GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

March 17, 1993 Docket No. STN 52-001

Chet Poslusny, Senior Project Manager
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 DFSER Outstanding Items for Chapters 3, 4, 6, 10, 14 and 20**

Dear Chet:

Enclosed are SSAR markups addressing the following DFSER outstanding items:

Open Items	Confirmatory Items	COL Action Items
6.2.4.1-1 (1/22/93*)	6.5.1-1 (Amendment 24*)	3.9.3.2-1
6.5.1-2 (1/22/93*)	7.7.1.5-1	3.9.3.3-1
20.3-2		7.1.4-1
20.3-3		10.3.1-1
20.3-4		14.1.3.3.7.2-1
		14.1.3.3.4.2-1
		14.1.3.3.2.1-1
		14.1.3.3.9.2-1
		14.1.3.3.6.4-2
		14.1.3.3.6.4-1
		14.1.3.3.4.3-1
		14.1.3.3.5.10-1

*Amplification of a previous markup or amendment

JF93-60

In addition to the above, GE recommends that the first two sentences of the fifth paragraph of DFSER Page 14-18 be replaced with the following:

> "The effect of pipe support stiffness on the piping response shall be considered in the analytical model. Supports shall be modeled in accordance with the SSAR. If supports are not modeled as stated in the SSAR, justification will be provided to validate the stiffness values used in the piping model."

Finally, GE recommends that the following COL Action Items be deleted:

Item	Rationale
14.1.3.3.6.12-1	COL Action Item 14.1.3.3.2.1-1 embraces the Staff-endorsed version of NF incorporating N-690.
14.1.3.3.5.1-1	SSAR will not include an option for the COL applicant to generate site-specific amplified building response spectra.

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

cc: Norman Fletcher (DOE)
Bernie Genetti (GE)
Maryann Herzog (GE)

1A.2.8 Rule Making Proceeding or Degraded Core Accidents [II.B.8]

Response to this TMI action plan item is addressed in Appendix 19A.

1A.2.9 Coolant System Valves-Testing Requirements [II.D.1]

NRC Position

Pressurized-water reactor and boiling-water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents.

Response

The ABWR safety/relief valve (SRV) is postulated to discharge steam only, not liquid or two phase flow under expected operating conditions for design-basis transients and accidents.

A generic test program was conducted through the BWR Owners Group (Reference 7) to satisfy the discharge of steam. These steam discharge test results will be used as the qualification basis for plant specific SRV models and discharge piping that are sufficiently similar to those reported in Reference 8. Plant specific SRV models and discharge piping that are not similar will be tested in accordance with NUREG-0737 requirements. See Subsection 1A.3.6 for COL license information.

The ABWR system logic for response to high water level conditions is described in Subsection 7.3.1.1.1.1(3) and is considered to be sufficiently redundant that the probability of steam line flooding by ECCS is extremely low. There is no high drywell pressure signal that would inhibit this logic system.

In the ABWR design, each of three RHR shutdown cooling lines has its own separate containment penetration and its own separate source of suction from the reactor vessel. Alternate shutdown using the SRV is therefore not required for ABWR in order to meet single failure rules. Hence, the ABWR does not require SRV testing with liquid under low pressure conditions associated with this event as required in past BWRs.

1A.2.10 Relief and Safety Valve Position Indication [II.D.3]

NRC Position

Reactor coolant system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe.

Response

The ABWR Standard Plant safety relief valves are equipped with linear variable differential transformers (LVDT's) which are qualified as Class 1E components. These LVDT's are mounted on the valve operators and are highly reliable sensors for monitoring valve position.

In addition, the downstream pipe from each valve line is equipped with thermocouples which signal the annunciator and the plant process computer when the temperature in the tailpipe exceeds the predetermined setpoint.

These sensors are shown on Figure 5.1-3 (Nuclear Boiler System P&ID).

1A.2.11 Systems Reliability [II.E.3.2]

This TMI action plan item superseded by USI A-45. USI A-45 is addressed in Appendix 19B.

1A.2.12 Coordinated Study of Shutdown Heat Removal Requirements [II.E.3.3]

This TMI action plan item superseded by USI A-45. USI A-45 is addressed in Appendix 19B.

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.2.3.21.1).

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:

(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 Testing of SRV and Discharge Piping

The COL applicant will confirm that any SRVs or discharge piping installed that is not similar to those that have been tested will be tested in accordance with Subsection 1A.2.9.

(c) The assemblies are subjected to a single pressure test at a pressure not less than its design pressure.

(d) The assemblies do not prevent the access required to conduct the inservice examination specified in item (7).

(7) A 100% volumetric inservice examination of all pipe welds would be conducted during each inspection interval as defined in IWA-2400, ASME Code, Section XI. See Subsection 3.6.5.3 for COL license information.

3.6.2.1.4.3 ASME Code Section Class 1 Piping In Areas Other Than Containment Penetration

With the exception of those portions of piping identified in Subsection 3.6.2.1.4.2, breaks in ASME Code, Section III, Class 1 piping are postulated at the following locations in each piping and branch run:

(a) At terminal ends*

(b) At intermediate locations where the maximum stress range as calculated by Eq. (10) exceeds 2.4 Sm.

If the calculated maximum stress range of Eq.(10) exceeds 2.4 Sm, the stress range calculated by both Eq.(12) and Eq.(13) in Paragraph NB-3653 should meet the limit of 2.4 Sm.

(c) At intermediate locations where the cumulative usage factor exceeds 0.1.

As a result of piping re-analysis due to differences between the design configuration and the as-built configuration, the highest stress or cumulative usage factor locations may be shifted; however, the initially determined intermediate break locations need not be changed unless one of the following conditions exists:

(i) The dynamic effects from the new (as-built) intermediate break locations are not mitigated by the original pipe whip restraints and jet shields.

(ii) A change is required in pipe parameters such as major differences in pipe size, wall thickness, and routing.

3.6.2.1.4.4 ASME Code Section III Class 2 and 3 Piping in Areas Other Than Containment Penetration

With the exception of those portions of piping indentified in Subsection 3.6.2.1.4.2, breaks in ASME Codes, Section III, Class 2 and 3 piping are postulated at the following locations in those portions of each piping and branch run:

(a) At terminal ends (see Subsection 3.6.2.1.4.3, Paragraph (a))

(b) At intermediate locations selected by one of the following criteria:

(i) At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve. Where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping run adjacent to the protective structure.

(ii) At each location where stresses calculated (see Subsection 3.6.2.1.4.2, Paragraph (1)(d)) by the sum of Eqs. (9) and (10) in NC/ND-3653, ASME Code, Section III, exceed 0.8 times the sum of the stress limits given in NC/ND-3653.

As a result of piping re-analysis due to differences between the design configuration and the as-built configuration, the highest stress

* *Extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping motion and thermal expansion. A branch connection to a main piping run is a terminal end of the branch run, except where the branch run is classified as part of a main run in the stress analysis and is shown to have a significant effect on the main run behavior. In piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve) a terminal end of such runs is the piping connection to this closed valve.*

3.6.5 Leak-Before-Break Analysis Report

As required by Reference 1, and LBB analysis report shall be prepared for the piping systems proposed for exclusion from analysis for the dynamic effects due to failure of piping failure. The report shall be prepared in accrodance with the guidelines presented in Appendix 3E and Submitted by the COL applicant to the NRC for approval. (See Subsection 3.6.3).

3.6.6 References

Attachment A

1. *Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Rupture,* Federal Register, Volume 52, No. 207, Rules and Regulations, Pages 41288 to 41295, October 27, 1987

2. *RELAP 3, A Computer Program for Reactor Blowdown Analysis,* IN-1321, issued June 1970, Reactor Technology TID-4500.

3. ANSI/ANS-58.2, *Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture.*

4. *Standard Review Plan; Public Comments Solicited,* Federal Register, Volume 52, No. 167, Notices, Pages 32626 to 32633, August 28, 1987.

5. NUREG-1061, Volume3, *Evaluation of Potential for Pipe Breaks, Report of the U.S. NRC Piping Review Committee,* November 1984.

6. Mehta, H. S., Patel, N.T. and Ranganath, S., *Application of the Leak-Before-Break Approach to BWR Piping,* Report NP-4991, Electric Power Research Institute, Palo Alto, CA, December 1986.

COL Action Item No. 14.1.3.3.7.2-1

ATTACHMENT A

3.6.5.3 Inservice Inspection of Piping in Containment Penetration Areas

The COL ~~a~~Applicant shall perform a 100 percent volumetric examination of circumferential and longitudinal pipe welds for those portions of piping within the break-exclusion region. The examination shall be performed in accordance with the requirements specified in Subsection 3.6.2.1.4.2(7).

excitation in each of the three axes is considered to act simultaneously. The excitations are combined by the SRSS method.

3.7.3.8.1.4 Flexible Subsystems

If the piping subsystem has more than two supports, it cannot be considered a rigid body and must be modeled as a multi-degree-of-freedom subsystem.

