thermal mass to the containment, significantly delaying the time of rupture disk opening. The important SSCs for the ACIWA System are the valves, the diesel-driven pump, and the onsite fire truck as they provide for the addition of water to the core and/or drywell.

(3) The Lower Drywell Flooder (LDF)

The LDF System was selected because it is important in providing cooling for corium released from the reactor vessel and in scrubbing fission products released from the corium in the event all the automatic and manual systems fail to inject water. The LDF fusible plug valves are important SSCs for the LDF System since they provide for flooding of the lower drywell.

(4) The Containment Overpressure Protection System (COPS)

The COPS is important since it prevents containment failure and assures a fission product release path through the suppression pool. This serves to limit the potential offsite dose after a core damage event. Sequences which result in slow pressurization will lead to a COPS operation, as opposed to the drywell failure. Since the suppression pool scrubs fission products before they enter the wetwell air space, this results in a much lower source term than does the case of a drywell head failure.

The COPS will also reduce the potential for a Class II sequence to lead to core damage. The predominant mechanism for core damage in Class II sequences is failure of containment or reactor building structures causing damage to long term heat removal equipment. Operation of the COPS directs the gas flow to the stack, preventing damage to the equipment. The COPS SSCs identified by the analysis are the rupture disks, which prevent containment failure and limit offsite doses after core damage, the isolation valves, and the flow lines.

(5) The RHR System

The RHR System is a primary source of decay heat removal. Decay heat removal is necessary to prevent fission product release from the containment in the unlikely event of a severe accident. Also, the drywell spray function of the RHR is an important feature in limiting the consequences of the Level 2 analysis. The technical specifications and valve and pump inservice testing (Table 3.9-8) requirements for the RHR System were reviewed and it was concluded that except for a maintenance requirement on the RHR Non-safety Related Valve, no additional reliability and maintenance actions are needed in the RAP for the RHR System.

The RAP activities for important SSCs identified by this Level 2 analysis are given in Table 19K-4.

19K.5 Determination of “Important Structures, Systems and Components” for Seismic Analysis

The seismic analysis considers the potential for core damage from plant damage resulting from a seismic event. The results of the seismic analysis identified key features by consideration of those SSCs important to reactor shutdown or to decay heat removal which could potentially be damaged by seismic action.

The seismic margins analysis calculated high confidence, low probability of failure (HCLPF) accelerations for important accident sequences and classes of accidents. The analysis showed that all SSCs in the analysis have HCLPF significantly greater than that of the safe shutdown earthquake (SSE). Because an important failure mode for beyond design bases earthquakes is the failure of the RHR heat exchanger in such a manner as to drain the suppression pool, the RHR heat exchanger was assigned a reasonably high HCLPF in the ABWR PRA-based seismic margins analysis.

The two methods that were used to identify important SSCs from the standpoint of seismic analysis are the following:

(1) Identification of the SSCs whose failure would provide the shortest path to core melt in terms of the number of failures required, and comparison of the seismic capacities of those SSCs.

(2) Identification of the most sensitive SSCs in terms of their effect on accident sequence and accident class HCLPFs resulting from variation of component seismic capacities.

The primary containment and the Reactor Building are the Category I structures in the design certification scope with the lowest values of HCLPF, but since both have high values of HCLPF no special RAP activities are deemed necessary for these structures. Other SSCs identified by the seismic analysis as being important are as follows:

- The diesel generators, 480 VAC transformers, motor control centers and circuit breakers of the emergency AC Power System
- The batteries and cable trays of the DC Power System
- The heat exchangers of the Residual Heat Removal System
- The pumps, pump house and air conditioners of the Service Water System
- The SLC tank, valves, and piping and the motor driven pumps of the Standby Liquid Control System
- The valves, piping, and diesel-driven pump of the Fire Water System
- The discharge lines of the SRVs of the Nuclear Boiler System

The RAP activities for important SSCs identified by this seismic analysis are given in Table 19K-4.

19K.6 Determination of “Important Structures, Systems and Components” for Fire Analysis

The fire analysis considers the potential for core damage from plant damage resulting from a fire. The important SSCs identified by this analysis are the room fire barriers, which prevent the fire from spreading to other rooms, the Smoke Removal System, which maintains pressure differentials to exhaust smoke rather than allow it to reach other areas, and the remote shutdown panel and control which are needed following a fire in the control room or HVAC failure in the control room.

The RAP activities for important SSCs identified by this fire analysis are given in Table 19K-4.

19K.7 Determination of “Important Structures, Systems and Components” for Flood Analysis

The flood analysis considers the potential for core damage from plant damage resulting from a flood. The important SSCs identified by this analysis are the ECCS rooms, RCW rooms, Reactor Service Water (RSW) pump rooms and RSW electrical equipment rooms, and control and reactor building external water tight doors, which prevent water from flowing into rooms other than the one with the leak; redundant supply side isolation valves on the RSW System prevent gravity drainage of the UHS basin, which limits the amount of water spilled into the control building or RSW pump rooms; circuit breakers that will trip RSW pumps, which also limits the amount of water spilled into the control building; isolation valves in the Circulating Water System (CWS); level switches in the turbine building condenser pit, the control building RCW rooms, and the RSW pump rooms; sump pump operation; overfill lines in reactor building sumps on floor BIF; and room drain lines.

The RAP activities for important SSCs identified by this flood analysis are given in Table 19K-4.

19K.8 Determination of “Important Structures, Systems and Components” for Shutdown Analysis

The shutdown analysis considers the potential for core damage during shutdown. Potential core damage during shutdown arises when the RHR System is lost. The important SSCs identified by this analysis are the ADS System, the RHR System for shutdown cooling and in the low pressure flooder (LPFL) mode, the High Pressure Core Flooder (HPCF) System, the AC-independent Water Addition (ACIWA) System, and the Control Rod Drive (CRD) System. Also important are the support systems, AC power and DC power. The important components are SRVs of the ADS System, valves and pumps of the RHR System and of the HPCF, ACIWA and CRD Systems.

The RAP activities for important SSCs identified by this shutdown analysis are given in Table 19K-4.

