GE Nuclear Energy

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Docket No. STN 52-001

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Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of the Nuclear Reactor Regulation

Subject: **Submittal Supporting Accelerated ABWR Review Schedule - Resolution of TMI-related Outstanding Items**

Dear Chet:

Enclosed are SSAR markuups addressing the following DFSER TMI-related outstanding items:

Open Items	COL Action Items
20.3-6	20.3-1
20.3-9	20.3.1-2
	20.3.1-3
	20.3.1-4
	20.3.1-5
	20.3-2

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

cc: Bill Fitzsimmons (GE)
Norman Fletcher (DOE)
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JF93-63

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elevation which would be covered by post-LOCA flooding for unloading the fuel.

6.2.5.2.5 Pressure Control

(1) In general, during startup, normal, and abnormal operation, the wetwell and drywell pressures is maintained greater than 0 psig to prevent leakage of air (oxygen) into the primary containment from secondary containment but less than the nominal 2 psig scram set point. Sufficient margin is provided such that normal containment temperature and pressure fluctuations do not cause either of the two limits to be reached considering variations in initial containment conditions, instrumentation errors, operator and equipment response time, and equipment performance.

(2) Nitrogen makeup automatically maintains a 530 kg/m^2 (0.75 psig) positive pressure to avoid leakage of air from the secondary into the primary containment.

(3) The drywell bleed sizing is capable of maintaining the primary containment pressure less than 880 kg/m^2 (1.25 psig) during the maximum containment atmospheric heating which could occur during plant startup.

6.2.5.2.6 Overpressure Protection

(1) The system is designed to passively relieve the wetwell vapor space pressure at 5.6 kg/cm^2g. The system valves are capable of being closed from the main control room using AC power and pneumatic air.

(2) The vent system is sized so that residual core thermal power in the form of steam can be passed through the relief piping to the stack.

(3) The initial driving force for pressure relief is assumed to be the expected pressure setpoint of the rupture disks.

(4) The rupture disks are constructed of stainless steel or a material of similar corrision resistance.

(5) A number of rupture disks are procured at the same time and made from the same sheet to provide uniformity of relief pressure.

(6) The rupture disks are capable of withstanding full vacuum in the wetwell vapor space without leakage.

(7) The piping material is carbon steel. The design pressure is 10.5 kg/cm^2g (150 psi), and the design temperature is 171°C.

6.2.5.2.7 Recombiner

(1) Two permanently installed recombiners are located in secondary containment. Each recombiner, as shown in Figure 6.2-40, takes suction from the drywell, passes the process flow through a heating section, a reactor chamber, and a spray cooler. The gas is returned to the wetwell. PACE 10.8

(2) The recombiners are normally initiated on high levels as determined by CAMS (if hydrogen is not present, oxygen concentrations are controlled by nitrogen makeup).

6.2.5.3 Design Evaluation

The ACS is designed to maintain the containment in an inert condition except for nitrogen makeup needed to maintain a positive containment pressure and prevent air (O_2) leakage from the secondary into the primary containment.

The primary containment atmosphere will be inerted with nitrogen during normal operation of the plant. Oxygen concentration in the primary containment will be maintained below 3.5 volume percent measured on a dry basis. OPEN 20.3-6

INSERT 6.2.5.3

Following an accident, hydrogen concentration will increase due to the addition of hydrogen from the specified design-basis metal-water reaction. Hydrogen concentration will also increase due to radiolysis. Any increase in hydrogen concentration is of lesser concern because the containment is inerted. Due to dilution, additional hydrogen moves the operating point of the containment atmosphere farther from the envelope of flammability.

met. The only role of this system related to drywell purging for re-entry is to indicate when oxygen levels are high enough to start taking the samples that will be used for determining compliance with entry criteria.

6.2.5.6 Personnel Safety

Entry into a nitrogen atmosphere is particularly hazardous due to the fact that the body cannot easily detect relative changes in the nitrogen content of the air. Low oxygen causes blood chemistry changes that can lead to an automatic increase in breathing rate, leading to hyperventilation. The individual can lose consciousness in twenty to forty seconds and be totally unable to save himself.