The subsystem is modeled as discussed in Subsection 3.7.3.3.1 in sufficient detail (i.e., number of mass points) to ensure that the lowest natural frequency between mass points is greater than 33 Hz. The mathematical model is analyzed using a time-history analysis technique or a response spectrum analysis approach. After the natural frequencies of the subsystem are obtained, a stress analysis is performed using the inertia forces and equivalent static loads obtained from the dynamic analysis for each mode.

In a response spectrum dynamic analysis, modal responses are combined as described in Subsection 3.7.3.7. In a response spectrum or time-history dynamic analysis, responses due to the three orthogonal components of seismic excitation are combined as described in Subsection 3.7.3.6.

9 See Subsection 3.7.6.1 for COL license information.

3.7.3.8.1.5 Static Analysis

A static analysis is performed in lieu of a dynamic analysis by applying the following forces at the concentrated mass locations (nodes) of the analytical model of the piping system:

(1) horizontal static load, $F_h = C_h W$, in one of the horizontal principal directions;

(2) equal static load, F_h, in the other horizontal principal direction; and

(3) vertical static load, $F_v = C_v W$;

where

C_h, C_v = multipliers of the gravity acceleration, g, determined from the horizontal and vertical floor response spectrum curves, respectively. (They are functions of the period and the appropriate damping of the piping system); and

W = weight at node points of the analytical model.

(1) ASME Code Case N-411-1 damping is not used.

(2) A support group is defined by supports which have the same time-history input. This usually means all supports located on the same floor, or portions of a floor, of a structure.

(3) The responses due to motions of supports in two or more different groups are combined by the SRSS procedure.

In lieu of the response spectrum analysis, the time history method of analysis subjected to distinct support motions may be used for multi-supported systems.

3.7.3.8.2 NSSS Piping Subsystems

3.7.3.8.2.1 Dynamic Analysis

As described in Subsection 3.7.3.3.1, pipe line is idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping subsystem is determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as changes in stiffness due to curved members.

Next, the mode shapes and the undamped natural frequencies are obtained. The dynamic response of the subsystem is usually calculated by using the response spectrum method of analysis. When the connected equipment is supported at more than two points located at different elevations in the building, the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternatively, the multiple excitation analysis methods may be used where acceleration time histories or response spectra are applied at all the equipment and piping attachment points.

In a response spectrum dynamic analysis, modal responses are combined as described in Subsection 3.7.3.7. In a independent support motion response spectrum analysis, group responses are combined as described in Subsection 3.7.3.8.1.10. In response spectrum or time-history dynamic analysis, responses due to the three orthogonal components of seismic excitation are combined as described in Subsection 3.7.3.6.

¶ See subsections 3.7.6.1 for COL license information.

the seismic switch location.

The peak acceleration level experienced by the reactor building basemat is available immediately following the earthquake. This is obtained by playing back the recorded THA data from the basemat location and reading the peak value from a strip chart recorder.

Significant response spectra from the reactor building basemat are available immediately following an earthquake for comparison with the OBE and SSE response spectra.

3.7.4.4 Comparison of Measured and Predicted Responses

Initial determination of the earthquake level is performed immediately after the earthquake by comparing the measured response spectra from the reactor building basemat with the OBE and SSE response spectra for the corresponding location. If the measured spectra exceed the OBE response spectra, the plant is shut down and a detailed analysis of the earthquake motion is undertaken.

After any earthquake, the data from all seismic recorders and recording instruments are retrieved. When the OBE has been exceeded, the data from these instruments are analyzed to obtain the seismic accelerations experienced at the location of major Seismic Category I structures and equipment. The measured response from the time-history accelerographs, peak-recording accelerographs, and response spectrum recorders are used to determine the response spectra at the location of each Seismic Category I structure and system. These spectra are compared with those used in the design to determine whether the structure or system is still adequate for future use. Peak-recording accelerographs mounted on equipment are used to determine whether the design limitation of that specific equipment has been exceeded.

The theoretical structural response and measured structural responses are compared to assess the degree of conservatism in the analytical predictions. Seismic levels are established to determine whether the plant can be brought back on line. The criteria consider system design and dynamic analysis in establishing the acceptable levels for continued operation.

3.7.4.5 In-Service Surveillance

Each of the seismic instruments will be demonstrated operable by the performance of the channel check, channel calibration, and channel functional test operations at the intervals specified in Table 3.7-9.

3.7.5 Seismic Parameters

The design basis horizontal g value is 0.3g for SSE. This is the maximum free-field ground acceleration at the site as measured at the existing grade level near the ABWR. The response spectra are presented in Subsection 3.7.1. The range of site parameters used to establish the design basis seismic parameters is presented in Appendix 3A.

Attachment B

3.7.~~6~~7 References

1. General Electric Company BWR/6-238 *Standard Safety Analysis Report (GESSAR)*, Docket No. STN 50-447, November 7, 1975.

2. E. H. Vanmarcke and C. A. Cornell, *Seismic Risk and Design Response Spectra*, ASCE Specialty Conference on Safety and Reliability of Metal Structures, Pittsburgh, Pennsylvania, November 1972.

3. NUREG-0800, *Standard Review Plan*, Section 3.7.1.

4. L. K. Liu, *Seismic Analysis of the Boiling Water Reactor*, symposium on seismic analysis of pressure vessel and piping components, First National Congress on Pressure Vessel and Piping, San Francisco, California, May 1971.

ATTACHMENT B

COL 14.1.3.3.4.2-1

3.7.6 COL License Information

3.7.6.1 **Piping Analysis, Modeling of Piping Supports**

The COL ~~A~~pplicant shall provide justification for methods used other than those described in Subsection 3.7.3.3.1.6 for determining pipe support stiffnesses used in the piping analysis. The justification should include verification that the pipe support stiffness values are representative of the types of supports used in the piping system. The alternative approach used to determine pipe support stiffness values and its bases should be submitted to the NRC staff for review and approval before its use. (see Subsection 3.7.3.8.1.4 and 3.7.3.8.2.1)

The results of the data analyses, vibration amplitudes, natural frequencies, and mode shapes are then compared to those obtained from the theoretical analysis.

Such comparisons provide the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained from previous vibration tests has been utilized in the generation of the dynamic models for seismic and loss of coolant accident (LOCA) analyses for this plant. The models used for this plant are similar to those used for the vibration analysis of earlier prototype BWR plants.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

COL 14.1.3.3.5.10-1

This section delineates the criteria for selection and definition of design limits and loading combination associated with normal operation, postulated accidents, and specified seismic and other reactor building vibration (RBV) events for the design of safety-related ASME Code components (except containment components which are discussed in Section 3.8).

This section discusses the ASME Class 1, 2, and 3 equipment and associated pressure retaining parts and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. A discussion of major equipment is included on a component-by-component basis to provide examples. Design transients and dynamic loading for ASME Class 1, 2, and 3 equipment are covered in Subsection 3.9.1.1. Seismic-related loads and dynamic analyses are discussed in Section 3.7. The suppression pool-related RBV loads are described in Appendix 3B. Table 3.9-2 presents the combinations of dynamic events to be considered for the design and analysis of all ABWR ASME Code Class 1, 2, and 3 components, component supports, core support structures and equipment. Specific loading combinations considered for evaluation of each specific equipment are derived from Table 3.9-2 and are contained in the design specifications and/or design reports of the respective equipment. See Subsections 3.9.7.4, 3.9.7.5, 3.9.7.6 and 3.9.7.8 for COL license information requirements.

Thermal stratification of fluids in a piping system is one of the specific operating conditions that is included in the loads and load combinations that are contained in the piping design specifications and design reports. It is known stratification can occur in the feedwater piping during plant startup and when the plant is in hot standby conditions following scram (see Subsection 3.9.2.1.3). If, during design or startup, evidence of thermal stratification is detected in any other piping system, then stratification will be evaluated. If it cannot be shown that the stresses in the pipe are low and that movement due to bowing is acceptable, then stratification will be treated as a design load. In general, if temperature differences between the top and bottom of the pipe are less than 50°F, it may be assumed design specification and stress reports need not be revised to include stratification. See Subsection 3.9.7.10 for COL license information.

The design life for the ABWR Standard Plant is 60 years. A 60 year design life is a requirement for all major plant components with reasonable expectation of meeting this design life. However, all plant operational components and equipment except the reactor vessel are designed to be replaceable, design life not withstanding. The design life requirement allows for refurbishment and repair, as appropriate, to assure the design life of the overall plant is achieved. In effect, essentially all piping systems, components and equipment are designed for a 60 year design life. Many of these components are classified as ASME Class 2 or 3 or Quality Group D. In the event any non-Class 1 components are subjected to cyclic loadings, including operating vibration loads and thermal transient effects, of a magnitude and/or duration so severe that the 60 year design life can be assured by required Code calculations. COL applicants will identify these components and either provide an appropriate analysis to demonstrate the required design life or provide designs to mitigate the magnitude or duration of the cyclic loads. Components excluded from this requirement are (1) tees where mixing of hot and cold fluids occurs and thermal sleeves have been provided in accordance with the P&IDs, (2) components, such as the quencher, for which a fatigue analysis has already been performed, providing the com-

COL 14.1.3.3.2.1-1
14.1.3.3.9.2-1

Pipe support base plate flexibility will be accounted for in the calculation of concrete anchor bolt loads, in accordance with IE Bulletin 79-02, Revision 2.