19K.9 Identification of Important Systems with Redundant Trains

Several plant systems have multiple trains of which only one is required to operate to perform the system safety function, the other trains providing redundancy. Because of this redundancy, components of the systems may not show up in a listing of high importance components. However, it is possible that operation or maintenance activities related to these systems could introduce some common cause failures which could affect all similar trains of a given system and, thereby, render all trains of such systems incapable of performing their safety functions. Engineering judgment was used to identify the multiple train systems having important safety functions that should be checked in addition to any identified component tests or maintenance. The systems selected are the RHR System in the shutdown cooling and the low pressure flooder (LPFL) mode, the High Pressure Core Flooder (HPCF) System, the Reactor Water Cleanup (CUW) System, the Reactor Service Water (RSW) System, the Reactor Building Cooling Water (RCW) and the AC Electrical System.

A single train of each of these systems should be designated for RAP by the COL applicant and the train should be given a walkdown inspection every refueling outage. The inspection should verify that system equipment is being operated and maintained properly so that there is no reason to suspect that other trains of the same system have problems that would preclude the system from performing its safety functions. The RAP activities for trains of systems identified by this analysis are given in Table 19K-4.

19K.10 Identification of Important Capabilities Outside the Control Room

Most safety-related actions by plant operators are conducted from inside the control room. However, in some sequences it is necessary for the operators to take appropriate action from stations outside the control room. Engineering judgment was used to identify activities that the operators should be capable of performing outside the control room, during internal flood, during reactor shutdown, or when the control room is inaccessible, such as in event of a fire.

The identified activities outside the control room are:

(1) Execution of the emergency operation procedures for operating the remote shutdown panels

(2) Manual operation of the RCIC System from outside the control room

(3) Closing water tight doors that are open (if there is flooding in the intact ECCS division) before opening doors to attempt corrective action

(4) Manual lineup of the combustion turbine generator and emergency diesel generators to non-safety-related buses

(5) Manual alignment of the AC-independent Water Addition System

(6) Manual bypass of the regenerative heat exchanger in the Reactor Water Cleanup System

(7) Connection of the diesel fire truck to the AC-independent Water Addition System after a seismic event

The RAP activities identified by these considerations are given in Table 19K-4.

19K.11 Reliability and Maintenance Actions

The individual SSCs identified as being “important” in Subsections 19K.3 through 19K.10 were reviewed to determine the appropriate reliability and maintenance actions. These actions are defined in this subsection.

The important SSCs are tabulated in Table 19K-4, showing the failure mode or cause, the recommended maintenance, the test or maintenance intervals and the basis for intervals, and the unavailability or failure rate. Where several components in one system are identified, such as for the RCIC, the ACIWA, and COPS, only the system unavailability is given. If the owner/operator cannot demonstrate each component meeting its unavailability assumption, the PRA assumptions will still be valid if the system unavailability assumption is met.

19K.11.1 Component Inspections and Maintenance

The system of greatest FV importance with respect to outage time is the RCIC System, which has been assigned a small value of unavailability for test and maintenance. The amount of time the RCIC System is unavailable because of test and maintenance should be monitored to assure that it remains within the specified assumption annually. Sensitivity studies of increased SSC unavailabilities showed that an increase in RCIC unavailability would cause the greatest increase in estimated core damage frequency of any SSC. The RCIC System was also found to be the most sensitive system to increased outage time assumptions. The highest contributor to uncertainties in the CDF as well as the CDF estimate was RCIC test and maintenance.

The RCW System and RSW System have high FV importance with respect to common cause failure impacts, as these systems support a number of front-line safety systems. Maintenance and testing tasks are provided for the key components in each division, including pumps, heat exchangers and the service water cooling tower fans.

The Remote Digital Logic Controller (RDLC) performs the Remote Input/output Function (RIF). Components that provide this function are identified by the Level 1 analysis as high importance components (i.e., CCF REMOTE MULTIPLEXING UNITS, CCF OF TRANSMISSION NETWORK (EMS), and CCF SYSTEM LOGIC UNIT FAILS). Safety system RDLCs have self-diagnostics that detect failures during on-line operation. In addition, one division of RDLCs can be bypassed and tested during plant operation without loss of system function. Such tests provide a periodic verification of the RDLC operability. During plant outages more detailed RDLC tests are possible, including a complete system test and identification of signal errors. These tests will include verification that the remote RDLCs function properly. RDLC RIF that are suggested as part of the RAP are given in Table 19K-4.

The turbine of the RCIC System is an important component, as identified in Table 19K-1. Periodic startup and operation of the RCIC turbine is one way to monitor this turbine, and less frequent turbine inspection and refurbishment are also recommended. The RCIC pump is tested at the same time by measurement of speed, flow rate, differential pressure, and vibration. Many of the RCIC valves are also tested when the turbine testing is done. These RAP activities are included in Table 19K-4.

Trip logic functions (TLFs) for the Reactor Protection System (RPS) represent another high importance component. Functional tests of these TLFs are performed at predetermined intervals. Additional offline, end-to-end (sensor input to trip actuator) testing of TLFs, which exercises the safety system logic and control logic processes, is important because it allows the detection of failures not sensed by the online system. The TLF tests that are suggested as part of the RAP are given in Table 19K-4.

Station batteries receive periodic checks in accordance with plant technical specifications. These checks will be adequate to assure that the batteries will have the reliability assumed in safety analyses.

For the normally closed, fail closed (NCFC) injection valves, the steam supply valves and the bypass valves of the RCIC System, which normally are not required to operate during plant operation, a quarterly full stroke test is judged to be appropriate for the RAP. Such tests are in compliance with ASME Code requirements for valves in nuclear plants. Detailed disassembly, inspection and refurbishment of valves would be done less frequently. The normally open, fail open (NOFO) bypass valves should be considered for similar tests. Suggested RAP activities and frequencies, and the basis for each suggested activity, are shown in Table 19K-4 for identified failure modes.

The HPCF maintenance valve is normally locked open, and its failure mode is being left closed following maintenance. To prevent this human error from occurring, administrative controls should require independent verification of the valve position following maintenance, positive control of the key to the valve lock, and control room verification of the valve position prior to startup. The RAP activities are in Table 19K-4.

The RCIC isolation signal logic should have a logic functional test every three months to assure it is functioning properly as shown in Table 19K-4.

Reliability of offsite power sources cannot be completely controlled by the plant. However, to assure that plant equipment does not contribute to power losses, inspection of switchyard equipment should be performed with a frequency of at least once every six months in accordance with site administrative procedures. Such inspections should include confirmation of secure structural mounting of equipment, physical condition of insulators and other supporting apparatus, and visual inspection of transformers and other oil filled equipment for oil leaks. Infrared thermography should be used to detect hot spots on electrical equipment and connections. All supports and supporting structures should be examined for structural integrity. In addition, suggested RAP activities given in Table 19K-4 for protective relay testing and for control power source components are recommended.