A general procedure which outlines the critical items to be included in any procedure controlling purged drywell entry is provided below. This procedure is intended to be a framework of minimum requirements for drywell entry and for general guidance. Specific, detailed site procedures and administrative controls must be developed by each utility to meet the specific needs of each particular physical plant and administrative setup.

General Procedure Drywell Entrance Control Following De-inerting

OPEN 20.3-9

(1) Inerting and de-inerting of the drywell shall be in conformance with applicable technical specifications. INSERT 6.2.5.6

(2) Personnel access to the drywell is normally prohibited at all times when the drywell has an oxygen-deficient atmosphere, unless an emergency condition arises in which case the procedure outlined in subsection 6.2.5.6(8) should be followed.

(3) The status of the drywell atmosphere shall be posted at the drywell entrance at all times, and the entrance locked, except when cleared for entry.

(4) Suitable authorization, control and recording procedures shall be established and remain in effect throughout the entry process.

(5) Prior to initial entry, the drywell shall be purged with air in accordance with operating procedure until drywell samples indicate the following conditions are met:

(a) Oxygen: Greater than 16.5 percent content by volume.

(b) Hydrogen: Less than 14 percent of the lower limit of flammability, or a limit of 0.57 percent hydrogen by volume. (The lower flammability limit is 4.1 percent hydrogen content by volume.)

(c) Carbon Monoxide: Less than 100 ppm.

(d) Carbon Dioxide: Less than 5000 ppm.

(e) Airborne Activity: Less than applicable limits in 10CFR20, or equivalent.

(6) During the purge, drywell atmosphere samples shall be drawn from a number of locations when the drywell oxygen analyzer indicates an oxygen concentration of 16.5 percent or greater.

Samples shall be analyzed for oxygen, hydrogen, carbon monoxide, carbon dioxide and airborne activity.

When the results of two successive samples taken at least one-half hour apart are found to be within the conditions in Subsection 6.2.5.6(5), initial entry may be authorized.

(7) Criteria for entry are:

(a) The initial entry will require a minimum of two (2) persons.

(b) Initial entry will require, in addition to normal protective clothing protective equipment consisting of self-contained breathing apparatus (such as Scott Air Pack), portable air sampling and monitoring equipment, and portable radiation survey meters.

INSERT 6.2.5.3

During normal operation, nitrogen makeup and containment pressure control are accomplished using only the 50 mm supply lines. The large valves (550 mm) in the containment ventilation lines are closed and flow to the plant stack through the overpressure protection line (350 mm) is prevented by the rupture disk.

The following conditions assure that the large (550 mm) containment purge and vent lines will be isolated following a LOCA:

(1) The valves remain closed at all times during normal operation and will only be opened for inerting or de-inerting at the beginning and end of ~~an outage~~. a shutdown.
(2) The valves and piping provide redundancy such that no single failure can prevent isolation of the ~~of the~~ purge and vent lines.
(3) In the event of a LOCA, the valves receive an isolation signal.
(4) The valves fail in the closed position. If electical power to the solenoids is lost or the pneumatic pressure fails, the valves will close.

INSERT 6.2.5.6

~~Insert 20.3-9k.~~ The sizing of the piping and rupture disk for overpressure protection is discussed in ~~Paragraph~~ Subsection 19E 2.8.1.

response which describes additional tests to be conducted during the preoperational and/or startup phase.

The specific training requirements for reactor operators are discussed in Section 13.2 of the SRP which is outside the scope of the ABWR Standard Plant. See Table 1.9-1 for COL license information requirements

The additional tests specified in Appendix E of the BWROG response are contained within the initial test program described in Chapter 14. See specifically Subsections 14.2.12.1.1(3)(a), 14.2.12.1.9(3)(j), and 14.2.12.1.44(3)(a) for the relevant testing.

1A.2.5 Reactor Coolant System Vents [II.B.1]

NRC Position

Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensible gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50, *General Design Criteria*. The vent system shall be designed with sufficient redundancy that assures a low probability of inadvertent or irreversible actuation.

Each license shall provide the following information concerning the design and operation of the high point vent system.

(1) Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with acceptance criteria of 10 CFR 50.46.