3.9.3.4 Component Supports

The design of bolts for component supports is specified in the ASME Code Section III, Subsection NF. Stress limits for bolts are given in NF-3225. The rules and stress limits which must be satisfied are those given in NF-3324.6 multiplied by the appropriate stress limit factor for the particular service loading level and stress category specified in Table NF-3225.2-1.

Moreover, on equipment which is to be, or may be, mounted on a concrete support, sufficient holes for anchor bolts are provided to limit the anchor bolt stress to less than 10,000 psi on the nominal bolt area in shear or tension.

Concrete anchor bolts (including under-cut type anchor bolts) which are used for pipe support base plates will be designed to the applicable factors of safety which are defined in I&E Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," Revision 2 dated November 8, 1979.

See Subsection 3.9.7.7 for COL license information

3.9.3.4.1 Piping

Supports and their attachments for essential ASME Code Section III, Class 1,2, and 3 piping are designed in accordance with Subsection NF* up to the interface of the building structure, with jurisdictional boundaries as defined by Subsection NF. The loading combinations for the various operating conditions correspond to those used for design of the supported pipe. The component loading combinations are discussed in Subsection 3.9.3.1. The stress limits are per ASME III, Subsection NF and Appendix F. Supports are generally designed either by load rating method per paragraph NF-3260 or by the stress limits for linear supports per paragraph NF-3231. The critical buckling loads for the Class 1 piping supports subjected to faulted loads that are more severe than normal, upset and emergency loads, are determined by using the methods discussed in Appendices F and XVII of the Code. To avoid buckling in the piping supports, the allowable loads are limited to two thirds of the determined critical buckling loads.

Maximum calculated static and dynamic deflections at support locations are checked to confirm that the support has not rotated beyond the vendor's recommended cone of action or the recommended arc of loading.

Suppoprts for ASME Code Section III instrumentation lines are designed and analyzed in accordance with ASME Code Section III, Subsection NF

The design of all supports for non-nuclear piping satisfies the requirements of ANSI\ASME B31.1 Power Piping Code, Paragraphs 120 and 121.

For the major active valves identified in Subsection 3.9.3.2.4, the valve operators are not used as attachment points for piping supports.

The design criteria and dynamic testing requirements for the ASME III Subsection NF piping supports are as follows:

(1) Piping Supports - All piping supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they have been installed. All piping supports are designed in accordance with the rules of Subsection NF of the ASME Code up to the building structure interface as defined by the jurisdictional boundaries in Subsection NF.

(2) Spring Hangers - The operating load on spring hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the operating load at both their hot and cold load settings. Spring hangers provide a specified down travel and up travel in excess of the specified thermal movement. Deflections

*Augmented by the following: (1) application of Code Case N-476, Supplement 89.1 which governs the design of single angle members of ASME Class 1,2,3 and MC linear component supports; and (2) when eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center or meet "Standard for Steel Support Design", analyses will be performed in accordance with torsional analysis methods such as: "Torsional Analysis of Steel Members, USS Steel Manual", Publication T114-2/83.

COL 14.1.3.3.6.4-2

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 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)~~

INSERT A

3.9.8 References

INSERT B

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,

INSERT A

COL Action Item No. 14.1.3.3.2.1-1

The COL ~~A~~pplicant shall ensure that the piping system design is consistent with the construction practices, including inspection and examination methods, of the ASME Code edition and addenda as endorsed in 10 CFR 50.55a in effect at the time of application.

The COL ~~A~~pplicant shall identify ASME Code editions and addenda other than those listed in Tables 1.8-21 and 3.2-3, that will be used to design ASME Code Class 1,2 and 3 pressure retaining components and supports. The applicable portions of the ASME Code editions and addenda shall be identified to the NRC staff for review and approval with the COL application.

(see Subsection 3.9.3.1)

INSERT B

3.9.7.5 ASME Class 1,2 and 3 Piping System Clearance Requirements

ASME Class 1,2 and 3 piping systems shall be designed to provide clearance from structures, systems, and components where necessary for the accomplishment of the structure, system, or component's safety function as specified in the respective structure or system design description. The COL applicant ~~licensee~~ shall verify that the maximum calculated piping system deflections under service conditions do not exceed the minimum clearances between the piping system and nearby structures, systems, or components. The COL licensee shall document in the certified design stress report that the clearance requirements have been met. (See Subsection 3.9.3.1)

3.9.7.6 As-Built Reconciliation Analysis For ASME Class 1,2 and 3 Piping Systems

For ASME Class 1,2 and 3 piping systems, the COL applicant ~~licensee~~ shall reconcile the as-built piping system with the as-designed piping system. The COL applicant ~~licensee~~ will perform an as-built inspection of the pipe routing, location and orientation, the location, size, clearances and orientation of piping supports, and the location and weight of pipe mounted equipment. This inspection will be performed by reviewing the as-built drawings containing verification stamps, and by performing a visual inspection of the installed piping system. The piping configuration and component location, size, and orientation shall be within the tolerances specified in the certified as-built piping Stress Report. The tolerances to be used for reconciliation of the as-built piping system with the as-designed piping system are provided in the EPRI report, "Guidelines for Piping System Reconciliation (NCIG-05, Revision 1)," NP-5639 dated May 1988. A reconciliation analysis using the as-built and as-designed information shall be performed. The certified as-built Stress Report shall document the results of the as-built reconciliation analysis. (see Subsection 3.9.3.1)

⟨The above 2 items are provided in support of piping ITAAC/DAC. Reviewed during the January 1993 ITAAC meetings.⟩

COL 3.9.3.3-1 &
COL 14.1.3.3.6.4-1

3.9.7.7 Pipe Support Baseplate and Anchor Bolt Design

COL ~~A~~pplicants shall provide justification for the use of safety factors for concrete anchor bolts other than those specified in Subsection 3.9.3.4. This justification shall be submitted to the NRC ~~staff~~ for review and approval prior to the installation of the concrete anchor bolts.

COL 14.1.3.3.6.4-2

COL ~~A~~pplicants shall account for pipe support base plate flexibility in the calculation of concrete anchor bolt loads in accordance with Subsection 3.9.3.4.

COL 14.1.3.3.9.2-1

3.9.7.8 Pipe-Mounted Equipment Allowable Loads

The COL ~~A~~pplicant shall inspect the piping design reports and document that the pipe applied loads on attached equipment; such as valves, pumps, tanks and heat exchangers, are less than the equipment vendor's specified allowable loads. (See Subsection 3.9.3.1)

COL 14.1.3.3.4.3-1

3.9.7.9 Benchmark Requirements for Computer Codes used to perform Piping Dynamic Analysis

The COL ~~A~~pplicant shall benchmark their computer code used for piping system dynamic analysis against the NRC Benchmark Problems for ABWR, defined in Reference 5. The results of the COL applicant's piping dynamic analysis shall be compared with the results of the Benchmark Problems provided in Reference 5. The piping results to be compared and evaluated and the acceptance criteria or range of acceptable values are specified in Reference 5. Any deviations from these values as well as justification for such deviations shall be documented and submitted by the COL applicant to the NRC staff for review and approval before initiating the final certified piping analysis. (See Subsections 3.9.1.2)

3.9.7.10 ASME Class 1, 2 and 3 Piping System Design Requirements for Thermal Stratification of Fluids

COL applicants shall design for thermal stratification of fluids in accordance with Subsection 3.9.3.1.

14.1.3.3.5.10-1

CO'. 10.3.1-1

10.3 Main Steam Supply System

The function of the main steam supply system is to convey steam generated in the reactor to the turbine plant. This section discusses that portion of the main steam system which ranges between, but does not include, the outermost containment isolation valves and the turbine stop valves.

The main steam line pressure relief system, main steam line flow restrictors, main steam line isolation valves (MSIVs), and main steam piping from the reactor nozzles through the outboard main steam isolation valve (MSIV) are described in Subsections 5.2.2, 5.4.4, 5.4.5, and 5.4.9 respectively.

10.3.1 Design Bases

10.3.1.1 Safety Design Bases

The main steam supply system is not required to effect or support safe shutdown of the reactor or to perform in the operation of reactor safety features, however, the main steam supply system is designed:

(1) To comply with applicable codes and standards in order to accommodate operational stresses such as internal pressure and dynamic loads without risk of failures and consequential releases of radioactivity in excess of the established regulatory limits;

(2) To accommodate normal and abnormal environmental conditions; and

(3) With suitable accesses to permit inservice testing and inspections.