Common-cause miscalibration of RHR flow meters, and of Level 8 sensors, and common-cause failure (CCF) of digital trip functions (DTFs), and of Level 2 sensors, will have acceptable probabilities if adequate administrative controls are exercised. Calibration procedures for RHR flow meters and for Level 8 sensors should include notes about the safety importance of these instruments. Historical trend analysis should be performed for Level 2 sensors at each calibration. The procedure for testing DTFs should include a warning about their importance to safety. Suggested RAP activities are given in Table 19K-4.

The CCF of safety relief valves (SRVs) can be kept to an acceptably low probability if the SRVs receive the appropriate inservice inspection, if identified problems receive root cause analysis and correction, and if the configuration and qualified life of the valves at the site (or elsewhere) is maintained correctly, including consideration for aging and wear of parts. The SRV control panel can also be tested, separate from valve operation, to assure that it works properly. An inservice check to detect for valve leakage that can lead to setpoint drift is the temperature alarm on the tail pipe. The inservice inspection of SRVs is included in Table 19K-4 for RAP.

Isolation check valves of the NBS are leak tested at refueling outages, and that test demonstrates that the valves move from open to closed. Subsequent plant operation of the feedwater system opens the valves, giving assurance that they have ability to open. The NBS manual isolation valve has a stroke test at each refueling outage to assure that it can function. Testable check valves of the RCIC System can also be checked at each refueling to assure that they would function properly if conditions required a change in position. These valve tests are included in Table 19K-4.

19K.11.2 RCIC System Testing

The Level 1 analysis identified the Reactor Core Isolation Cooling (RCIC) System as one whose failures contribute substantially to CDF. Failure of RCIC to start or failure to continue operation after start are failure modes that are identified as significant. To provide assurance that the RCIC operation will be reliable, it is suggested that the system be started and operated long enough to demonstrate stable operation at least once every three months. The flow rate of RCIC should be measured to verify that it meets design requirements for injection into the RPV. Quarterly tests are with flow to the suppression pool. The RCIC System test will accomplish many of the RCIC turbine, pump and valve tests and will demonstrate that the Division 1 distribution panel is functioning. Components of RCIC that have been identified as significant, including many valves and instruments, are included in Table 19K-4 with identified failure modes and suggested RAP activities.

19K.11.3 Depressurization

The ADS technical specifications were reviewed, and it was concluded that no additional reliability and maintenance actions are needed. Testing of ADS System SRVs is included in Table 19K-4 with the other RAP activities.

19K.11.4 Lower Drywell Flooder (LDF)

In order to assure a dry cavity at the time of vessel failure, it is important that there be negligible probability of premature or spurious actuation of the passive flooder valves at temperatures less than 533 K (500°F) or under differential pressures associated with reactor blowdown and pool hydrodynamic loads.

Activities suggested for RAP are given in Table 19K-4 and discussed below.

(1) The ten fusible plug valve flanges and outlets should be inspected every refueling outage to assure there is no leakage.

(2) Two of ten fusible plug valves should be removed, inspected and their temperature setpoints tested every two refueling outages. (See testing and inspection requirements, Subsection 9.5.12.4.)

19K.11.5 AC-Independent Water Addition (Firewater) System

Inspection and testing of this system should be included in RAP. However, because of the importance of manual alignment, lining up the firewater should be specifically included in the training programs to assure that the system benefits are obtained. Specific procedures are required to be developed by the COL applicant to align the ACIWA System for vessel injection or drywell spray. See Subsection 19.9.7.

The strategy discussed below is recommended to test key components to assure that pumps and valves are operable and that there is no significant flow blockage in the flow paths from the Fire Water System to the reactor pressure vessel and to the drywell spray. Component testing is included in Table 19K-4.

(1) Onsite fire truck (pumper) maintenance should be conducted in accordance with the utility's normal fire protection maintenance procedures. A site service test of fire truck performance should be performed annually or after any major repairs in conformance with Chapter 11 of NFPA 1901, "Standard on Automotive Fire Apparatus". These tests should demonstrate that the pumper/engine combination is capable of meeting the performance requirements of the original certification or acceptance tests. Fire truck reliability for supporting the water injection function is

assumed to be high. A satisfactory service test should consist of pumping water to the ground or back to the suction source as follows:

(a) Twenty minutes of pumping 100% rated capacity, preferably at draft, at 1.034 MPa net pump pressure

(b) Ten minutes of pumping 70% rated capacity at 1.379 MPa net pump pressure

(c) Ten minutes of pumping 50% rated capacity at 1.724 MPa net pump pressure

Engine speed should be recorded for each condition. A “spurt” test need not be conducted, but if care is taken to ensure that the pump does not cavitate, running the pumper with wide open throttle at 1.138 MPa net pump pressure may give a good indication of engine condition.

(2) As a part of the normal testing required by the utility’s fire protection procedures, the following tests should be considered:

(a) Once every two refueling outages or every four years (which ever is most convenient) the fire truck should be used to pressurize the fire protection system and test the flow capacity. Suction should be from both fire protection tanks and the ultimate heat sink water supply.

(b) Once every two refueling outages or every four years the flow capacity of both the AC-driven and the direct diesel-driven fire pumps should be tested. This flow test can be alternated with the fire truck flow test (2a above). The diesel-driven fire water pump is assumed to have high reliability for supporting the water injection function.

(3) Once every two years the RHR non-safety-related valve (E11-F103C of Figure 5.4-10, Sheet 7) which must operate to provide flow to the vessel, or to the drywell spray or wetwell spray, should be manually opened and closed. Safety-related valves E11-F101C and E11-F102C are exercised every three months as part of the valve inservice testing program.

(4) Once every four years the AC-independent Water Addition (ACIWA) System flow and flow monitoring instrumentation from the fire protection system (FPS) to the RHR main loop should be tested. This can be accomplished during a reactor shutdown by initially isolating and closing off the branch lines of the RHR main loop C (however, the heat exchanger throttle valve E11-F004C remains open) and stopping both pumps, C001C and C002C. After ACIWA valves E11-F101C and E11-F102C are opened to apply the FPS pressure to the RHR main loop, the shutoff head pressure should be verified. With the RHR main loop closed off, no flow should occur. Then for a short time period, the flushing drain to the radwaste using valves E11-F029C and E11-F030C, Figure 5.4-10, Sheet 6, can be opened. The resulting flow can be measured with flow meter E11-FE012B, Figure 5.4-10, Sheet. 4.