(2) Submit procedures and supporting analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

Response

The capability to vent the ABWR reactor coolant system is provided by the safety relief valves and reactor coolant vent line as well as other systems. The capability of these systems and their satisfaction of Item II.B.1 is discussed below. COL 20.3-1 INSERT 1A.2.5

The ABWR design is provided with eighteen power-operated relief valves which can be manually operated from the control room to vent the reactor pressure vessel. The point of connection to the main steam lines which exits near the top of the vessel to these valves is such that accumulation of gases above that point in the vessel will not affect removal of gases from the reactor core region.

These power-operated relief valves satisfy the intent of the NRC position. Information regarding the design, qualification, power source, etc., of these valves is provided in Subsection 5.2.2.

The BWR Owners' Group position is that the requirement of single-failure criteria for prevention of inadvertent actuation of these valves, and the requirement that power be removed during normal operation, are not applicable to BWR's. These dual-purpose safety/relief valves serve an important pressure relief function in mitigating the effects of transients and concurrently provide ASME code overpressure protection via their independent safety mode of operation. Therefore, the addition of a second "block" valve to the vent lines would result in a less safe design and a violation of the code. Moreover, the inadvertent opening of a relief valve in a BWR is a design-basis event and results in a controllable transient.

In addition to these automatic (or manual) relief valves, the ABWR design includes various other means of high-point venting. Among these are:

(1) Normally closed reactor vessel head vent valves, operable from the control room, which discharge to the drywell. The reactor coolant vent line is located at the very top of the reactor vessel as shown in the nuclear boiler P&ID (Figure 5.1-3~~9~~0). This 2-inch line contains two safety-related Class 1E motor-operated valves that are operated from the control room. The location of this line permits it to vent the entire reactor core

INSERT 1A.2.5

The COL Applicant will develop plant-specific procedures to govern the operator's use of the relief mode for venting the reactor. (see Subsection 1A.3.6)

1A.2.17 Instrument for Monitoring Accident Conditions [II.F.3]

NRC Position

Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

Response

The ABWR Standard Plant is designed in accordance with Regulatory Guide 1.97, Revision 3. A detailed assessment of the Regulatory Guide, including the list of instruments, is found in Section 7.5.

1A.2.18 Safety-Related Valve Position Indication [II.K.1(5)]

NRC Position

(1) Review all valve positions and positioning requirements and positive controls and all related test and maintenance procedures to assure proper ESF functioning, if required.

(2) Verify that AFW valves are in open position.

Response

(1) The ABWR Standard Plant is equipped with status monitoring that satisfies the requirements of Regulatory Guide 1.47. See Subsection 7.1.2 for detailed information on the status monitoring equipment and capability provided in the ABWR Standard Plant design.

In addition to the status monitoring, the COL applicant plant specific procedures (see Subsection 1A.3.2) will assure that independent verification of system line-ups is applied to valve and electrical line-ups for all safety-related equipment, to surveillance procedures, and to restoration following maintenance. Through these procedures, approval will be required for the performance of surveillance tests and maintenance, including equipment removal from service and return to service.

COL 20.3.1-1

(2) This requirement is not applicable to the ABWR. It applies only to Babcock & Wilcox designed reactors.

and to comply with IE Bulletin 79-08.

1A.2.19 Review and Modify Procedures for Removing Safety-Related Systems From Service [II.K.1(10)]

NRC Position

Review and modify (as required) procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known. INSERT 1A.2.19 COL 20.3.1-2

Response

See Subsection 1A.3.2 for COL license information requirements.

1A.2.20 Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable [II.K.1(22)]

NRC Position

For boiling water reactors, describe the automatic and manual actions necessary for proper functioning of the auxiliary heat removal systems that are used when the main feedwater system is not operable (see Bulletin 79-08, item 3).

Response

If the main feedwater system is not operable, a reactor scram will be automatically initiated when reactor water level falls to Level 3. The operator can then manually initiate the RCIC system from the main control room, or the system will be automatically initiated as hereinafter described. Reactor water level will continue to decrease due to boil-off until the low- low level setpoint, Level 2, is reached. At this point, the reactor core isolation cooling (RCIC) system will be automatically initiated to supply makeup water to the RPV. This system will continue automatic injection until the reactor water level reaches Level 8, at which time the RCIC steam supply valve is closed.