10.3.1.2 Power Generation Design Bases

Power Generation Design Basis One - The system is designed to deliver steam from the reactor to the turbine-generator system for a range of flows and pressures varying from warmup to rated conditions. It also provides steam to the reheaters, the steam jet air ejectors, the turbine gland sealing and the deaerating section of the main condenser and the turbine bypass system.

10.3.2 Description

10.3.2.1 General Description

The main steam supply system is illustrated in Figure 10.3-1. The main steam piping consists of four 28-inch diameter lines from the outboard main steam line isolation valves to the main turbine stop valves. The four main steam lines are connected to a header upstream of the turbine stop valves to permit testing of the main steam line isolation valves during plant operation with a minimum load reduction. This header arrangement is also provided to ensure that the turbine bypass and other main steam supplies are connected to operating steam lines and not to idle lines. The main steam process downstream of the turbine stop valves is illustrated in Figure 10.3-2.

The design pressure and temperature of the main steam piping is 1250 psig and 600 ^{o}F, respectively, the same values as the design parameters of the reactor. The main steam lines are classified as discussed in Section 3.2.

A drain line is connected to the low points of each main steam line, both inside and outside the containment. Both sets of drains are headered and connected, with isolation valves to allow drainage to the main condenser. To permit intermittent draining of the steam line low points at low loads, orificed lines are provided around the final valve to the main condenser. The steam line drains maintain a continuous downward slope from the steam system low points to the orifice located near the condenser. The drain line from the orifice to the condenser also slopes downward. To permit emptying the drain lines for maintenance, drains are provided from the line low points, going to the radwaste system.

The drains from the steam lines inside containment are connected to the steam lines outside containment to permit equalizing pressure across the main steam line isolation valves during startup and following a steam line isolation.

10.3.2.2 Component Description

The main steam system lines are made of carbon steel and are sized for a normal steady state velocity of 150 feet per second, or less. The lines are designed to permit hydrotesting following construction and major repairs without addition of temporary pipe supports.

see subsection 10.3.7.2 for COL license information pertaining to allowable MSIV leakage.

welds in locations of restricted direct physical and visual accessibility.

(a) The performance qualification should require testing of the welds when conditions of accessibility to production welds are less than 30 to 35 cm (12-14 inches) in any direction from the joint.

(b) Requalification is required for different accessibility conditions or when other essential variables listed in the Code, Section IX, are changed.

(c) The qualification and requalification tests required by (a) and (b) above may be waived provided that the joint is to be 100% radiographed or ultrasonically examined after completion of the weldment. Examination procedures and acceptance standards should meet the requirements of the ASME Code Section III. Records of the examination reports and radiographs should be retained and made part of the Quality Assurance documentation of the completed weld.

(2) Regulatory Guide 1.37, *Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants*, describes acceptable procedures for cleaning and handling Class 2 components of the steam and feedwater systems. Vented tanks with deionized or demineralized water are an acceptable source of water for final cleaning or flushing of finished surfaces. The oxygen content of the water in these vented tanks need not be controlled.

(3) Acceptance criteria for nondestructive examination of tubular products are given in the ASME Code, Section III, paragraphs NC 2550 through 2570.

10.3.7 COL License Information

10.3.7.1 Procedures to Avoid Steam Hammer and Discharge Loads

The COL applicant will provide operating and maintenance procedures that include adequate precautions to avoid steam hammer and discharge loads (see Subsection 10.3.3).

10.3.7.2 MSIV Leakage

The COL applicant will provide the amount of allowable MSIV leakage for review by the NRC. (See Subsection 10.3.2)

6.7 HIGH PRESSURE NITROGEN GAS SUPPLY SYSTEM

6.7.1 Functions

The high pressure nitrogen gas supply system is divided into two independent divisions, with each division containing a safety-related emergency stored nitrogen supply. The essential stored nitrogen supply is Safety Class 3, Seismic Category I, designed for operation of the main steam S/R valve ADS function accumulators.

The function of the nonsafety-related, makeup nitrogen gas supply system is:

(1) relief function accumulators of main steam S/R valves,

(2) pneumatically operated valves and instruments inside the PCV,

(3) leak detection system radiation monitor calibration

(4) ADS function accumulators to compensate for the leakage from main steam S/R solenoid valves during normal operation

6.7.2 System Description

Normally, nitrogen gas for both the essential and nonessential makeup systems is supplied from the nitrogen gas evaporator via the makeup line to the atmospheric control (AC) system. The nitrogen supply system shall supply nitrogen which is oil-free with a moisture content of less than 2.5 ppm. This nitrogen is filtered in the HPIN system to remove particles larger than 5 microns. All equipment using this nitrogen shall be capable of operating with nitrogen of the quality listed above. If nitrogen is not available from the AC system to supply the essential system, nitrogen is supplied from high pressure nitrogen gas storage bottles. The essential system is separated into two divisions. There are tielines between the nonessential and each division of the essential system. Each tieline has a motor operated shutoff valve. For details, see Figure 6.7-1 and Table 6.7-1.

Each division of the essential system has ten bottles. Normally, outlet valves from five of the ten bottles are kept open. Each division has a pressure control valve to depressurize the nitrogen gas from the bottles.

The bottles are mechanically restrained to preclude generation of high-pressure missiles during an SSE. The bottles are also covered by a heavy steel plate, which serves as a barrier to potential missiles.

Flow rate and capacity requirements are divided into an initial requirement and a continuous supply. An initial requirement for each ADS SRV provides for actuations of the valve against drywell pressure. Fifty gallon accumulators supplied for each main steam ADS SRV actuator fulfill the steam valve requirement. The continuous supply is divided into safety and nonsafety portions.

Compressed nitrogen at a rate adequate to make up the nitrogen leakage of each serviced valve is provided by the safety portion. This assumes an air leakage rate for each valve of 1 scfh for a period of at least seven days. The essential system with associated lines, valves and fittings are classified as Safety Class 3, Seismic Category I.

The nonsafety portion provides compressed nitrogen at a rate adequate to recharge the ADS SRV accumulators. The nonessential system has two pressure control valves to depressurize the nitrogen gas from the AC system. One is to depressurize to 200 psi for the SRV accumulators and the other is to depressurize to 100 psi for other pneumatic uses.

The continuous supply portion of the pneumatic system, extending from the AC system to the isolation valve prior to the essential system is not safety related.

Nonsafety piping and valves of the system are designed to ANSI B31.1, Power Piping Code, and the requirements of Quality Group D of Regulatory Guide 1.26. Pressure vessels and heat exchangers are designed to ASME Section VIII, Division I.

System design pressure ~~is 20~~ ~~the~~ temperature
~~system design temperature at 15~~
are shown in Figure 6.7.

Category I, ASME Code III, Class 3, Quality Group C and Quality Assurance B requirements, except for the piping and valves for the containment and drywell penetrations which are designed to Seismic Category I, ASME Code III, Class 2, Quality Group B and Quality Assurance B requirements.

The essential high pressure nitrogen gas supply is separated into two independent divisions, with each division capable of supplying 100% of the requirements of the division being serviced. Each division is mechanically and electrically separated from the other. The system satisfies the components' nitrogen demands during all plant operation conditions (normal through faulted).

Safety grade portions of the high pressure nitrogen gas supply system are capable of being isolated from the nonsafety parts and retaining their function during LOCA and/or seismic events under which any nonsafety parts may be damaged.

Pipe routing of Division 1 and Division 2 nitrogen gas is kept separated by enough space so that a single fire, equipment dropping accident, strike from a single high energy whipping pipe, jet force from a single broken pipe, internally generated missile or wetting equipment with spraying water cannot prevent the other division from accomplishing its safety function. Separation is accomplished by spatial separation or by a reinforced concrete barrier, to ensure separation of each pneumatic air division from any systems and components which belong to the other pneumatic air division.

6.7.4 Inspection and Testing Requirements

Periodic inservice inspection of components, in accordance with ASME Section XI, to ensure the capability and integrity of the system is mandatory. Nitrogen quality shall be tested periodically to assure compliance with ANSI MC11.1.

The nitrogen isolation valves are capable of being tested to assure their operational integrity by manual actuation of a switch located in the control room and by observation of associated position indication lights. Test and vent connections are provided at the containment isolation valves in order to verify their leaktightness. Operation of valves and associated equipment used to switch from the nonsafety to safety nitrogen supply can be tested to assure operational integrity by manual actuation of a switch located in the control room and by observation of associated position indication lights. Periodic tests of the check valves and accumulators shall be conducted to assure valve operability.

Periodic testing of the safety relief valves, the accumulator check valve, and the relief valve if present shall be conducted to confirm that the nitrogen leakage is within the assumed value of 28 liters per hour (1 scfh) for each safety relief valve.

6.7.5 Instrumentation Requirements

A pressure sensor is provided for the safety nitrogen supply, and an alarm signals low nitrogen pressure.

A remote manual switch and open-closed position lights are provided in the control room for valve operation and position indication.