Throttling valve E11-F030C can be used to turn the flow on and off and limit the flow to the desired rate and duration. The flow duration should be minimized to reduce the load to radwaste. The test should be repeated first with valve E11-F101C closed, then with the fire truck hose connection and valves E11-F101C and E11-F103C opened, Figure 5.4-10, Sheet 7.

(5) Once every five years all fire protection and RHR piping which forms the AC-Independent Water Addition System should be tested to ensure that it is structurally intact and properly supported.

(6) Seismic-related inspections listed in Subsection 19K.11.7 should be done.

19K.11.6 Containment Overpressure Protection System (COPS)

The COPS is identified in Subsection 19K.4 as important to limiting fission product release. Suggested system component testing as part of RAP is identified in Table 19K-4. Also, system flow testing and special operator training should be considered for inclusion in the RAP.

(1) Air-operated valves (AOVs) in series with rupture disks should be maintained in the same manner as containment isolation valves. It is suggested that, during preoperational testing and during each R/M outage, each valve be exercised and proper open and closed local and control room indications be checked. Any position other than full open should alarm in the control room. After valves are returned to the open position, indication should be verified locally and in the main control room. These tests are included in Table 19K-4.

(2) Rupture disks should be maintained as required by the ASME code. The rupture disk manufacturer should perform the necessary tests to certify that the rupture disks will open at a pressure within 5% of the rated value. Every five years, the disks should be tested and replaced. These tests are included in Table 19K-4.

(3) A flow test should be conducted every five years to assure that there are no obstructions in the pressure relief path.

(4) Special training on operator actions following rupture disk opening should be included in the plant training program.

19K.11.7 Seismic-Related Inspections

The seismic capability of the following equipment is identified (Subsection 19K.5) as risk-significant: emergency diesel generators, 480 VAC transformers, motor control centers and circuit breakers of the AC Power System; batteries and cable trays of the DC Power System; the SLCS tank, valves and piping and motor driven pumps of the Standby Liquid Control System; the Service Water System pumps, pump house and air conditioner; the heat exchanger of the RHR System; the valves, piping and diesel-driven pump of the Fire Water System, and the discharge lines of the SRVs. For this equipment, the seismic related inspections detailed in Subsection 19H.5 should be conducted once every 10 years or after any earthquake equal to or greater than that corresponding to the cumulative absolute velocity (CAV) shutdown threshold.

19K.11.8 Plant Structures

No maintenance activities other than those already associated with the inservice surveillance of the seismic instruments defined in Subsection 3.7.4.5 are needed for seismic events. The seismic instrumentation program (Subsection 3.7.4) is designed to provide information on the input ground motion and resultant responses of representative Category I structures and equipment in the event an earthquake occurs sufficient to activate the seismic instrumentation. If the earthquake exceeds that corresponding to the CAV shutdown threshold, the plant is shut down, manually if necessary, and a detailed post-earthquake evaluation is undertaken. When it is determined that plant structures and equipment were not damaged, the plant can be safely re-started on the basis of seismic considerations.

19K.11.9 Hydraulic Control Units and Control Rod Drives

The technical specifications associated with the hydraulic control units and control rod drives were reviewed. It was concluded that no additional reliability and maintenance actions are needed beyond those in technical specifications.

19K.11.10 Emergency Diesel Generators

Maintenance for the emergency diesel generators is expected to be performed in accordance with site procedures and the manufacturer’s recommendations. Surveillance testing is required in accordance with Regulatory Guide 1.9, “Design, Qualification and Testing of Diesel Generators”, and with the surveillance requirements described in the Technical Specifications (Subsection 16.11.1) beginning with SR 3.8.1.4. Seismic-related inspections noted in Subsection 19K.11.7 should be done.

Maintaining emergency diesel generator reliability is a basic part of the station blackout rule (10CFR50.63). A reliability assurance program is required which maintains a target reliability. In view of the existing requirements noted above, it is judged that additional reliability and maintenance activities are not needed. If a diesel generator fails to start or run following a demand, recovering it in a relatively short time is very important.

19K.11.11 Combustion Turbine Generator

Maintenance for the combustion turbine generator (CTG) is expected to be performed in accordance with site procedures and the manufacturer’s recommendations. Suggested surveillance testing includes quarterly operation at rated speed and rated load until temperatures reach steady state values, approximately one hour. Also quarterly there should be a check of oil levels and assurance that there are no oil or fuel leaks. Quarterly the oil should be sampled and analyzed for acceptable quality. At each refueling/maintenance outage CTG fuel oil and lube oil should be inspected for deterioration; and replaced as necessary. Also, the fuel, lube oil and air filters should be replaced. There should be a thorough inspection of the entire assembly to assure that the inlet and outlet plenums are not blocked or deteriorating. Also, a complete visual inspection of the power unit should be made to assure that support bolts are secured and that there are no cracks and no blown gasket or engine hot spots. These tests and preventive maintenance activities are included in Table 19K-4.

19K.11.12 Fire Protection

The room fire barriers, the Smoke Removal System, and the remote shutdown panel and control were determined to be relatively important (Subsection 19K.6). Fire barriers, including penetrations, should be inspected periodically to assure that they retain their integrity with respect to confining a fire. The Smoke Removal System should be operated annually to demonstrate that it will be able to maintain a negative pressure in a room with a fire so that probability of propagation of fire and/or smoke to other rooms is low.

Smoke Removal System testing will be performed on each smoke removal zone in the reactor building, the service building and the radwaste building. Smoke Removal System testing will be patterned after damper alignment intended for smoke removal operation of the system. This consists of reducing normal exhaust from adjoining zones, to increase their pressure, and bypassing exhaust filters or small exhaust fans in the zone being tested to increase its exhaust flow rate. This will establish a pressure differential between zones to reduce the possibility that smoke will get into zones not directly affected by a fire.

Personnel entry to an area experiencing a fire is gained from an adjacent fire area which, by design, is at a positive pressure with respect to the area containing the fire. The pressure differential is sufficient to provide adequate velocity through the open door to push the combustion products back into the zone of the fire. The flow through the open door into the area of the fire and out of the area through the fire's exhaust duct system is enhanced by the positive pressure of the non-fire area. The HVAC Systems with recirculated air are manually switched over to a once-through system during a fire or test, so there is no direct mixing of smoke from one room to another.