In the nonaccident case, the RCIC system is normally the only makeup system utilized to furnish subsequent makeup water to the RPV. When level reaches Level 2 again due to loss of inventory through the main steam relief valves or to the main condenser, the RCIC system automatically restarts as described in Subsection 1A.2.22. This system then

INSERT 1A.2.19 The COL applicant must verify the operability of safety-related systems after performing maintenance or tests as part of the test to restore a system to service.

and engineered safeguards systems.

(4) Fuel-zone, water-level range: This range used for its RPV taps the elevation above the main steam outlet nozzle and the taps just above the internal recirculation pump (RIP) deck. The zero of the instrument is the bottom of the active fuel and the instruments are calibrated to be accurate at 0 psig and saturated condition. The water-level measurement design is the condensate reference type, is not density compensated, and uses differential pressure devices as its primary elements. These instruments provide input to water-level indication only.

There are common condensate reference chambers for the narrow-range; wide-range; and fuel-zone, water-level ranges.

The elevation drop from RPV penetration to the drywell penetration is uniform for the narrow range and wide range water-level instrument lines in order to minimize the change in water-level with changes in drywell temperature.

Reactor water-level instrumentation that initiates safety systems and engineered safeguards is shown in Figure 5.1-3.

1A.2.21.1 Failure of PORV or Safety to Close [II.K.3.(3)]

NRC Position

COL 20.3.1-3

Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report. This requirement is to be met before fuel load.

Response

See Subsection 1A.3.4 for COL license information requirements.

1A.2.22 Separation of HPCI AND RCIC System Initiation Levels [II.K.3(13)]

NRC Position

Currently, the reactor core isolation cooling (RCIC) system and the high-pressure coolant injection (HPCI) systems both initiate on the same low-water-level signal and both isolate on the same high-water-level signal. The HPCI system will restart on low water level but the RCIC system will not. The RCIC system is a low-flow system when compared to the HPCI system. The initiation levels of the HPCI and RCIC system should be separated so that the RCIC system initiates at a higher water level than the HPCI system. Further, the initiation logic of the RCIC system should be modified so that the RCIC system will restart on low water level. These changes have the potential to reduce the number of challenges to the HPCI system and could result in less stress on the vessel from cold water injection. Analyses should be performed to evaluate these changes. The analysis should be submitted to the NRC staff and changes should be implemented if justified by the analysis.

Response

The ABWR Standard Plant design is consistent with this position. The high pressure core flooder (HPCF) system is initiated at Level 1-1/2, and the RCIC system is initiated at Level 2. At Level 8, the injection valves for the HPCF and the RCIC steam supply and injection valves will automatically close in order to prevent water from entering the main steam lines.

In the unlikely event that a subsequent low level recurs, the RCIC steam supply and injection valves will automatically reopen to allow continued flooding of the vessel. The HPCF injection valves will also automatically reopen unless the operator previously closed them manually. Refer to Subsections 7.3.1.1.1.1 (HPCF) and 7.3.1.1.1.3 (RCIC) for additional details regarding system initiation and isolation logic.

1A.2.23 Modify Break-Detection Logic to Prevent Spurious Isolation of HPCI And RCIC Systems [II.K.3(15)]

NRC Position

The high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems use differential pressure sensors on elbow taps in the steam lines to their turbine drives to detect and isolate pipe breaks in the systems. The pipe-break-detection circuitry has resulted in spurious isolation of the HPCI and RCIC systems due to the pressure spike which accompanies startup of the systems. The pipe-break-detection circuitry should be modified to

differential pressure signals which isolate the RCIC turbine are processed through the leak detection and isolation system (LDS). Spurious trips are avoided because the RCIC has a bypass start system controlled by valves F037 and F045 (see Figure 5.4-8, RCIC P&ID).

On receipt of RCIC start signals, bypass valve F045 opens to pressurize the line downstream and accelerate the turbine. The bypass line via F045 is small (1-inch) and naturally limits the initial flow surge such that a differential pressure spike in the upstream pipe will not occur.

After a predetermined delay (approximately 5-10 seconds), steam supply valve F037 opens to admit full steam flow to the turbine. At this stage, the line downstream is already pressurized. Thus, it is highly unlikely that a differential pressure spike could occur during any phase of the normal start-up process. This process is tested in Subsection 14.2.12.2.22.