OI 20.3-3

ation of the ADS. The results of this program were submitted to the NRC in a letter report from D. B. Waters, Chairman of BWR Owners' Group, to D. G. Eisenhut, Director (NRC), dated December 29, 1980. A summary of this evaluation follows.

The cases analyzed in the letter report above show that, based on core cooling considerations, no significant improvement can be achieved by a slower depressurization rate. A significantly slower depressurization will result in increased core uncovery times before ECCS injection. Furthermore, a moderate decrease in the depressurization rate necessitates an earlier action time to initiate ADS. Such an earlier actuation time has the negative impact of providing less time for the operator to start high pressure ECCS without obtaining a significant benefit to vessel fatigue usage. This earlier actuation time necessitates a higher initiation level which would result in an increased frequency of ADS actuation.

It should be noted that the ADS is not a normal core cooling system, but is a backup for the high pressure core cooling systems such as feedwater, RCIC or HPCF. If ADS operation is required, it is because normal and/or emergency core cooling is threatened. As a full ADS blowdown is well within the design basis of the RPV and the system is properly designed to minimize the threat to core cooling, no change in depressurization rate is required or appropriate.

19A.2.12 Evaluation of Alternative Hydrogen Control Systems [Item (1) (xii)]

NRC Position

Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f) (2) (ix) of 10CFR50.34(f). As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include:

(A) A comparison of costs and benefits of the alternative systems considered.

(B) For the selected system, analyses and test data to verify compliance with the requirements of (f) (2) (ix) of 10CFR50.34.

(C) For the selected system, preliminary design descriptions of equipment, function, and layout.

Response

The ABWR primary containment is inerted and is, therefore, protected from hydrogen combustion regardless of the amount or rate of hydrogen generation. In fact, increasing amounts of hydrogen moves the primary containment oxygen concentration further from the flammable regime. The ABWR is also provided with permanently-installed recombiners which prevent the buildup of oxygen, due to radiolysis, from creating a potentially flammable mixture. Radiolysis is the only potential source of oxygen in the ABWR primary containment.

¶ See Subsection 6.2.7.1 for COL license information pertaining to alternate hydrogen control.

19A.2.13 Long-Term Training Upgrade [Item (2) (i)]

NRC Position

Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCA's. (Applicable to construction permit applicants only.) [I.A.4.2]

Response

COL license information, see Subsection 19A.3.1.

19A.2.14 Long-Term Program of Upgrading of Procedures [Item (2) (ii)

NRC Position

Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (Applicable to construction permit applicants only.) [I.C.9]

Response

COL license information, see Subsection 19A.3.2.

19A.2.15 Control Room Design Reviews [Item (2) (iii)]

NRC Position

Provide, for Commission review, a control room design that reflects state-of-the-art human factor prin-

2.2.11 Process Computer System

Design Description

The Process Computer System (PCS) is classified as a non-safety-related system and has no functional safety design basis; however, it is designed so that the functional capabilities of safety-related systems are not affected by it. Input data for the PCS are derived from both Class 1E and non-Class 1E sources. Division to division and safety you non-safety PCS interfaces are made up of fiber optic cables, which act as optical isolators for electrical separation.

The purpose of the PCS is to:

(1) perform the functions and calculations for the evaluation of nuclear power plant operation;

(2) provide the capability for supervisory control of the plant by supplying setpoint commands to independent automatic control systems as changing load demands and plant conditions dictate;

(3) provide a permanent record and historical perspective for plant operating activities and abnormal events;

(4) provide analysis, evaluation and recommendation capabilities for start-up, normal operation, safe plant shutdown and abnormal operating and emergency conditions;

(5) provide capability to monitor plant performance through presentation of video displays in the main control room and elsewhere throughout the plant;

(6) provide the ability to directly control certain non-safety-related plant equipment through on-screen technology and

(7) provide a plant simulator for training and for development and analysis of operational techniques.

The calculations performed by the PCS include process validation and conversion, nuclear system supply performance calculations and balance-of-plant performance calculations.

Failures of process input signals are isolated and identified by the process computer.

CNFM 7.7.1.5-1 INSERT A →

The power for the PCS is supplied from a redundant, constant voltage-constant frequency uninterruptible, non-Class 1E power supply.

The PCS consists of two subsystems, Performance Monitoring and Control (PMC) and Power Generation Control (PGC).

Performance Monitoring and Control System

PMC~~S~~ System provides nuclear steam supply (NSS) performance and prediction calculations, video display control, point log and alarm processing and balance of plant (BOP) performance calculations.

The NSS Performance module of PMC takes reactor and in-core data from a plant data acquisition system and procedures current state and predicted core performance information. NSS performance calculations are done to provide three- dimensional simulation ~~of BWR performance~~ of its performance.

Power Generation Control System

PGC monitors the overall plant conditions, exercises the algorithms for the automated control sequences associated with plant power range operation and issues reactor command signals to the automatic power regulator (APR) system, which implements them.

In the event that conditions which are not expected during normal plant operations occur in the plant or in the PGC, PGC reverts to the manual mode.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.11 provides a definition of the inspections, tests, and/or analyses together with associated acceptance criteria which will be undertaken for the PCS.

Table 2.2.11: Process Computer System

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Test, Analysis	Acceptance Criteria
1. The Process Computer System identifies and isolates failure of ~~process~~ input signals.	1. Tests will be performed by simulating input signal failures to the Process Computer System.	1. The Process Computer System output signal is based upon the remaining valid input signals and failed inputs will be identified.
~~2. The Process Computer System is powered by redundant, constant voltage-constant frequency uninterruptible, non-Class 1E power sources.~~	~~2. A test shall be performed by simulating failure of each redundant, constant voltage-constant frequency power supply, one at a time, power supply to the Process Computer System.~~	~~2. There is no loss of process computer functions during the loss of one power supply.~~
~~3.~~ 2 In the event that conditions which are not expected during plant operations develop in the plant or trouble occurs in the Process Computer System, PGC automatically decouples from the plant control circuits and plant operation reverts to the manual mode.	~~3.~~ 2 The system shall be tested by: a. Simulating communications failure with one low level controller. b. Simulating failure within PCS resulting in a loss of a PCS function. c. Simulating a plant transient.	~~3.~~ 2 Following the simulated failure, the PGC decouples and plant operation reverts to manual mode.

INSERT "B"

INSERT B

~~Table 2.2.11: Process Computer System~~
~~Inspections, Tests, Analyses and Acceptance Criteria~~

~~Certified Design Commitment~~	~~Inspections, Tests, Analyses~~	~~Acceptance Criteria~~
3 ~~1.~~ The Process Computer shall log both the tripped and reset conditions of the RPS-related sensor instrument channels and the RPS automatic or manual conditions of the RPS-related sensor instrument channels and the RPS automatic or manual trip systems. The computer shall identify the specific trip variable, divisional channel identity and specific automatic or manual trip system for all conditions that cause a reactor trip.	3 ~~1.~~ The system shall be tested by: a. Simulating trips and resets of the RPS-related sensor instrument channels. b. Simulating RPS automatic or manual conditions of the RPS-related sensor instrument channels and the RPS automatic or manual trip systems.	3 ~~1.~~ The process computer logs the tested conditions and the logs contain the specific trip variable, the divisional channel identity and the specific automatic or manual trip system for all conditions simulated.

INSERT "A" → The process computer also logs both the tripped and reset conditions of the RPS-related sensor instrument channels and the RPS automatic or manual conditions of the RPS-related sensor instrument channels and the RPS automatic or manual systems. The logs contain the identity of the specific trip variables, the divisional channels and the specific automatic or manual trip system for all conditions that cause a reactor trip.

10.3 Main Steam Supply System

The function of the main steam supply system is to convey steam generated in the reactor to the turbine plant. This section discusses that portion of the main steam system which ranges between, but does not include, the outermost containment isolation valves and the turbine stop valves.

The main steam line pressure relief system, main steam line flow restrictors, main steam line isolation valves (MSIVs), and main steam piping from the reactor nozzles through the outboard main steam isolation valve (MSIV) are described in Subsections 5.2.2, 5.4.4, 5.4.5, and 5.4.9 respectively.

10.3.1 Design Bases

10.3.1.1 Safety Design Bases

The main steam supply system is not required to effect or support safe shutdown of the reactor or to perform in the operation of reactor safety features, however, the main steam supply system is designed:

(1) To comply with applicable codes and standards in order to accommodate operational stresses such as internal pressure and dynamic loads without risk of failures and consequential releases of radioactivity in excess of the established regulatory limits;

(2) To accommodate normal and abnormal environmental conditions; and

(3) With suitable accesses to permit inservice testing and inspections.

10.3.1.2 Power Generation Design Bases

Power Generation Design Basis One - The system is designed to deliver steam from the reactor to the turbine-generator system for a range of flows and pressures varying from warmup to rated conditions. It also provides steam to the reheaters, the steam jet air ejectors, the turbine gland sealing and the deaerating section of the main condenser and the turbine bypass system.