The differential pressure between zones will be greater if all doors are closed, but each zone is relatively large, so one or two open doors between zones will not have a significant impact on the tests or on Smoke Removal System operation during a fire. Personnel should be advised that it is permissible to open doors during a test (or during a fire), but that doors should normally be closed at those times. This will allow personnel access to all related areas, and will not unduly restrict fire fighting personnel in event of a fire.

The remote shutdown panel should be tested periodically to show that it can perform its functions that will lead to safe shutdown. These RAP activities related to fire protection are included in Table 19K-4.

19K.11.13 Flood Protection

The important SSCs for flood protection are the water tight doors on external entrances to the control and reactor buildings and in ECCS, RSW pump, and RCW rooms, the RSW and CWS isolation valves, the circuit breakers that trip the RSW pumps and water level sensors in the turbine building condenser pit, RSW pump rooms, and control building RCW rooms; sump pump operation; overfill lines on reactor building sumps on floor BIF; and room drain lines (Subsection 19K.7). Periodically room water barriers should be inspected to assure that they will prevent the spread of flooding, room drain lines should be checked to ensure no blockage exists, RSW isolation valves (MOVs) should be stroke tested (normally accomplished by switching from one pump to the standby pump in a given loop), CWS isolation valves should be stroke tested, the ability of RSW pump circuit breakers to trip upon receipt of a trip signal should be demonstrated, as well as RSW System isolation capability. These RAP activities are included in Table 19K-4.

19K.11.14 Shutdown Protection

The shutdown analysis (Subsection 19K.8) identified as important components the SRVs of the ADS System and valves and pumps of the RHR system (including the LPFL mode) and of the HPCF, ACIWA and CRD Systems. RAP activities for SRVs are covered in Subsection 19K.11.3, and those for ACIWA components are covered in Subsection 19K.11.5. Testing of valves and pumps of the HPCF and RHR Systems and for the LPFL function of the RHR are covered by the technical specifications and valve and pump inservice testing (Table 3.9-8) for these systems. These testing requirements were reviewed and it was concluded that no additional reliability and maintenance actions are needed. This RHR testing also provides adequate assurance that the suppression pool temperature will be maintained below its high temperature limit (Table 19K-2).

The CRD System is normally operating, but system flow can be increased by opening some partially closed valves and/or by operating the second pump in addition to the operating one. The RAP activity, in Table 19K-4, is to review the CRD operating procedures and verify that they include steps to increase flow when necessary.

During plant shutdown the normal cooling for the reactor will be by one division of the RHR System, in the shutdown cooling mode. This RHR division is powered by its divisional AC power with instrumentation power from the divisional DC power. A second RHR division of safety system with its supporting AC and DC power will be in standby, ready to operate at any time. (Electrical equipment from other systems is expected to be operating on the power systems that are in standby for the RHR function.) The third division of safety system is completely available for maintenance.

During shutdown, the failure of the operating RHR loop is one initiating event with an assumed low probability. Testing of key RHR components, consistent with Table 3.9-8 for in-service testing, is identified with RAP activities in Table 19K-4. Operators should monitor RHR loop failure rate and take corrective action if the failure rate exceeds the assumed probability during operation.

Testing and maintenance activities will be possible on AC and DC Power Systems in the third division which is in maintenance. Inspections related to reliability of offsite AC Power Systems are discussed in Subsection 19K.11.1, as are periodic checks on station batteries. Testing of emergency diesel generators and the combustion turbine generator are covered in Subsections 19K.11.10 and 19K.11.11, respectively. Since the two operating power systems are continuously monitored, it is not necessary to identify additional special tests or maintenance as part of the RAP for the AC and DC Power Systems.

19K.11.15 Prevention of Intersystem LOCA

The Reactor Water Cleanup (CUW) System provides a negligible benefit in the ABWR PRA by removing decay heat at high pressure. It would only be used in this mode if the containment cooling mode of the RHR system was disabled. During all operating modes, an unisolated CUW break could cause serious consequences, therefore these CUW isolation valves must be capable of automatically isolating against a differential pressure equal to the operating pressure of the reactor coolant system in the event of a LOCA in the CUW. If the automatic isolation valves fail to close, the operator can close the remote manual shutoff valve from the control room to terminate the LOCA. The RAP activities to assure reliability of these isolation valves are listed in Table 19K-4.

19K.11.16 Determination of "Important Structures, Systems and Components" for Suppression Pool Bypass Analysis

The suppression pool is an important containment feature for severe accident progression and fission product removal, since releases from the reactor vessel are either directly routed to the pool (e.g., transients with actuation of ADS) or pass through the pool via the drywell-wetwell connecting vents.

If an event leads to pressurization of the wetwell to the extent that the containment rupture disks open, the vacuum breakers would open to equalize pressures in the wetwell and drywell. The breakers would then close, thereby isolating the drywell from the wetwell. Failure of a DW-WW vacuum breaker to close following the assumed event would provide a significant bypass from the drywell into the wetwell airspace. If the rupture disk is open and one of the vacuum breakers has not closed there would be a direct pathway from the drywell to the wetwell and to the environment.

The following are critical to assuring a low risk from wetwell/drywell vacuum breaker bypass:

(1) A low probability of vacuum breaker leakage

(2) A low probability that the vacuum breakers fail to close

(3) A high availability of drywell or wetwell sprays (and ACIWA as a backup) to condense steam which bypasses the suppression pool.

Recommendations for testing of DW-WW vacuum breakers and ACIWA System RAP activities are included in Table 19K-4.

19K.11.17 Offsite Power

The availability of offsite power is considered to be very important in preventing core damage. If offsite power is lost, then recovering it in a relatively short time is important. However, this is not addressed specifically in Table 19K-4, because offsite power is required to keep the plant in operation.

19K.11.18 Condensate and Feedwater System

The Condensate and Feedwater System is very important in preventing core damage. If feedwater is lost following an initiator, then it is important to restore it. However, this item is not specifically addressed in Table 19K-4 because the Condensate and Feedwater System is required to keep the plant in operation.

19K.12 References

19K-1 “GE ABWR Licensing Review Bases”, August 1987.

19K-2 “ABWR Shutdown Risk Evaluation,” Toshiba UTLR-0013.