COL 20.3.1-4

1A.2.24 Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification [II.K.3.(16)]

NRC Position

The record of relief-valve failures to close for all boiling water reactors (BWRs) in the past 3 years of plant operation is appproximately 30 in 73 reactor-years (0.41 failures per reactor-year). This has demonstrated that the failure of a relief valve to close would be the most likely cause of a small-break loss-of-coolant accident (LOCA). The high failure rate is the result of a high relief-valve challenge rate and a relatively high failure rate per challenge (0.16 failures per challenge). Typically, five valves are challenged in each event. This results in an equivalent failure rate per challenge of 0.03. The challenge and failure rates can be reduced in the following ways:

(1) Additional anticipatory scram on loss of feedwater,

(2) Revised relief-valve actuation setpoints,

(3) Increased emergency core cooling (ECC) flow,

(4) Lower operating pressures,

(5) Earlier initiation of ECC systems,

(6) Heat removal through emergency condensers,

(7) Offset valve setpoints to open fewer valves per challenge,

(8) Installation of additional relief valves with a block or isolation-valve feature to eliminate opening of the safety/relief valves (SRV's), consistent with the ASME Code,

(9) Increasing the high steam line flow setpoint for main steam line isolation valve (MSIV) closure,

(10) Lowering the pressure setpoint for MSIV Closure,

(11) Reducing the testing frequency of the MSIV's,

(12) More stringent valve leakage criteria, and

(13) Early removal of leaking valves.

An investigation of the feasibility and constraints of reducing challenges to the relief valves by use of the aforementioned methods should be conducted. Other methods should also be included in the feasibility study. Those changes which are shown to reduce relief-valve challenges without compromising the performance of the relief valves or other systems should be implemented. Challenges to the relief valves should be reduced substantially (by an order of magnitude).

Response

General Electric and the BWR Owners' Group reponded to this requirement in Reference 6. This response, which was based on a review of existing operating information on the challange rate of relief valves, concluded that the BWR/6 product line had already achieved the "order of magnatude" level of reduction in SRV challange rate. The ABWR relief valve system also has similar design features which also reduce the SRV challenge rate. With regard to inadvertently opened relief valves (IORV), the BWR/6 plnt design evaluated for the Owners' Group report reflected a reduced level of IORC compared with previous design because of

1A.2.25 Report on Outages of Emergency Core-Cooling Systems Licensee Report and Proposed Technical Specification Changes [II.K.3(17)]

NRC Position

Several components of the emergency core cooling (ECC) systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last 5 years of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation).

Clarification

The present technical specifications contain limits on allowable outage times for ECC systems and components. However, there are no cumulative outage time limitations on these same systems. It is possible that ECC equipment could meet present technical specification requirements but have a high unavailability because of frequent outages within the allowable technical specifications.

The licensees should submit a report detailing outage dates and length of outages for all ECC systems for the last 5 years of operation, including causes of the outages. This report will provide the staff with a quantification of historical unreliability due to test and maintenance outages, which will be requirements in the technical specifications.

Based on the above guidance and clarification, a detailed report should be submitted. The report should contain (1) outage dates and duration of outages; (2) causes of the outage; (3) ECC systems or components involved in the outage; and (4) corrective action taken. Tests and maintenance outages should be included in the above listings which are to cover the last 5 years of operation. The licensee should propose changes to improve the availability of ECC equipment, if needed.

Applicants for an operating license shall establish a plan to meet these requirements.

Response

See Subsection 1A.3.5 for Interface Requirements

COL Licensing Information

1A.2.26 Modification of Automatic De-Pressurization System Logic - Feasibility for Increased Diversity for Some Event Sequences [II.K.3(18)]

NRC Position

The automatic depressurization system (ADS) actuation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment study is required to determine the optimum approach. One possible scheme that should be considered is ADS actuation on low reactor-vessel water level provided no high-pressure coolant injection (HPCI) or high-pressure coolant system (HPCS) flow exists and a low-pressure emergency core cooling (ECC) system is running. This logic would complement, not replace, the existing ADS actuation logic.