10.3.2 Description

10.3.2.1 General Description

The main steam supply system is illustrated in Figure 10.3-1. The main steam piping consists of four 28-inch diameter lines from the outboard main steam line isolation valves to the main turbine stop valves. The four main steam lines are connected to a header upstream of the turbine stop valves to permit testing of the main steam line isolation valves during plant operation with a minimum load reduction. This header arrangement is also provided to ensure that the turbine bypass and other main steam supplies are connected to operating steam lines and not to idle lines. The main steam process downstream of the turbine stop valves is illustrated in Figure 10.3-2.

The design pressure and temperature of the main steam piping is 1250 psig and 600 $^{\circ}$F, respectively, the same values as the design parameters of the reactor. The main steam lines are classified as discussed in Section 3.2.

A drain line is connected to the low points of each main steam line, both inside and outside the containment. Both sets of drains are headered and connected, with isolation valves to allow drainage to the main condenser. To permit intermittent draining of the steam line low points at low loads, orificed lines are provided around the final valve to the main condenser. The steam line drains maintain a continuous downward slope from the steam system low points to the orifice located near the condenser. The drain line from the orifice to the condenser also slopes downward. To permit emptying the drain lines for maintenance, drains are provided from the line low points, going to the radwaste system.

The drains from the steam lines inside containment are connected to the steam lines outside containment to permit equalizing pressure across the main steam line isolation valves during startup and following a steam line isolation.

10.3.2.2 Component Description

The main steam system lines are made of carbon steel and are sized for a normal steady state velocity of 150 feet per second, or less. The lines are designed to permit hydrotesting following construction and major repairs without addition of temporary pipe supports.

see subsection 10.3.7.2 for COL license information pertaining to allowable MSIV leakage.

welds in locations of restricted direct physical and visual accessibility.

(a) The performance qualification should require testing of the welds when conditions of accessibility to production welds are less than 30 to 35 cm (12-14 inches) in any direction from the joint.

(b) Requalification is required for different accessibility conditions or when other essential variables listed in the Code, Section IX, are changed.

(c) The qualification and requalification tests required by (a) and (b) above may be waived provided that the joint is to be 100% radiographed or ultrasonically examined after completion of the weldment. Examination procedures and acceptance standards should meet the requirements of the ASME Code Section III. Records of the examination reports and radiographs should be retained and made part of the Quality Assurance documentation of the completed weld.

(2) Regulatory Guide 1.37, *Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants*, describes acceptable procedures for cleaning and handling Class 2 components of the steam and feedwater systems. Vented tanks with deionized or demineralized water are an acceptable source of water for final cleaning or flushing of finished surfaces. The oxygen content of the water in these vented tanks need not be controlled.

(3) Acceptance criteria for nondestructive examination of tubular products are given in the ASME Code, Section III, Paragraphs NC 2550 through 2570.

10.3.7 COL License Information

10.3.7.1 Procedures to Avoid Steam Hammer and Discharge Loads

The COL applicant will provide operating and maintenance procedures that include adequate precautions to avoid steam hammer and discharge loads (see Subsection 10.3.3).

10.3.7.2 MSIV Leakage

The COL applicant will provide the amount of allowable MSIV leakage for review by the NRC. (See Subsection 10.3.2)

6.7 HIGH PRESSURE NITRO GAS SUPPLY SYSTEM

6.7.1 Functions

The high pressure nitrogen gas supply system is divided into two independent divisions, with each division containing a safety-related emergency stored nitrogen supply. The essential stored nitrogen supply is Safety Class 3, Seismic Category I, designed for operation of the main steam S/R valve ADS function accumulators.

The function of the nonsafety-related, makeup nitrogen gas supply system is:

(1) relief function accumulators of main steam S/R valves,

(2) pneumatically operated valves and instruments inside the PCV,

(3) leak detection system radiation monitor calibration

(4) ADS function accumulators to compensate for the leakage from main steam S/R solenoid valves during normal operation

6.7.2 System Description

Normally, nitrogen gas for both the essential and nonessential makeup systems is supplied from the nitrogen gas evaporator via the makeup line to the atmospheric control (AC) system. The nitrogen supply system shall supply nitrogen which is oil-free with a moisture content of less than 2.5 ppm. This nitrogen is filtered in the HPIN system to remove particles larger than 5 microns. All equipment using this nitrogen shall be capable of operating with nitrogen of the quality listed above. If nitrogen is not available from the AC system to supply the essential system, nitrogen is supplied from high pressure nitrogen gas storage bottles. The essential system is separated into two divisions. There are tielines between the nonessential and each division of the essential system. Each tieline has a motor operated shutoff valve. For details, see Figure 6.7-1 and Table 6.7-1.

Each division of the essential system has ten bottles. Normally, outlet valves from five of the ten bottles are kept open. Each division has a pressure control valve to depressurize the nitrogen gas from the bottles.

The bottles are mechanically restrained to preclude generation of high-pressure missiles during an SSE. The bottles are also covered by a heavy steel plate, which serves as a barrier to potential missiles.

Flow rate and capacity requirements are divided into an initial requirement and a continuous supply. An initial requirement for each ADS SRV provides for actuations of the valve against drywell pressure. Fifty gallon accumulators supplied for each main steam ADS SRV actuator fulfill the steam valve requirement. The continuous supply is divided into safety and nonsafety portions.

Compressed nitrogen at a rate adequate to make up the nitrogen leakage of each serviced valve is provided by the safety portion. This assumes an air leakage rate for each valve of 1 scfh for a period of at least seven days. The essential system with associated lines, valves and fittings are classified as Safety Class 3, Seismic Category I.

The nonsafety portion provides compressed nitrogen at a rate adequate to recharge the ADS SRV accumulators. The nonessential system has two pressure control valves to depressurize the nitrogen gas from the AC system. One is to depressurize to 200 psi for the SRV accumulators and the other is to depressurize to 100 psi for other pneumatic uses.

The continuous supply portion of the pneumatic system, extending from the AC system to the isolation valve prior to the essential system is not safety related.

Nonsafety piping and valves of the system are designed to ANSI B31.1, Power Piping Code, and the requirements of Quality Group D of Regulatory Guide 1.26. Pressure vessels and heat exchangers are designed to ASME Section VIII, Division I.

System design pressure ~~is 200 psig with the system design temperature at 150°F.~~ and temperature are shown in Figure 6.7-1.

Category I, ASME Code III, Class 3, Quality Group C and Quality Assurance B requirements, except for the piping and valves for the containment and drywell penetrations which are designed to Seismic Category I, ASME Code III, Class 2, Quality Group B and Quality Assurance B requirements.

The essential high pressure nitrogen gas supply is separated into two independent divisions, with each division capable of supplying 100% of the requirements of the division being serviced. Each division is mechanically and electrically separated from the other. The system satisfies the components' nitrogen demands during all plant operation conditions (normal through faulted).

Safety grade portions of the high pressure nitrogen gas supply system are capable of being isolated from the nonsafety parts and retaining their function during LOCA and/or seismic events under which any nonsafety parts may be damaged.

Pipe routing of Division 1 and Division 2 nitrogen gas is kept separated by enough space so that a single fire, equipment dropping accident, strike from a single high energy whipping pipe, jet force from a single broken pipe, internally generated missile or wetting equipment with spraying water cannot prevent the other division from accomplishing its safety function. Separation is accomplished by spatial separation or by a reinforced concrete barrier, to ensure separation of each pneumatic air division from any systems and components which belong to the other pneumatic air division.

6.7.4 Inspection and Testing Requirements

Periodic inservice inspection of components, in accordance with ASME Section XI, to ensure the capability and integrity of the system is mandatory. Nitrogen quality shall be tested periodically to assure compliance with ANSI MC11.1.

The nitrogen isolation valves are capable of being tested to assure their operational integrity by manual actuation of a switch located in the control room and by observation of associated position indication lights. Test and vent connections are provided at the containment isolation valves in order to verify their leaktightness. Operation of valves and associated equipment used to switch from the nonsafety to safety nitrogen supply can be tested to assure operational integrity by manual actuation of a switch located in the control room and by observation of associated position indication lights. Periodic tests of the check valves and accumulators shall be conducted to assure valve operability.

6.7.5 Instrumentation Requirements

A pressure sensor is provided for the safety nitrogen supply, and an alarm signals low nitrogen pressure.

A remote manual switch and open-closed position lights are provided in the control room for valve operation and position indication.

Periodic testing of the safety relief valves, the accumulator check valve, and the relief valve if present shall be conducted to confirm that the nitrogen leakage is within the assumed value of 28 liters per hour (1 scfh) for each safety relief valve.

ation of the ADS. The results of this program were submitted to the NRC in a letter report from D. B. Waters, Chairman of BWR Owners' Group, to D. G. Eisenhut, Director (NRC), dated December 29, 1980. A summary of this evaluation follows.