19K-3 “ABWR PRA-Based Reliability,” Toshiba UTLR-0012.

19K-4 “ABWR Probabilistic Evaluations,” Toshiba UTLR-0011.

Table 19K-1 Not part of the DCD (Refer to Reference 19K-3)

Table 19K-2 Not part of the DCD (Refer to Reference 19K-3)

Table 19K-3 ABWR Initiating Event Contribution to CDF, Level 1 Analysis

Initiating Event	Events Per Year*	Total CDF *	Percent CDF Contribution*
Loss of Offsite Power**			
Reactor Shutdown			
Turbine Trip			
Isolation/Loss of Feedwater			

* Not part of DCD (Refer to Reference 19K-3).

** Station blackout events are a subset of loss of offsite power events.

Table 19K-4 Failure Modes and RAP Activities

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
RCIC System	System failure	See following items	See below	See below	*
RCIC System	Unavailable due to test or maintenance	Monitor unavailable time, compare with assumed 2%	Annually	Level 1 analysis	†
ECF	Common cause failure of all ECF to give proper signals	System functional test	3 months	Experience	
		Complete system test, error check	2 years	Experience	
One ESF RIF for Div 1 or one SLF/ECF Link for SLF Div 1	Failure of RIF or link between RIF and SLF	System functional test	3 months	Experience	*
		Complete system test, error check	2 years	Experience	
RCIC Turbine & Pump (System Test)	Mechanical failure to operate	Turbine startup and operation; measure pump vibration velocity & displacement, flow, speed, diff. pressure.	3 months	Experience‡	†
		Turbine inspection, refurbishment	5 years	Experience‡	
RPS Trip Logic Functions	Failure to trip upon demand	System functional test	3 months	Experience	*
		Complete system test, error check	R/M outage	Experience	
RCIC Check Valve F038	Failure to open	Open and close during system test	2 years	Table 3.9-8	†
RCIC Check Valves F003 & F005	Failure to open	Open and close test	R/M outage	Experience‡	†
RCIC Isolation Signal Logic	Failure to provide isolation signal when conditions warrant	Logic functional test	3 months	Experience	†
RCIC Min Flow Bypass Valve (NOFO or NCFC)	Failure to operate because of mechanical problems	Stroke test	3 months	Experience‡; ASME Code ISI	†
		Visual and penetrant inspection of stem, ultrasonic inspection of stem; replace if necessary.	10 years	Low failure rate; ASME Code ISI.	
	Failure to operate because of electrical problems	Electrical circuit test	3 months	Experience‡	

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
RCIC Injection Valve and Turbine Steam Supply Valve	Failure to open because of mechanical problems	Stroke test	3 months	Experience‡; ASME Code ISI	†
		Visual and penetrant inspection of stem, ultrasonic inspection of stem; replace if necessary.	10 years	Low failure rate; ASME Code ISI.	
	Failure to open because of electrical problems	Electrical circuit test	3 months	Experience‡	
RCIC Isolation Valves (NOFC)	Spurious failure because of mechanical problems	Stroke test	3 months	Experience‡; ASME Code ISI	†
		Visual and penetrant inspection of stem, ultrasonic inspection of stem; replace if necessary.	10 years	Low failure rate; ASME Code ISI.	
	Spurious failure because of electrical problems	Electrical circuit test	3 months	Experience‡	
Limit Switch on RCIC Turbine Exhaust Isolation Valve	Failure of switch to change position when valve movement occurs	Observation of limit switch actuation during valve stroke test	3 months	Experience‡	†
RCIC Flow Sensor FT-007-2	Sensor fails	Calibration of sensor	R/M outage	Experience	†
	Miscalibration	Review calibration procedures for note about potential safety considerations	R/M outage	Judgment	
RCIC Pressure Sensor PIS-Z605	Sensor fails	Calibration of sensor	R/M outage	Experience	†
	Miscalibration	Review calibration procedures for note about potential safety considerations	R/M outage	Judgment	
NBS Isolation Check Valves 003B & 004B	Fails to open	Leak rate test and subsequent operation of valves	R/M outage	Experience	*
NBS Manual valve F005B (NOFC)	Normally open valve fails closed	Stroke test	R/M outage	Experience	*

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
HPCF Maintenance Valve	Failure to open valve after maintenance	Independent verification of valve position following maintenance; position verification before startup	After maintenance, before startup	Judgment	*
Switch Yard Equipment	Failure results in loss of offsite power	Inspect switch yard equipment for signs of incipient failure, such as insecure structures, degraded insulators, leaking oil. Use thermography to detect hot spots on transformers, insulators, circuit breakers & connectors. Repair as necessary. See also the following items.	3 years	Experience	*
Switchyard Protective Relay	Relay failure to open or close on demand	Calibration, maintenance and test	1 year	Industry practice	ƒ
Auxiliary Relay Panels	Failure to provide power to loads	Routine cleaning and inspection	2 years	Industry practice	ƒ
Radio Batteries for microwave and fiber optic equipment	Battery failure	Routine test and maintenance	2 years	Industry practice	ƒ
Battery Chargers	Failure to provide charging current to batteries	Routine test and maintenance	18 months	Industry practice	ƒ
DC Power 125V Battery	Battery failure	Routine test and maintenance	1 month	Tech Spec	*

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
Feedwater Pumps	Failure during operation	Walkdown/external visual observation: -Oil level -Leaks -Land vibration	1 week	Experience	N/A
		Motor winding temperature, bearing temperature	1 week	Experience	
		Seal leakage, temperature, pressure	1 month	Experience	
		Oil sample/analyze	3 months	Experience	
		Performance data: -Discharge pressure -Inlet pressure -Flow rate -Peak vibration velocity -Motor current	3 months	Experience	
RHR Flow Meters	Common mode miscalibration	Review calibration procedures for note about potential safety considerations	Annual	Judgment	*
Level 2 Sensors	Common mode failure	Analyze Level 2 calibration data for trends of drifting or other CCF indications	R/M Outage	Judgment	*
Level 8 Sensors	Common mode miscalibration	Review calibration procedures for note about potential safety considerations	Annual	Judgment	*
Digital Trip Functions	Common cause failure to trip	Review trip unit test procedure to assure note about potential safety considerations	Annual	Judgment	*
Wetwell/Drywell Vacuum Breakers	Fail to close or leakage after close	Cycle through full open to full close. Check for leakage	R/M Outage	Experience	*