Response

An 8 minute high drywell pressure bypass timer has been added to the ADS initiation logic to address TMI action item II.K.3.18. This timer will initiate on a Low Water Level 1 signal. When it times out, it bypasses the need for a high drywell signal to initiate the standard ADS initiation logic.

For all LOCAs inside the containment, a high drywell signal will be present and ADS will actuate 29 seconds after a Low Water Level 1 signal is reached. All LOCAs outside the containment become rapidly isolated and any one of the three high pressure ECCS can control the water level. The high drywell pressure bypass timer in the ADS initiation logic will only affect the LOCA response if all high pressure ECCS fail following a break outside the containment. For this case the ADS will automatically initiate within 509 seconds (8 minute timer plus 29 second standard ADS logic delay) following a Low Water Level 1 signal.

1A.3 COL LICENSE INFORMATION

1A.3.1 Emergency Procedures and Emergency Procedures Training Program

Emergency procedures, developed from the emergency procedures guidelines, shall be provided and implemented prior to fuel loading. (See Subsection 1A.2.1).

1A.3.2 Review and Modify Procedures for Removing Safety-Related Systems From Service

Procedures shall be reviewed and modified (as required) for removing safety-related systems from service (and restoring to service) to assure operability status is known. (See Subsections 1A.2.18 and 19).

1A.3.3 In-Plant Radiation Monitoring

Equipment and training procedures shall be provided for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during the accident. (See Subsection 1A.2.35).

1A.3.4 Reporting Failures of Reactor System Relief Valves

Failures of reactor system relief valves shall be reported in the annual report to the NRC. (See Subsection 1A.2.21.1).

1A.3.5 Report on ECCS Outages

Starting from the date of commercial operations, an annual report should be submitted which includes instance of emergency core cooling system unavailability because of component failure, maintenance outage (both forced or planned), or testing, the following information shall be collected:

COL 20.3.1-5

(1) Outage date
(2) Duration of outage
(3) Cause of outage
(4) Emergency core cooling system or component involved
(5) Corrective action taken

The above information shall be assembled into a report, which will also include a discussion of any changes, proposed or implemented, deemed appropriate, to improve the availability of the emergency core cooling equipment. (See Subsection 1A.2.2.5).

1A.3.6 Procedures for Reactor Venting

Procedures shall be developed for the operators use of the reactor vents. (See Subsection 1A.2.5)

Response

This requirement is not applicable to the ABWR. It applies only to PWR-type reactors.

19A.2.26 Isolation Dependability [Item (2) (xiv)]

NRC Position

Provide containment isolation systems that: [II.E.4.2]

(A) Ensure all non-essential systems are isolated automatically by the containment isolation system,

(B) For each non-essential penetration (except instrument lines) have two isolation barriers in series,

(C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal,

(D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation,

(E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

Response

This item is addressed in Subsection 1A.2.14.

19A.2.27 Purging [Item (2) (xv)]

NRC Position

Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. [II.E.4.4]

Response

The ABWR primary containment vessel (PCV) operates with an inert atmosphere. During normal operation, all large valves in containment ventilation lines are closed. Only small, 2", nitrogen-makeup valves are opened during power operation. These are air-operated valves with rapid closure times, presenting little opportunity for substantial releases from the PCV in the event of a transient requiring containment isolation. Note that under the technical specifications, containment inerting and purging with the larger ventilation lines is permitted during power operation above 15% for limited periods at either end of the operating cycle. The process of purging the containment with air also serves to remove any potential activity for ALARA considerations prior to actual personnel entry into the PCV.

The large ventilation valves will be tested regularly and after any valve maintenance to assure that closing times are within the limits assured in the radiological design basis. See Subsection 19A.3.3 for COL license information. These tests are part of the inservice test program detailed in Subsection 3.9. COL 20.3-2

19A.2.28 Design Evaluator [Item (2) (xvi)]

NRC Position

Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe over cooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only.) [II.E.5.1]

Response

This requirement is not applicable to the ABWR. It applies only to PWR-type (B&W designed) reactors.

19A.2.29 Additional Accident Monitoring Instrumentation [Item (2) (xvii)]

NRC Position

Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. [II.F.1]

3.9.7 COL License Information

3.9.7.1 Reactor Internals Vibration Analysis, Measurement and Inspection Program

The first COL applicant will provide, at the time of application, the results of the vibration assessment program for the ABWR prototype internals. These results will include the following information specified in Regulatory Guide 1.20.