The cases analyzed in the letter report above show that, based on core cooling considerations, no significant improvement can be achieved by a slower depressurization rate. A significantly slower depressurization will result in increased core uncovery times before ECCS injection. Furthermore, a moderate decrease in the depressurization rate necessitates an earlier action time to initiate ADS. Such an earlier actuation time has the negative impact of providing less time for the operator to start high pressure ECCS without obtaining a significant benefit to vessel fatigue usage. This earlier actuation time necessitates a higher initiation level which would result in an increased frequency of ADS actuation.

It should be noted that the ADS is not a normal core cooling system, but is a backup for the high pressure core cooling systems such as feedwater, RCIC or HPCF. If ADS operation is required, it is because normal and/or emergency core cooling is threatened. As a full ADS blowdown is well within the design basis of the RPV and the system is properly designed to minimize the threat to core cooling, no change in depressurization rate is required or appropriate.

19A.2.12 Evaluation of Alternative Hydrogen Control Systems [Item (1) (xii)]

NRC Position

Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f) (2) (ix) of 10CFR50.34(f). As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include:

(A) A comparison of costs and benefits of the alternative systems considered.

(B) For the selected system, analyses and test data to verify compliance with the requirements of (f) (2) (ix) of 10CFR50.34.

(C) For the selected system, preliminary design descriptions of equipment, function, and layout.

Response

The ABWR primary containment is inerted and is, therefore, protected from hydrogen combustion regardless of the amount or rate of hydrogen generation. In fact, increasing amounts of hydrogen moves the primary containment oxygen concentration further from the flammable regime. The ABWR is also provided with permanently-installed recombiners which prevent the buildup of oxygen, due to radiolysis, from creating a potentially flammable mixture. Radiolysis is the only potential source of oxygen in the ABWR primary containment.

(a) See Subsection 6.2.7.1 for COL license information pertaining to alternate hydrogen control.

19A.2.13 Long-Term Training Upgrade [Item (2) (i)]

NRC Position

Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCA's. (Applicable to construction permit applicants only.) [I.A.4.2]

Response

COL license information, see Subsection 19A.3.1.

19A.2.14 Long-Term Program of Upgrading of Procedures [Item (2) (ii)

NRC Position

Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (Applicable to construction permit applicants only.) [I.C.9]

Response

COL license information, see Subsection 19A.3.2.

19A.2.15 Control Room Design Reviews [Item (2) (iii)]

NRC Position

Provide, for Commission review, a control room design that reflects state-of-the-art human factor prin-

these lines result in no significant safety consideration. All of the lines terminate below the minimum drawdown level in the suppression pool.

The test return lines are also used for suppression pool return flow during other modes of operation. In this manner, the number of penetrations are reduced, thus minimizing the potential pathways for radioactive material release. Typically, pump minimum flow bypass lines join the respective test return lines downstream of the test return isolation valve. The bypass lines are isolated by motor-operated valves in series with a restricting orifice.

6.2.4.3.2.2.1.2 RCIC Turbine Exhaust and Pump Minimum Flow Bypass Lines

The RCIC turbine exhaust line which penetrates the containment and discharges to the suppression pool is equipped with a normally open, motor-operated, remote-manually actuated gate valve located as close to the containment as possible. In addition, there is a simple check valve upstream of the gate valve which provides positive actuation for immediate isolation in the event of a break upstream of this valve. The gate valve in the RCIC turbine exhaust is designed to be locked open in the control room and is interlocked to preclude opening of the inlet steam valve to the turbine until the turbine exhaust valve is in its full open position. The RCIC pump minimum flow bypass line is isolated by a normally closed, remote manually actuated valve outside containment.

6.2.4.3.2.2.1.3 SPCU Discharge Line

The suppression pool cleanup (SPCU) system discharge line to the suppression pool (i.e., containment penetration, piping and isolation valves) is designed to Seismic Category I, ASME Section III, Class 2 requirements.

6.2.4.3.2.2.2 Effluent Lines from Suppression Pool

OPEN 6.2.4.1-1 20.3-4 Figure 6.2-38 identifies the isolation provisions in the effluent lines from the suppression pool.

6.2.4.3.2.2.2.1 RHR, RCIC and HPCF Lines

The RHR, RCIC, and HPCF suction lines contain motor-operated, remote-manually actuated gate valves which provide assurance of isolating these lines in the event of a break. These valves also provide long-term leakage control. In addition, the suction piping from the suppression pool must be available for long-term usage following a design basis LOCA, and, as such, is designed to the quality standards commensurate with its importance to safety. The RHR discharge line fill system suction lines have manual valves for operational purposes. These systems are isolated from the containment by the respective RHR pump suction valves from suppression pool.

6.2.4.3.2.2.2.2 SPCU Suction Line

The SPCU system suction line has two isolation valves. However, because the penetration is under water, both the isolation valves are located outside containment. The first valve is located as close as possible to the containment, and the second is located to provide adequate separation from the first.

6.2.4.3.2.2.3 ACS Lines To Containment

The Atmospheric Control System (ACS) has both influent and effluent lines which penetrate the containment. Both isolation valves on these lines are outside of the containment vessel to provide accessibility to the valves. The Inboard valve is are located as close as practical to the containment vessel. The piping to both valves is an extension of the containment boundary. INSERT 6.2.4.3.2.2.3

6.2.4.3.2.2.4 Conclusion on Criterion 56

In order to assure protection against the consequences of accidents involving release of significant amounts of radioactive materials, pipes that penetrate the containment have been demonstrated to provide isolation capabilities on a case-by-case basis in accordance with Criterion 56.

INSERT 6.2.4.3.2.2.3 Each containment penetration and piping extension is single failure proof. Environmental qualification of each safety related valve assures common mode failure potential is acceptably low. Similar piping and isolation is used in Susquehanna and Limerick BWR atmospheric control systems.

Influent and effluent lines of this group are isolated by automatic or remote-manual isolation valves located as close as possible to the containment boundary.

6.2.4.3.2.4 Evaluation Against Regulatory Guide 1.11

Instrument lines that connect to the RCPB and penetrated the containment have 1/4-inch orifices and manual isolation valves, in compliance with Regulatory Guide 1.11 requirements.

6.2.4.3.3 Evaluation of Single Failure

A single failure can be defined as a failure of a component (e.g., a pump, valve, or a utility such as offsite power) to perform its intended safety functions as a part of a safety system. The purpose of the evaluation is to demonstrate that the safety function of the system will be completed even with that single failure. Appendix A to 10CFR50 requires that electrical systems be designed specifically against a single passive or active failure. Section 3.1 describes the implementation of these standards as well as General Design Criteria 17, 21, 35, 38, 41, 44, 54, 55 and 56.

Electrical as well as mechanical systems are designed to meet the single-failure criterion, regardless of whether the component is required to perform a safety action. Even though a component, such as an electrically-operated valve, is not designed to receive a signal to change state (open or closed) in a safety scheme, it is assumed as a single failure if the system component changes state or fails. Electrically-operated valves include valves that are electrically piloted but air operated, as well as valves that are directly operated by an electrical device. In addition, all electrically-operated valves that are automatically actuated can also be manually actuated from the main control room. Therefore, a single failure in any electrical system is analyzed, regardless of whether the loss of a safety function is caused by a component failing to perform a requisite mechanical motion or a component performing an unnecessary mechanical motion.

6.2.4.4 Test and Inspections

The containment isolation system is scheduled to undergo periodic testing during reactor operation. The functional capabilities of power-operated isolation valves are tested remote-manually from the control room. By observing position indicators and changes in the affected system operation, the closing ability of a particular isolation valve is demonstrated.

Air-testable check valves are provided on influent emergency core cooling lines of the HPCF and RHR systems whose operability is relied upon to perform a safety function.

A discussion of testing and inspection of isolation valves is provided in Subsection 6.2.1.6. Instruments are periodically tested and inspected. Test and/or calibration points are supplied with each instrument. Leakage integrity tests shall be performed on the containment isolation valves with resilient material seals at least once every 3 months.

6.2.5 Combustible Gas Control in Containment

The atmospheric control system (ACS-T31) is provided to establish and maintain an inert atmosphere within the primary containment during all plant operating modes except during shutdown for refueling or equipment maintenance and during limited periods of time to permit access for inspection at low reactor power. The flammability control system (FCS-T49) is provided to control the potential buildup of oxygen from design-basis radiolysis of water. The objective of these systems is to preclude combustion of hydrogen and damage to essential equipment and structures. INSERT 6.2.5

OPEN 20.3-3

COL 6.2.5-1

6.2.5.1 Design Bases

Following are criteria that serve as the bases for design:

(1) Since there is no design requirement for the ACS or FCS in the absence of a LOCA and there is no design-basis accident in the ABWR that results in core uncovery or fuel failures, the following requirements mechanistically assume that a LOCA

INSERT 6.2.5

The COL applicant is required to provide a comparison of costs and benefits for any optional alternate systems of hydrogen control.

Included in the leak rate test summary report will be, a report detailing the containment inspection, a report detailing any repairs necessary to pas. .he tests, and the leak rate test results.

6.2.6.5 Special Testing Requirements

The maximum allowable leakage rate into the secondary containment and the means to verify that the inleakage rate has not been exceeded, as well as the containment leakage rate to the environment, are discussed in Subsections 6.2.3 and 6.5.1.3.