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
ADS System SRVs	Failure of several SRVs to open on demand or failure to remain open	Inspect and replace degradable parts and test for correct operation	5 years (max)	Environmental qualification	*
		Remove valve, test for setpoint pressure, adjust setpoint as necessary, test for seat leakage, repair. Stagger testing of valves, 50% at one outage	3 years	Experience, ANSI/ASME OM-1	
		Control Panel Test	3 months	Experience	
Non-ADS SRVs	Common mode failure of SRVs to open on demand or failure to remain open	Inspect and replace degradable parts and test for correct operation	5 years (max)	Environmental qualification	*
		Remove valve, test for setpoint pressure, adjust setpoint as necessary, test for seat leakage, repair. Stagger testing of valves, 50% at one outage	3 months	Experience, ANSI/ASME OM-1	
		Control Panel Test	3 months	Experience	
LDF Fusible Plug Valves	Failure to open at temperature	Two of ten plugs replaced; tested to verify temperature setpoint	2 R/M outages	Judgment	*
	Leakage	Inspect for leakage	R/M outage	Judgment	
ACIWA System	System unavailable	See following items	See below	See below	*
ACIWA Flow Instrumentation	Failure to accurately monitor flow	Measure zero flow and full system flow	4 years	Judgment	**
ACIWA Manual Valves (in RHR System)	Stuck closed	Stroke test	3 months	Experience[‡]; ASME Code ISI	**

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
Firewater System Pumps on Fire Truck	Failure of pumps to provide required flow at pressure	20 min pump at 100% rated flow, 1.13 MPa (150 psi)	1 year	Judgment	**
		10 min pump at 70% rated flow, 1.48 MPa (200 psi)	1 year	Judgment	
		10 min pump at 50% rated flow, 1.82 MPa (250 psi)	1 year	Judgment	
	Failure of system to deliver required flow	Test system flow with fire truck pumps, water from tanks & from UHS	4 years	Judgment	
		Test system flow with AC-driven and diesel-driven pumps, water from tanks & from UHS	4 years	Judgment	
ACIWA Diesel Pump	Failure to pump on demand	Pump start test	3 months	Experience	**
		Pump flow test	4 years	Experience	
RHR Non-Safety-Related Valve	Failure to open on demand	Manually open and close valve	2 years	Experience	**
Piping of AC-Independent Water Addition System	Piping failure that precludes successful operation	Piping visual inspection under operating pressure to assure no leaks	5 years	Judgment	**
		Piping support visual inspection to assure structural adequacy	5 years	Judgment	
COPS System	System failure	See following items	See below	See below	*
COPS AOVs	Inadvertently left closed following maintenance	Stroke test; position indication check; verification of local and control room indication following test	R/M outage	Experience	††
COPS Rupture Disks	Failure to open on demand	Disk replacement	5 years	ASME Code	††
		Verification of actuation within ±5% of rated pressure	5 years	ASME Code	

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
COPS Flow Lines	Flow blockage	Flow test to assure no blockage in line	5 years	Judgment	††
Fire Barriers Between Rooms	Failure to retain integrity	Inspection of fire barriers, including seals and penetrations	1 year & after major maintenance	Judgment	N/A
Smoke Removal System	Failure to maintain low room pressure	Operate system to assure that it functions as designed	1 year	Judgment	N/A
Remote Shutdown Panel	Failure to provide control for reactor shutdown	Demonstrate ability to shut down reactor and remove decay heat by operation at remote shutdown panel	R/M outage	Judgment	N/A
RCIC System	Failure to start or operate RCIC from remote location	Start and operate RCIC from stations outside the main control room	10 years	Judgment	N/A
Control and Reactor Building, RSW Pump House, and ECCS Room Watertight Doors	Failure to retain integrity	Inspection of watertight doors, including penetrations	1 year & after major maintenance	Judgment	N/A
RSW and CWS Isolation Valves	Failure to close on demand	Stroke test	1 month	Experience	*
RSW Pump Circuit Breakers	Failure to trip pump on demand	Breaker trip test to assure trip on demand	6 months	Judgment	*
CRD System Flow Increase	Failure to increase CRD flow in shutdown	Review CRD operating procedures to assure that steps to provide increased flow are included.	2 years	Reference 19K-2	*
DC Div 1 Distribution Panel (including Diode S1D)	Panel or diode failure	Panel function is demonstrated by system test	3 months	Experience	*

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
Room Sump Level Switches	Failure to detect water in sump	Observation of proper operation upon actuation	Annual	Judgment	*
Div 1 ECF	Failure	System functional test	3 months	Experience	*
		Complete system test, error check	2 years	Experience	
Sump Pumps	Failure to pump water out of sump	Start sump pump and observe operation	Annual	Judgment	*
Overfill Line	Line clogged	Inspect lines for debris	5 years	Judgment	*
	Water seal dry	Observe level in seal	Weekly	Judgment	N/A
Room Drain Lines	Line clogged	Inspect lines for debris	5 years	Judgment	N/A
Combustion Turbine Generator (CTG)	Failure to start and run	Start and operate CTG at rated speed and load for 1 hour	3 months	Experience	*
		Check oil levels, check for leaks	3 months	Experience	
		Sample, analyze oil. Replace as necessary	3 months	Experience	
		Inspect lube oil and fuel oil for deterioration. Replace oil filters as necessary; inspect inlet and outlet plenums and entire assembly	R/M outage	Experience	

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
Structures of Emergency AC Power EDGs, 480 VAC Transformers, MCCs & circuit breakers; DC Batteries and Cable Trays; RHR Heat Exchangers; SLC Tank, Valves, Piping & Pumps; Valves, Piping & Pump of ACIWA; SWS pumps, pump house and air conditioner; & SRV Discharge Piping of the NBS	Structural failure of supports during seismic event	Seismic walkdown to assure structural integrity	10 years	Judgment	N/A
		Visual inspection, support structures & devices.	10 years	Judgment	N/A
		Post-earthquake evaluation	After inspection level earthquake or larger quake	Judgment	N/A
Single Train of RHR System (Shutdown Cooling & LPFL Modes)	Common mode type failure	System walkdown to identify CCF type problems	R/M outage	Reference 19K-2	*
Single Train of HPCF System	Common mode type failure	System walkdown to identify CCF type problems	R/M outage	Judgment	*
Single Train of CUW System	Common mode type failure	System walkdown to identify CCF type problems	R/M outage	Judgment	N/A
Single Train of RSW System	Common mode type failure	System walkdown to identify CCF type problems	R/M outage	Judgment	N/A
Single Train of AC Electrical System	Common mode type failure	System walkdown to identify CCF type problems	R/M outage	Judgment	N/A
Emergency Diesel Generator	Failure to start and run	Start up to full load	1 month	Tech Spec	*