R. G. 1.20	Subject
C.2.1	Vibration Analysis Program
C.2.2	Vibration Measurement Program
C.2.3	Inspection Program
C.2.4	Documentation of Results

NRC review and approval of the above information on the first COL applicant's docket will complete the vibration assessment program requirements for prototype reactor internals.

In addition to the information tabulated above, the first COL applicant will provide the information on the schedules in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals.

Subsequent COL applicants need only provide the information on the schedules in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals. (See Subsection 3.9.2.4).

3.9.7.2 ASME Class 2 or 3 or Quality Group D Components with 60 Year Design Life

COL applicants will identify ASME Class 2 or 3 or Quality Group D components that are subjected to cyclic loadings, including operating vibration loads and thermal transients effects, of a magnitude and/or duration so severe the 60 year design life can not be assured by required Code calculations and, if similar designs have not already been evaluated, either provide an appropriate analysis to demonstrate the required design life or provide designs to mitigate the magnitude or duration of the cyclic loads. (See Subsection 3.9.3.1.)

3.9.7.3 Pump and Valve Inservice Testing Program

COL 20.3-2

COL applicants will provide a plan for the detailed pump and valve inservice testing and inspection program. This plan will

(1) Include baseline pre-service testing to support the periodic in-service testing of the components required by technical specifications. Provisions are included to disassemble and inspect the pump, check valves, and MOVs within the Code and safety-related classification as necessary, depending on test results. (See Subsections 3.9.6, 3.9.6.1, 3.9.6.2.1 and 3.9.6.2.2)

(2) Provide a study to determine the optimal frequency for valve stroking during inservice testing. (See Subsection 3.9.6.2.2)

(3) Address the concerns and issues identified in Generic Letter 89-10; specifically the method of assessment of the loads, the method of sizing the actuators, and the setting of the torque and limit switches. (See Subsection 3.9.6.2.2)

3.9.7.4 Audit of Design Specification and Design Reports

COL applicants will make available to the NRC staff design specification and design reports required by ASME Code for vessels, pumps, valves and piping systems for the purpose of audit. (See Subsection 3.9.3.1)

3.9.8 References

1. *BWR Fuel Channel Mechanical Design and Deflection*, NEDE-21354-P, September 1976.

2. *BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings*, NEDE-21175-P, November 1976.

3. NEDE-24057-P (Class III) and NEDE-24057 (Class I) Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants,

Table 3.9-8 (Continued)

IN-SERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

T22 Standby Gas Treatment System Valves

No.	Qty	Description (h)(l)	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig. (g)
F012	2	Filter train DOP sampling line valve downstream of after HEPA	3	B	P		E1	6.5-1(2,3)
F014	2	STGS sample line valve	3	B	P		E1	6.5-1(2,3)
F015	2	PRM discharge to stack valve	3	B	P		E1	6.5-1(2,3)
F500	2	Filter unit vent line valve	3	B	P		E1	6.5-1(2,3)
F501	2	Filter unit drain line valve	3	B	P		E1	6.5-1(2,3)
F504	2	Filter unit vent line valve	3	B	P		E1	6.5-1(2,3)
F505	2	Exhaust fan vent line valve	3	B	P		E1	6.5-1(2,3)
F506	2	Filter train vent line valve	3	B	P		E1	6.5-1(2,3)
F507	2	Filter train vent line valve	3	B	P		E1	6.5-1(2,3)
F508	2	Filter train vent line valve	3	B	P		E1	6.5-1(2,3)
F509	2	Filter train vent line valve	3	B	P		E1	6.5-1(2,3)
F510	2	Filter train vent line valve	3	B	P		E1	6.5-1(2,3)
F511	2	Exhaust stack drain line valve	3	B	P		E1	6.5-1(2,3)
F700	2	Filter unit demister dp instrument line valve	3	B	P		E1	6.5-1(2,3)
F701	2	Filter unit demister dp instrument line valve	3	B	P		E1	6.5-1(2,3)
F705	2	Filter train prefilter dp instrument line valve	3	B	P		E1	6.5-1(2,3)
F706	2	Filter train prefilterdp instrument line valve	3	B	P		E1	6.5-1(2,3)
F707	2	Filter train preHEPA dp instrument line valve	3	B	P		E1	6.5-1(2,3)
F708	2	Filter train preHEPA dp instrument line valve	3	B	P		E1	6.5-1(2,3)
F709	2	Fliter train charcoal adsorber dp inst. line vlv	3	B	P		E1	6.5-1(2,3)
F710	2	Filter train charcoal adsorber dp inst line vlv	3	B	P		E1	6.5-1(2,3)
F711	2	Filter train after HEPA dp inst line valve	3	B	P		E1	6.5-1(2,3)
F712	2	Filter train after HEPA dp inst line valve	3	B	P		E1	6.5-1(2,3)
F713	2	Filter train exhaust flow instrument line valve	3	B	P		E1	6.5-1(2,3)
F714	2	Filter train exhaust flow instrument line valve	3	B	P		E1	6.5-1(2,3)