OPEN 20.3-3
COL 6.2.5-1

(change TOC)

OPEN 6.2.6-6

6.2.7 COL License Information

6.2.7.1 Alternate Hydrogen Control

A comparison of costs and benefits will be provided for alternate hydrogen control in accordance with Subsection 6.2.5.

6.2.7.2 Administrative Control Maintaining Containment Isolation

The COL applicant will maintain the primary containment boundary by administrative controls in accordance with Subsection 6.2.6.3.1

6.2.8 References

1. W.J. Bilanin, *The G.E. Mark III Pressure Suppression Containment Analytical Model*, June 1974, (NEDO-20533).

2. F.J. Moody, *Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels*, General Electric Company, Report No. NEDO-21052, September, 1975.

3. W.J. Bilanin, *The G.E. Mark III Pressure Suppression Containment Analytical Model*, Supplement 1, September 1975 (NEDO-20533-1).

4. J.P. Dougherty, SCAM-*Subcompartment Analysis Method*, January 1977, (NEDE-21526).

refueling floor ventilation exhaust, both SGTS trains are automatically operated. When the operation of both the trains is assured, one train is placed in standby mode. In the event a malfunction disables an operating train, the standby train is automatically initiated.

6.5.1.2.3.2 Manual

in accordance with the technical specifications

OPEN 6.5.1-2

The SGTS is on standby during normal plant operation, ~~and~~ It may be manually initiated ~~before or during~~ for primary containment purging (de-inerting) or when required to limit the discharge of contaminants to the environment within 10CFR20 limits. ~~It~~ a single train may be manually initiated for ~~testing or whenever its use may be needed to avoid exceeding radiation monitor setpoints.~~ surveillance testing.

6.5.1.2.3.3 Decay Heat Removal

Cooling of the SGTS filters may be required to prevent the gradual accumulation of decay heat in the charcoal. This heat is generated by the decay of radioactive iodine adsorbed on the SGTS charcoal. The charcoal is typically cooled by the air from the process fan.

A water deluge capability is also provided, but primarily for fire protection since redundant process fans are provided for air cooling. Since the deluge is available, it may also be used to remove decay heat for sequences outside the normal design basis. Temperature instrumentation is provided for control of the SGTS process and space electric heaters. This instrumentation may also be used by the operator to [re-]establish a cooling air flow post-accident, if required.

Water is supplied from the fire protection system and is connected to the SGTS via a spool piece.

6.5.1.3 Design Evaluation

6.5.1.3.1 General

(1) A slight negative pressure is normally maintained in the secondary containment by the reactor building HVAC system (Subsection 9.4.5). On SGTS initiation per Subsection 6.5.1.2.3.1, the secondary containment HVAC is automatically isolated.

(2) The SGTS filter particulate and charcoal efficiencies are outlined in Table 6.5-1. Dose analyses of events requiring SGTS operation, described in Subsections 15.6.5 and 15.7.4, indicate that offsite doses are within the limits established by 10 CFR 100.

(3) The SGTS is designated as an engineered safety feature since it mitigates the consequences of a postulated accident by controlling and reducing the release of radioactivity to the environment. The SGTS, except for the deluge, is designed and built to the requirements for Safety Class 3 equipment as defined in Section 3.2, and 10 CFR 50, Appendix B.

The SGTS has independent, redundant active trains. Should any active train fail, SGTS functions can be performed by the redundant train. ~~The electrical devices of independent components are~~ Each redundant train is powered from separate Class 1E electrical busses. CNFM 6.5.1-1

The two SGTS trains are mechanically and electrically separated. They are located in two side by side compartments (separated by rated fire barriers) adjacent to the HVAC system exhaust

(4) The SGTS is designed to Seismic Category I requirements as specified in Section 3.2. The SGTS is housed in a Category I structure. All surrounding equipment, components, and supports are designed to appropriate safety class and seismic requirements.

(5) A secondary containment draw-down analysis will be performed to demonstrate the capability of the SGTS to maintain the design negative pressure following a LOCA including inleakage from the open, non-isolated penetration lines identified during construction engineering and the vent of the worst single failure of a secondary isolation valve to close. (See Subsection 6.5.5.1 for COL license information requirements).

6.5.1.3.2 Sizing Basis

Figure 6.5-2 provides an assessment of the secondary containment pressure after the design-basis LOCA assuming an SGTS fan capacity of 6800 m^3/hr (21°C, 1 atmosphere) per fan. Credit for secondary containment as a fission product control system is only taken if the secondary containment is actually at a negative pressure by considering the potential effect of wind on the ambient pressure in the vicinity of the reactor building. For the ABWR dose analysis, direct transport of containment leakage to the environment

TABLE 6.5-1

STANDBY GAS TREATMENT SYSTEM COMPONENT DESCRIPTION

Filter Train	Consists of a moisture separator, an electric process heater, prefilter, pre-HEPA filter, charcoal adsorber, post-HEPA filter and space electric heaters.
Quantity	2
Capacity	6800 m^3/hr
Moisture Separator	
General	Woven wire, stainless steel mesh pads
Quantity	1 bank of standard size moisture separators per filter train
Efficiency	per ASME N509, Section 5.4
Electric Process Heater	
General	Electric, finned tubular
Quantity	1 per ~~dryer~~ filter train
Rating	5.3 kW minimum, 26.2 kW maximum
Relative humidity	
Inlet	* 100% @ 66°C
Outlet	70% @ 75°C
Air ΔT	9°C
Prefilter	
General	Cartridge type
Quantity	1 bank of standard size filters per filter train
Media	Glass fiber
Efficiency	Per ASME N509, Section 5.3

OPEN 6.5.1-1

* Capacity of the heater is sufficient to reduce the relative humidity to ≤ 70% at any temperature ≤ 66°C.

reactor coolant pressure boundary and the release of radioactive materials from either the reactor coolant pressure boundary or from the fuel and equipment storage pools.

7.1.1.3.3 Wetwell and Drywell Spray Mode of RHR

Instrumentation and control provides manual initiation of wetwell spray and manual initiation of drywell spray (when high drywell pressure signal is present) to condenses steam in the containment and remove heat from the containment. The drywell spray has an interlock such that drywell spray is possible only in the presence of a high drywell pressure condition.

7.1.1.3.4 Suppression Pool Cooling Mode of RHR (SPC-RHR)

Instrumentation and control is provided to manually initiate portions of the RHR system to effect cooling of the suppression pool water.

7.1.1.3.5 Standby Gas Treatment System

420.29 Instrumentation and Control is provided to maintain negative pressure in the secondary containment and for automatically limiting airborne radioactivity release from containment if required.

7.1.1.3.6 Emergency Diesel Generator Support Systems

Instrumentation and control is provided to assure availability of electric control and motive power under all design basis conditions. 420.29 The function of the diesel generator is to provide automatic emergency AC power supply for the safety-related loads (required for the safe shutdown of the reactor) when the offsite source of power is not available.

7.1.1.3.7 Reactor Building Cooling Water System

Instrumentation and control is provided to assure availability of cooling water for heat removal from the nuclear system as required. 420.29 Safety-related portions of this system start automatically on receipt of a LOCA and/or LOPP signal.

7.1.1.3.8 Essential HVAC Systems

Instrumentation and control is provided to automatically maintain an acceptable thermal environment for safety equipment and operating personnel. 420.29

7.1.1.3.9 HVAC Emergency Cooling Water System

Automatic instrumentation and control is provided to assure that adequate cooling is provided for the main control room, the control building essential electrical equipment rooms, and the diesel generator cooling coils. 420.29

7.1.1.3.10 High Pressure Nitrogen Gas Supply System

Automatic instrumentation and control is provided to assure adequate instrument high pressure nitrogen is available for ESF equipment operational support. 420.29

7.1.1.4 Safe Shutdown Systems

7.1.1.4.1 Alternate Rod Insertion Function (ARI)

Though not required for safety, instrumentation and controls for the ARI provide a function for mitigation of the consequences of anticipated transient without scram (ATWS) events. Upon receipt of an initiation signal (high reactor dome pressure or low reactor water level), the fine-motion control rod drive (FMCRD) motor shall automatically drive all rods full-in. This provides a method, diverse from the hydraulic control units (HCUs) for scramming the reactor. 420.29

7.1.1.4.2 Standby Liquid Control System (SLCS)

Instrumentation and controls are provided for the manual initiation of an independent backup system which can shut the reactor down from rated power to the cold condition in the event that all withdrawn control rods cannot be inserted to achieve reactor shutdown. In addition, should the FMCRD fail to shutdown the reactor during an ATWS event as described in subsection 7.1.1.4.1, then instrumentation and controls are provided for the automatic initiation of SLCS.

7.1.1.4.3 Residual Heat Removal (RHR) System / Shutdown Cooling Mode

Instrumentation and controls provide manual initiation of cooling systems to remove the decay and sensible heat from the reactor vessel.