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
CUW Isolation Valves (NO, FAI)	Failure to operate because of mechanical problems	Stroke test	3 months	Experience‡; ASME Code ISI	*
		Visual and penetrant inspection of stem, ultrasonic inspection of stem; replace if necessary	10 years	Low failure rate; ASME Code ISI	
	Failure to operate because of electrical problems	Electrical circuit test	3 months	Experience‡	
CUW Remote Manual Shutoff Valve (NO, FAI)	Failure to operate because of mechanical problems.	Stroke test	Refueling outage	Judgement (non-safety-related)	*
	Failure to operate because of electrical problems.	Electrical circuit test	Refueling outage	Judgement (non-safety-related)	
Operating RHR Shutdown Cooling Loop	Failure to Operate because of mechanical or electrical problems	See following items	See below	See below	*
RHR Pumps	Failure to provide adequate flow at desired pressure	Discharge pressure test Inlet pressure test Flow test Vibration test	3 months	Table 3.9-8	‡‡
RHR Injection Valves, F005	Failure to operate	Stroke test	Cold shutdown	Table 3.9-8	‡‡
RHR Isolation Valves, F010, F011	Failure to operate	Stroke test	Cold shutdown	Table 3.9-8	‡‡
RHR Admission Valves, F012	Failure to operate	Stroke test	3 months	Table 3.9-8	‡‡
RCW Pumps	Failure to provide adequate flow at desired pressure	Discharge pressure test, inlet pressure test, flow test, vibration test	3 months	Table 3.9-8	
		Monitor pump parameters on the normally running pump to detect abnormalities	Weekly	Judgment	

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
RCW Heat Exchangers	Plugging/Fouling	Monitor heat exchanger flow and delta temperature/pressure to detect existence of fouling	Weekly	Experience	*
		Internal inspection of heat exchangers for plugging and fouling	R/M Outage	Experience	
RSW Pumps	Failure to provide adequate flow at desired pressure	Discharge pressure test, inlet pressure test, flow test, vibration test	3 months	Table 3.9-8	
		Monitor pump parameters on the normally running pump to detect abnormalities	Weekly	Judgment	
RSW Strainers	Plugging	Monitor RSW flow rate and strainer delta pressure for indications of plugging	Weekly	Experience	*
UHS Fans	Failure to provide adequate fan flow through tower	Flow test, vibration test	3 months	Experience	*
		Inspection and cleaning/lubrication	R/M Outage	Experience	
HPCF Pumps	Failure to provide adequate flow at desired pressure	Discharge pressure test, inlet pressure test, flow test, vibration test	3 months	Table 3.9-8	ƒƒ
HPCF Injection Valves F003 and F005	Failure to open because of mechanical problems	Stroke test	3 months	Experience; ASME code ISI	*
		Visual and penetrant inspection of stem, ultrasonic inspection of stem; replace if necessary	10 years	Low failure rate; ASME code ISI	
	Failure to open because of electrical problems	Electrical circuit test	3 months	Experience	
		Visual and penetrant inspection of stem, ultrasonic inspection of stem; replace if necessary	10 years	Low failure rate; ASME code ISI	

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
RHR Injection Valves F001, F003, F005, F006	Failure to open because of mechanical problems	Stroke test	3 months	Experience; ASME code ISI	‡‡
	Failure to open because of electrical problems	Electrical circuit test	3 months	Experience	
RHR Heat Exchangers	Fouling	Monitor and trend delta temperature across heat exchanger during RHR testing and operation	R/M Outage	Judgment	‡‡
ESF SLF Divisions	Failure to operate; failure to properly generate initiation signals	System functional test Complete system test; error check	3 months 2 years	Experience Experience	*
Suppression Pool	Loss of structural integrity; leakage	Periodic inspection of suppression pool structural elements to detect degradation, incipient leakage or corrosion	R/M Outage	Experience	N/A
Suppression Pool Temperature Sensors T53-TRS-601A and B	Sensor fails	Calibration of sensor	R/M Outage	Experience	*
	Common mode failure	Analyze Level 2 calibration data for trends of drifting or other CCF indications	R/M Outage	Judgment	*
Containment Penetrations	Leakage	Periodic inspection of penetrations to detect indications of degradation	R/M Outage	Experience	N/A
		Local leak rate testing	R/M Outage	Tech Spec	

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
Common Cause Failures					N/A
RHR System (Shutdown Cooling & LPFL Modes)	Common mode failure	System walkdown to identify CCF type problems	R/M outage	Judgment	
- Pumps	- Start and Run				
- Room Air Conditioners	- Start and Run				
- Injection MOVs	- Open				
HPCF System	Common mode failure	System walkdown to identify CCF type problems	R/M outage	Judgment	
- Pumps	- Start and Run				
- Injection MOVs	- Open				
- F005	- Mispositioning	Position check	Quarterly		
RCW System	Common mode failure	System walkdown to identify CCF type problems	R/M outage	Judgment	
- Pumps	- Run				
RSW System	Common mode failure	System walkdown to identify CCF type problems	R/M outage	Judgment	
- Pumps	- Run				
- Strainers	- Plug	Operation verification	Quarterly		
UHS System	Common mode failure	System walkdown to identify CCF type problems	R/M outage	Judgment	
- Fans	- Run				

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
Emergency Diesel Generators	Common mode failure - Start and run	Start up to full load	1 month	Tech Spec	
DC Power 125V Battery	Common mode failure	Routine test and maintenance	1 month	Tech Spec	

* Not part of DCD (refer to Reference 19K-3).
† RCIC component failure rates are included within the system unavailability.
‡ These types of valves have been used in operating BWRs, so there is much experience to guide owners/operators in care of the equipment.
ƒ Switchyard component failure rates are included within the switchyard equipment failure rate.
** ACIWA component failure rates are included within the system unavailability.
†† COPS component failure rates are included within the system unavailability. (Failure of the rupture disks to actuate upon demand before structural failure of the containment dominates failure of COPS.)
‡‡ RHR component failure rates are included within the system unavailability.
ƒƒ HPCF component failure rates are included within the system unavailability.