T31 Atmospheric Control System Valves

F001	1	N2 supply line from Reactor Building HVAC	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39(1)
F002	1	N2 supply line to drywell inboard containment isoaltion valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39(1)
F003	1	N2 supply line to wetwell inboard containment isoaltion valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39(1)
F004	1	Containment atmosphere exhaust line from drywell isoaltion valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39(1)
F005	1	Drywell atmosphere exhaust line valve T31-F004 bypass line	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39(1)
F006	1	Containment atmosphere exhaust line form wetwell isolation valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39(1)
F007	1	Wetwell overpressure line valve	2	A	P	L,P	2 yrs	6.2-39(1)

Table 3.9-8 (Continued)

IN-SERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

T31 Atmospheric Control System Valves

No.	Qty	Description (h)(i)	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig. (g)
F008	1	Containment atmosphere exhaust line to SGTS	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39(1)
F009	1	Containment atmosphere exhaust line to R/B HVAC	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39(1)
F010	1	Drywell overpressure line valve	2	A	P	L,P	2 yrs	6.2-39(1)
F025	1	N2 supply line from K-5 outboard containment isolation valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39(1)
F039	1	N2 supply line from K-5 outboard containment isolation valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39(1)
F040	1	N2 supply line from K-5 to drywell inboard isolation valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39(1)
F041	1	N2 supply line from K-5 to wetwell inboard isolation valve	2	A	I,A	L,P S	2 yrs 3 mo	6.2-39(1)
F044	8	Drywell/wetwell vacuum breaker valve	2	C	A	P R	RO E3	6.2-39(2)
F050	1	N2 supply line to drywell test line valve	2	B	P		E1	6.2-39(1)
F051	1	Containment atmosphere exhaust line test line valve	2	B	P		E1	6.2-39(1)
F054	1	Drywell personnel air lock hatch test line valve	2	B	P		E1	6.2-39(2)
F055	1	N2 supply line from test line valve	2	B	P		E1	6.2-39(1)
F056	1	Wetwell personnel air lock hatch test line valve	2	B	P		E1	6.2-39(2)
F700	1	N2 supply line to drywell FE upstream instrument line	2	B	P		E1	6.2-39(1)
F701	1	N2 supply line to drywell FE downstream instrument line	2	B	P		E1	6.2-39(1)
F702	1	N2 supply line to wetwell FE upstream instrument line	2	B	P		E1	6.2-39(1)
F703	1	N2 supply line to wetwell FE downstream instrument line	2	B	P		E1	6.2-39(1)
F720	8	DW/WW vacuum breaker valve N2 supply line isolation valve	2	A	I,P	L	RO	6.2-39(2)
F730	1	Drywell pressure instrument line isolation valve	2	B	P		E1	6.2-39(2)
F731	1	Drywell pressure instrument line solenoid isolation valve	2	A	I,P	L,P	RO	6.2-39(2)
F732	2	Drywell pressure instrument line valve	2	B	P		E1	6.2-39(2)
F733	2	Drywell pressure instrument line solenoid isolation valve	2	A	I,P	L,P	RO	6.2-39(2)
F734	4	Drywell pressure instrument line for NBS valve	2	B	P		E1	6.2-39(2)